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NMC
Committed to Nuclear Excellence

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U. S. Nuclear Regulatory Commission
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Ladies and Gentlemen:

DOCKETS 50-266 AND 50-301
2000 ANNUAL RESULTS AND DATA REPORT
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed is the 2000 Annual Results and Data Report for Point Beach Nuclear Plant, Units 1 and 2. This report is submitted in accordance with Technical Specification 15.6.9.1.B.

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 2000.

Sincerely,

A. J. Cayia
Plant Manager

dwd

Enclosure

cc: NRC Regional Administrator, Region III
NRC Resident Inspector

A001

NUCLEAR MANAGEMENT COMPANY, LLC

ANNUAL RESULTS AND DATA REPORT 2000

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
Independent Spent Fuel Storage Installation
Docket No. 72-005

PREFACE

This Annual Results and Data Report for 2000 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, and Docket 72-005 for Independent Spent Fuel Storage Installation (ISFSI).

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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, 2000, through December 31, 2000, included: A manual trip on January 21 because of low forebay water levels and icing in the intake structure; unit shutdown on February 20 to investigate an indication of loose parts in a steam generator (after a thorough investigation and inspection, no loose parts were found and the unit was restarted on March 5); the unit was manually tripped on October 27 as a result of loss of communications with a diver in the forebay (the diver was not injured and the unit was returned to 100% power on October 28).

Unit 1 operated at an average capacity factor of 92.3% (MDC net) and an electrical/thermal efficiency of 33.06%. Unit and reactor availability were 95.6% and 95.8%, respectively. Unit 1 generated its 99 billionth kilowatt hour on March 10, 2000; its 100 billionth kilowatt hour on May 29, 2000; its 101 billionth kilowatt hour on August 17, 2000; and its 102 billionth kilowatt hour on November 8, 2000.

UNIT 2

Highlights for the period January 1, 2000, through December 31, 2000, included: Unit 2 began a five-day maintenance outage on May 4, ending an online run of 353 days (200 cubic yards of zebra mussels were removed from the Unit 2 surge chamber, forebay and pumphouse); the unit was manually shut down to begin its twenty-fourth refueling/maintenance outage on October 13 (unexpected delays in returning Unit 2 to power caused several outage extensions); the unit tripped from 63% power on December 20 during recovery from the refueling outage (reactor trip was caused by a turbine trip at >50% reactor power).

Unit 2 operated at an average capacity factor of 78.4% (MDC net) and an electrical/thermal efficiency of 33.24%. Unit and reactor availability were 80.8% and 81.6%, respectively. Unit 2 generated its 98 billionth kilowatt hour on February 9, 2000; its 99 billionth kilowatt hour on April 28, 2000; its 100 billionth kilowatt hour on July 20, 2000; and its 101 billionth kilowatt hour on October 8, 2000.

III. AMENDMENTS TO FACILITY OPERATING LICENSES

During 2000 there were eight amendments issued by the U.S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach Nuclear Plant Unit 1. Eight amendments were issued to Facility Operating License DPR-27 for Point Beach Nuclear Plant Unit 2. The License amendments are listed by date of issue and summarized below:

Amendment 193 to DPR-24; Amendment 198 to DPR-27, February 08, 2000: The amendments changed the design and operation of the fuel cycles with upgraded Westinghouse fuel and at higher core power peaking factors than previously permitted by the Technical Specifications.

Amendment 194 to DPR-24; Amendment 199 to DPR-27, March 20, 2000: The amendments removed the methodology of using the reference to infinite multiplication factor (K_∞) when verifying the acceptability of reactor fuel placement and storage in the spent fuel pool. This amendment was based an advisory letter received from the NSSS vendor dated February 26, 1999.

Amendment 195 to DPR-24; Amendment 200 to DPR-27, March 22, 2000: The amendments changed the required frequency of the control rod exercise test from every two (2) weeks to a frequency of once per quarter.

Amendment 196 to DPR-24; Amendment 201 to DPR-27, June 27, 2000: The amendments implemented changes for containment tendon surveillance to better satisfy the intent of Regulatory Guide 1.35, Revision 3.

Amendment 197 to DPR-24; Amendment 202 to DPR-27, August 07, 2000: The amendments transferred operating authority for the Point Beach Nuclear Plant from Wisconsin Electric Power Company to the Nuclear Management Company, LLC.

Amendment 198 to DPR-24; Amendment 203 to DPR-27, August 15, 2000: The amendments eliminate one of the license conditions and associated implementation dates from Appendix C to the licenses. These amendments were in response to an NRC letter dated April 7, 2000.

Amendment 199 to DPR-24; Amendment 204 to DPR-27, November 17, 2000: The amendments more clearly define the requirements for service water system operability. Analyses were performed to support the previous TS that had been approved as Amendments 174 and 178 for Units 1 and 2, respectively, on July 9, 1997. Since that time, a number of inconsistencies between the system model and configuration were identified that questioned whether the previous analyses bounded system operation. These amendments corrected the identified inconsistencies.

IV. 10 CFR 50.59 & 72.48 SAFETY EVALUATIONS

PROCEDURE CHANGES

The following procedure changes requiring safety evaluations (SEs) were implemented in 2000:

AOP-10A: Safe Shutdown in Local Control, Revision 29 (Permanent)
AOP-10B: Safe to Cold Shutdown in Local Control, Revision 2 (Permanent)

Spurious opening of the SI-851A/B valves were not previously addressed in AOP-10A and AOP-10B. The valves could potentially spuriously open if a fire occurred in the cable spreading room or PAB El. 26' central areas. These are alternate shutdown areas. The SI-850A/B valves in the containment sump will not hold back flow (because they are hydraulic valves designed to prevent flow or allow flow from the sump, but not prevent flow to the sump). If the SI-851A/B valves spuriously open, this will open a path to the containment sump from the refueling water storage tank (RWST) since the SI-850A/B valves will not hold flow back and the SI-856A/B valves, the only other valves between the RWST and containment sump, are normally open. The RWST will begin draining to the containment sump. Appendix R calculations show that to stop the drain down and retain sufficient inventory in the RWST to support cool down of the unit, the SI-856A/B valves should be shut within 15 minutes. Previously, in AOP-10A, the SI-856A/B valves were not shut until approximately 30 minutes.

Summary of Safety Evaluation: AOP-10A was revised to minimize the possibility of a spurious opening of the SI-851A/B valves and provide actions, if it does occur, to shut the (in series) SI-856A/B valves in a timely manner to prevent drain down of the RWST. AOP-10B was revised to verify the SI-851A/B valves are shut prior to going on RHR recirculation to prevent a possible loss of RCS inventory to the containment sump.

The procedure changes ensure there are no adverse consequences to the potential spurious operation of the SI-851A/B valves. The changes do not significantly affect the timeline of the procedure. The change does not pose a USQ nor does it require a change to the TS.
(SE 2000-0093)

AOP-10A: Safe Shutdown in Local Control, Revision 28 (Permanent)

AOP-10A, Attachment E was revised to incorporate two changes. The first change involves a revision to the method used to provide emergency power to portable fans used to ventilate the control room, vital switchgear room, cable spreading room, computer room, AFW pump room, and the PAB electrical equipment rooms in the event of a loss of ventilation to these rooms commensurate with an Appendix R Fire. The previous method involved operation of portable gasoline-fueled generators running on the building exterior with extension cords strung to the respective portable fans. The revised method allows the use of in-plant 480 V power receptacles (PR-93 or 2PR-49), one of which will be available for Appendix R fire scenarios as established in the Safe Shutdown Analysis and FPEE 2000-003. Utilization of these receptacles is via the use of a 30 kva portable power transformer and extension cords routed to the portable fans.

Summary of Safety Evaluation: The changes made to AOP-10A were evaluated by vendor Calculation WE-0005-19 and FPEE 2000-003. As a result, a more simple power source for the supplemental ventilation requirements was identified. Implementation of these procedure changes to incorporate this information ensures the operability of process monitoring instrumentation for fires. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0061)

AOP-14A: Main Power Transformer Backfeed, Revision 5 (Permanent)

The procedure was revised to delete the steps pertaining to Unit 2 since procedure 2-SOP-19KV-001, "Main Power Transformer Backfeed" replaced the steps required for Unit 2. The procedure as revised only applies to Unit 1 in the unlikely event backfeed is needed as a result of an abnormal event involving loss of X04 transformer. Deletion of the steps for Unit 2 were prescreened to 2-SOP-10KV-001.

During the development of 2-SOP-19-KV-001 it was discovered that additional jumpers should be installed to defeat the exciter field breaker interlock to allow closure of 1A52-01 and 1A52-17, normal supply to 1A01/2. A corresponding step was added. This step allows closure of the main generator breaker, thus allowing power from the 345 KV grid to go through 1X01 and 1X02 to supply power to 1A01 and 1A02, which is then used to power 1A03, 1A04 and safeguards loads. Prior to closing the generator output breaker, the generator disconnect links are removed. This affords protection of the main generator while on backfeed.

Summary of Safety Evaluation: This procedure is used to supply both the non safeguards and safeguards buses for Unit 1. These changes allow alignment of the electrical systems to aid in accident mitigation and do not increase the probability of occurrence of an accident or event in the CLB nor create the possibility of an accident of a different type.

Main generator protection is provided by the removal of the main generator disconnect links prior to closing the main generator breaker. Shutting the main generator breaker must occur to establish backfeed to the non-safeguards buses. Installation of this jumper and subsequent closure of the main generator breaker will not create the possibility of an equipment malfunction important to safety.

Safety-related loads relied upon to maintain margins of safety continue to be supplied from safety-related buses supplied by the emergency diesel generators (EDGs) until backfeed is established. Once backfeed is established, the EDGs are returned to a standby status improving the reliability of electrical power to safeguards equipment. There is no change in the margin of safety and no increase in the radiological consequences posed by these changes. The unit is shut down and the generator disconnect links removed. Therefore, this change has no impact on the operation of the main generator. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0114)

CSP-Z.1: Response to High Containment Pressure, Unit 1 Revision 15 Unit 2 Revision 15 (Permanent)

The changes revise steps to place one train of containment spray on containment sump recirculation from the RHR train not being used for containment sump recirculation. The previous revision aligned one train of containment spray for sump recirculation assuming both SI trains are on containment sump recirculation. Revisions to EOP-1.3, "Containment Sump Recirculation," place one SI train on sump recirculation while the other train, although aligned for sump recirculation, remains idle in a standby mode. The revision to CSP-Z.1 aligns one train of containment spray for containment sump recirculation assuming one RHR train is idle, which is consistent with procedure EOP-1.3, "Containment Sump Recirculation."

Summary of Safety Evaluation: Steps were revised to place one train of containment spray on containment sump recirculation from the RHR train not being used for core cooling in the sump recirculation mode. The other SI train is also aligned for containment sump recirculation, but is in a standby mode with the RHR and SI pumps not running. The change aligns containment sump recirculation using the RHR train that is in standby. Since an accident is already in progress when this procedure is used, there will be no increase of an accident occurring previously evaluated in the CLB nor will the possibility of an accident of a different type as currently evaluated in the CLB be created.

When in the recirculation alignment, containment spray is not expected to be needed to maintain containment pressure reduction since multiple failures would be required for high containment pressure to occur. However, if a challenge to containment integrity arises while on containment sump recirculation, CSP-Z.1, "Response To High Containment Pressure," will direct the operator to align one train of containment spray for sump recirculation and initiate recirculation spray flow using the discharge flow from the RHR train not being used to supply core cooling. Since this procedure directs the operation of equipment and components as designed, there is no increase in the probability of an equipment malfunction important to safety.

The containment spray train aligned for sump recirculation will be the same train as the RHR train in standby. Given a single failure of a containment spray train not being available, the SI train supplying core cooling can be swapped with the idle SI train, thus allowing the available train of containment spray to be supplied from the same RHR train. There is no reduction of the margin of safety and no increase in the radiological consequences of an accident. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0101)

DCS-3.1.7: Service Water System Operability, Revision (Permanent)

Additional administrative controls were incorporated to ensure that the design of the automatic service water (SW) isolation valves associated with the spent fuel pool (SFP) heat exchangers (HXs) remain functional under credited conditions. The administrative controls were necessary pending TS amendment to ensure simultaneous failures of the redundant valves would not be permitted.

Summary of Safety Evaluation: The additional requirements do not alter the physical plant. The requirements ensure the design isolation occurs if needed. Isolation of cooling water to the SFP HXs is a design feature of the current license configuration. The activity does not degrade nor circumvent this or other design or license requirements.

A change to the TS is required. The activity implements appropriate administrative controls in the interim, pending approval of TSCR 206 and supplements. This is an acceptable compensatory measure in accordance with NRC Administrative Letter 98-10, dated December 29, 1998. The change does not pose an unreviewed safety question USQ nor does it require a change to the TS. (SE 2000-035)

EOP-1.3: Transfer to Containment Sump Recirculation, Revision 26 (Permanent)
EOP-3: Steam Generator Tube Rupture, Revision 29 (Permanent)
ECA-0.0: Loss of All AC Power Unit 1, Revision 28/Unit 2, Revision 29 (Permanent)
ECA-3.2: SGTR With Loss of Reactor Coolant-Saturated Recovery Desired, Revision 24 (Permanent)
CSP-ST.0: Critical Safety Function Status Trees, Revision 1 (Permanent)
CSP-S.1: Response to Nuclear Power Generation/ATWS Unit 1, Revision 21/Unit 2, Revision 22 (Permanent)
CSP-C.1: Response to Inadequate Core Cooling, Revision 21 (Permanent)
CSP-C.2: Response to Degraded Core Cooling, Revision 18 (Permanent)

As part of the safety assessment system (SAS)/plant process computer system (PPCS) upgrade, in MR 98-002*M, four (two per unit) dedicated core exit temperature recorders are installed to provide a new safety-related/seismic display location for the core exit temperature data. This change is being made to address software validation issues for the new computers being installed as part of the SAS/PPCS upgrade. Additionally, installation of the dedicated core exit temperature recorders allows the seismic safety parameter display system (SSPDS) to be removed from service in MR 98-002*C without the need for local monitoring of core exit thermocouple reading as required by AOP-21. The dedicated recorders provide the qualified safety-related display location for the core exit temperature data.

Summary of Safety Evaluation: The procedures were revised to add the two new dedicated core exit temperature recorders to provide a new safety related/seismic display location for the core exit temperature data. These changes meet the requirements of modification MR 98-002*M.and not issued until MR 98-002*M is accepted. Therefore, plant operation was not affected by these procedure revisions. The interim configurations created by MR 98-002*M do not involve a USQ nor require a change to the TS. (SE 2000-0103)

EOP-1.3: Transfer to Containment Sump Recirculation, Units 1 & 2, Revision 26 (Permanent)

The following changes were made to improve guidance for establishing containment sump recirculation: (1) Recirculation alignments were provided for all sizes of loss of coolant accidents (LOCAs) with no assumption of actions being completed in previous procedures. (2) Train B was designated to be the first train for recirculation alignment to ensure multiple flow paths to the core so the final configuration ensures at least two flow paths, regardless of break location. If Train B failure requires Train A alignment, the SI pump discharge cross-connect is opened to provide multiple flow paths to the core through the Train B flow paths. (3) The local action to shut both SI-876A&B SI pump discharge recirculation to SI test line valves was eliminated. (4) Only one RHR pump will be operated on recirculation for a large break LOCA to prevent exceeding the temperature capacity SI suction piping design. (5) Steps to place containment spray on containment sump recirculation were deleted because they were redundant to steps provided in CSP-Z.1, "Response to High Containment Pressure."

Summary of Safety Evaluation: Provisions are made to provide simultaneous vessel and cold leg injection paths by aligning Train B SI injection flow path to meet requirements specified in an NRC safety evaluation report (SER) dated December 24, 1975, to prevent boron precipitation. In the event of a Train B failure, the SI pump discharge cross connect valves are opened to provide the same flow path for Train A SI pump. This ensures two flow paths are provided between the RHR pump and Train B SI flow paths regardless of break location. Minimum core cooling is provided while preventing pump run-out conditions.

Modification of the SI-857A & B RHR heat exchanger outlet to SI pump suction valves provides a relief path through the SI pump suction side disc. Shutting both SI-876A & B SI pump discharge recirculation to SI test line valves is no longer required to prevent leakage from one SI pump discharge from over pressurizing the suction of an idle SI pump.

A single train is placed on containment sump recirculation since both trains of RHR on recirculation could result in exceeding SI pump suction piping design temperatures. Cooling capacity is not exceeded for single train injection. Since core cooling requirements are met with a single train, the second train is aligned and ready for use in the event of a failure occurring of the operating train. The change does not pose a USQ, nor does it require a change to the TS. (SE 2000-0078)

FEP 4.01: PAB West and Central, Revision 6 (Permanent)
FEP 4.03: PAB North, Revision 5 (Permanent)

Source range monitor 2N-32 could potentially not be available for a postulated fire event in the component cooling water (CC) pump room (Fire Zone 142) and the Unit 2 motor control center room (Fire Zone 166) of the primary auxiliary building (PAB) El. 8'. Cable And Raceway Data System (CARDS) shows incorrect fire zone routing for Unit 2 white instrument power cable ZQ2DY03A from 2-83/DY-03 static transfer switch to 2Y-203 distribution panel. 2N-32 is powered from the white channel which is now lost, and 2N-31 and 2N-40 are lost due to instrumentation/signal cable losses in FZ 142 and 166. This affects Unit 2 only.

FEP 4.01 and FEP 4.03 were revised to add guidance to use a special extension cable to restore power to 2N-32. The Fire Protection Evaluation Report (FPER) was revised to account for this potential loss and the necessary manual operator action to be performed.

Summary of Safety Evaluation: Fire detection and suppression in the area mitigates the possibility of an accident or event of a different type and the significance of this condition. It provides a high likelihood that postulated plant fires are prevented or adequately controlled and that safe shutdown equipment remains available. If the postulated loss of source range instrumentation does occur, then the procedure changes provide guidance to restore 2N-32. This ensures that the equipment can be restored in a timely manner and that the probability of an accident event is not increased.

The facility, as described in the FPER, was changed and the implementing procedures were revised to account for this loss. Appropriate manual operator actions were established and procedural guidance provided to respond to the condition. These changes do not increase or introduce an accident or event not previously evaluated in the CLB.

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. Procedural guidance to achieve and maintain safe shutdown in the event of fire and operation actions demonstrates that the plant is capable of achieving and maintaining safe shutdown per the requirements of Appendix R. With compensatory measures in place (revisions to the FPER, FEP 4.1, FEP 4.3, and PC 6 Part 1) and with the special extension cord staged, the 2N-32 source range monitor is fully operable for an Appendix R event. The change does not pose a USQ nor does it require a change to the TS. (SE 2000-046)

OI 62A: Motor-Driven Auxiliary Feedwater System (P-38A & P-38B), Revision 19 (Permanent)

The change provides another method of injecting steam Generator (SG) chemicals using the motor-driven auxiliary feedwater pumps (MDAFP). The new method employs a cart with a tank and an attached pump to inject carbohydrazide, hydrazine, morpholine, and ammonium hydroxide into the nearest drain upstream of the auxiliary feed pump (AFP) suction. The difference between the new method and the old is injection upstream of the MDAFPs as opposed to the previous downstream method. The previous practice added chemicals to tank T-047A or B. These tanks have a capacity of 8 gallons. The tank inlet and outlet valves are opened to provide a parallel path to auxiliary feed flow to the SGs, followed by shutting a motor-operated valve to isolate the normal auxiliary feed flow and force flow through the tanks. After five minutes of flushing the tanks, normal auxiliary feed flow is restored and the tanks are isolated.

Summary of Safety Evaluation: The chemicals have no measurable chemical effect on the MDAFPs. The cart consists of a tank, a pump, hosing to connect all parts, a check valve, a relief valve, and a vent with a filter to control organic vapors. The pump on the cart is rated for a deadhead pressure less than the AF system piping, so a piping rupture cannot occur. The connecting hose is rated for the same pressure as AF system piping. An adapter with a check valve is added to prevent the condensate tanks from draining into the AFP room in the event of a hose rupture. The FSAR was revised to note this new method is available and preferred.

Since the new addition method is upstream of the AFPs, upon a trip of the opposite unit, there is a possibility the chemicals will enter the steam generator of the tripped unit. If chemicals were injected into a steam generator on a tripped unit, the high pH and high cation conductivity alarms might activate, but the conditions in the steam generator will not cause tube degradation. There is no action level associated with high pH for hot standby. When cation conductivity is corrected for organic content, there will be no action level entered. If an action level was entered, the corrective actions are listed in plant administrative procedures. The chemicals will have no detectable effect upon the materials of the AFPs. The change in injection point has no other effect on the plant. At least one Level 3 dedicated operator will attend the cart at all times during injection from the cart. No accidents are initiated by the change in injection method, and no radiological consequences will occur due to the change. The amount of weight added to the AF system piping is less than 20 lbm, which is the minimum needed for a seismic evaluation. If a chemical spill occurred, the same effects and response as a spill of a 5-gallon bottle of hydrazine that is currently used will be implemented.

This change increases operator safety and more efficiently provides for addition of chemicals to the steam generators. There is a small possibility of adding chemicals to an unintended steam generator, but the effect will be minimal, and will not damage the affected steam generator. There is no increase in probability of an accident. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0112)

OI 155, Chemical Treatment of Service Water System for Zebra Mussels Revision 6 (Permanent)

The procedure provides for the periodic treatment of the SW system with a chemical called EVAC to kill off adult zebra mussels in the system. Periodic performance of OI-155 is scheduled throughout the year to kill zebra mussels in the SW system before they are expected to grow to a size at which they could block off the smallest diameter tubes in safety-related SW heat exchangers (e.g., G01 and G02 EDG HXs). Previous zebra mussel kills were performed once a year and resulted in HX tube plugging. Entry of zebra mussels of appreciable size is considered insignificant since the main SW header Zurn strainers with 1/8" maximum openings are normally on-line.

Summary of Safety Evaluation: The planned chemical treatment consists of injecting approximately 2-4 ppm of active amines into approximately 15,000 gpm of service water for approximately 8-12 hours. The chemical injection lines were connected to the discharge pressure gauge tubing lines of two SW pumps. A secondary chemical containment is in place to prevent potential spills from interacting with equipment or draining to Lake Michigan. The EVAC chemical and injection equipment has been reviewed and zero negative impact was identified on the SW system components from chemical attack, pressure integrity concerns, seismic integrity concerns or other concerns. Equipment parameters that may indicate blockage of components by zebra mussel shells were monitored and contingency plans for blockage were developed. Monitoring of operating parameters during and after the treatment and inspections performed after the treatment ensures continued operability of SW equipment.

This treatment is not specifically addressed in the CLB. It aids in meeting regulatory commitments to ensure the operability of safety-related SW system equipment as required by GL 89-13. The treatment cannot initiate an accident or event, whether new or previously evaluated. In the unlikely event that flow restriction in a SW component important to safety occurs, precautions are in place to diagnose and correct the condition prior to an effect occurring upon the operation or performance of either unit. The addition of EVAC presented no radiological issues. A sufficient margin of safety was maintained for service water flows by controlling the equipment that is in service to meet the SW flow analysis. The change does not pose a USQ nor does it require a change to the TS. (SE 2000-0054)

OI 156A, Rydlyme™ Treatment of 1HX-15A Containment Fan Cooler Coils, Revision 0 (Permanent)
OI 156B, Rydlyme™ Treatment of 1HX-15B Containment Fan Cooler Coils, Revision 0 (Permanent)
OI 156C, Rydlyme™ Treatment of 1HX-15C Containment Fan Cooler Coils, Revision 0 (Permanent)
OI 156D, Rydlyme™ Treatment of 1HX-15D Containment Fan Cooler Coils, Revision 0 (Permanent)
OI 157A, Rydlyme™ Treatment of 2HX-15A Containment Fan Cooler Coils, Revision 0 (Permanent)
OI 157B, Rydlyme™ Treatment of 2HX-15B Containment Fan Cooler Coils, Revision 0 (Permanent)
OI 157C, Rydlyme™ Treatment of 2HX-15C Containment Fan Cooler Coils, Revision 0 (Permanent)
OI 157D, Rydlyme™ Treatment of 2HX-15D Containment Fan Cooler Coils, Revision 0 (Permanent)

The SW system is treated to remove zebra mussels that have accumulated in system components exposed to raw water from Lake Michigan. Since there is concern that zebra mussel shells may cause one or more containment fan coolers (CFCs) to become inoperable because of tube blockage, new procedures were developed to dissolve and remove the blockage by recirculating an industrial descaler (Rydlyme™) solution through the CFCs.

Since fouling of the CFC has an adverse affect on the operability of the CFC, the new procedure can also be used to treat the CFC tubes to reduce the fouling factor determined during periodic testing. If one or two CFCs are blocked because of shells, the affected unit enters a 72-hour LCO as required by TS 15.3.3.B.2.a. If three or four CFCs are inoperable on a single unit, the affected unit is shut down per TS 15.3.0. The procedures allow removal of the blockage or reduction in the fouling factor in the affected CFCs. The affected CFCs are isolated. A tank and circulation pump are used to circulate the chemical to dissolve and clear the blockage. The pump draws suction from the tank and injects the Rydlyme™ into the local drain valve in the SW supply. The treatment occurs at ambient temperature with the injection pumps injecting chemicals at a pressure that is less than the SW design pressure of the CFCs. The drain in the return line discharges into the tank to allow the liquid to evolve carbon dioxide and be strained for solids prior to recirculation. The waste chemical effluent is disposed of to the retention pond. This disposal method has been approved by the Wisconsin Department of Natural Resources.

Summary of Safety Evaluation: Since the shells are largely composed of calcium carbonate, an industrial descaler (Rydlyme™) is used to dissolve the shells. Rydlyme™ is added to the existing SW in the CFCs to make an approximate 50% solution to clean the CFC. The resulting solution contains approximately 5% hydrochloric acid, which as shown to cause minor corrosion concerns for an evaluated 100-hour exposure time. Following the treatment with Rydlyme™, the CFC is flushed until the concentration of Rydlyme™ is reduced to a level comparable to sample results of the SW system. This stops the increased corrosion rates prior to returning the CFC to operation. Rydlyme™ is biodegradable and has no toxic fumes. Carbon dioxide gas is evolved when Rydlyme™ reacts with scale. Since carbon dioxide is known to be non-toxic and non-flammable, the only concern is to ensure sufficient ventilation to dissipate the expected amount of evolved gas. Since only a small amount of shells are expected, the amount of gas evolved is minimal and within the capacity of local ventilation. The CFCs are isolated before Rydlyme™ is applied. Therefore, 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. The manipulated valves are shown on FSAR Figures 5.2-35-1 and 5.2-35-2 as containment boundary valves. The 8" manual containment isolation valves for the SW side of the CFCs located outside containment in the PAB are shut. The 1" inlet and outlet drain valves for the SW side of the CFCs, also located outside containment in the PAB, are open and controlled by a Level 2 dedicated operator in accordance with TS 15.3.6.A.1.b and procedurally controlled. 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. Following the treatment and flushing, operability testing is performed to confirm design flow rates are established. The change does not pose a USQ nor does it require a change to the TS.
(SE 2000-032)

<u>OP 2A</u>: Normal Power Operation, Revision 35 (<u>Permanent</u>)

OP 2A was revised based on an evaluation performed by System Protection regarding 345 KV system/grid stability as related to power generation and grid configuration.

Load reductions will be performed at the direction of System Control. It is the responsibility of System Control to monitor system/grid configuration and load and proactively direct power generation to maintain a stable grid.

The title of Section 10.1 was changed from "Operation With Only Two 345 KV Incoming Lines Available," to "Operation With Less Than Four 345 KV Incoming Lines Available." System Control should be consulted for all line/grid configuration change impacts on power generation. The 50% power generation restrictions when the number of lines to the switchyard is reduced to two were based on a 1992 analysis. Subsequent analysis in 1995 clarified guidance for the various line element configurations. Steps directed power reduction to 50% whenever the number of lines to the switchyard was reduced to two. This was not necessarily a conservative measure as a generation reduction may have an adverse impact on system integrity and stability.

<u>Summary of Safety Evaluation</u>: The event of concern is a loss of AC power. These procedure changes enhance grid integrity in that the procedure formerly required power generation reductions for most two-line configurations to the switchyard. There are two line configurations for which a power generation reduction is not warranted from a system stability perspective. A power generation reduction might have an adverse impact on offsite power supply to PBNP, depending on system/grid power supply and configuration. The decision to reduce generation needs to be based on an evaluation of system/grid power supply and configuration at the time of occurrence. Therefore, these procedure changes do not increase the probability of occurrence of an accident or event previously evaluated in the CLB.

Equipment malfunctions at the 345 KV level are not specifically addressed in the CLB. There is a TS regarding the need to reduce generation in the event that the number of lines to the switchyard is reduced to one. Power generation reduction decisions need to be made at the time of system/grid configuration change by System Control at the time of configuration change. These procedure changes enhance the integrity of the offsite power supply to PBNP. Therefore, procedure changes do not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB.

The basis for power generation reductions to address electrical system/grid stability concerns is not discussed in the TS Bases. These procedure changes do not reduce the margin of safety defined in the basis for any TS. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0096)

OP 4D Part 1: Draining the Reactor Coolant System, Revision 51. (Temporary)
OP 4D Part 1: Draining the Reactor Coolant System, Revision 52. (Permanent)

The procedure revision adds new instructions for demonstrating single train (redundant) decay heat removal (DHR) capability prior to draining to three-quarter pipe.

Summary of Safety Evaluation: The new section provides the necessary instructions for demonstrating the capability for refueling shutdown (less than or equal to 140°F at 1000 gpm) and for cold shutdown (less than or equal to 160°F at 1200 gpm; consistent with calculations P-90-023 and N-89-010). The new section further demonstrates single train decay heat removal (DHR) capability for a specified "desired" RCS temperature at 1000 gpm, to ensure adequate DHR capability at the final planned reactor coolant system (RCS) temperature. The method for performing the demonstrations is the same as previously contained in the procedure, except that the flow rates are changed. Residual heat removal (RHR) is placed in single train operation, valves are aligned to establish required flow rates, adjustments are made to ensure component cooling and service water flows are correctly balanced. Adequate DHR capability is then verified by checking that the required temperature is maintained. The new section provides for restoration and system realignments required to maintain the final target RCS temperature.

OP 4D Part 1 had required RHR flow to be reduced to 1000 gpm when draining the RCS to 22-25% level. The change adds steps to verify DHR capability at this reduced flow rate. The RHR system is operated via the normal control system. The changes do not impact the ability to operate or control the RHR system. The change does not pose a USQ nor does it require a change to the TS. (SE 2000-027)

PBTP 103: Unit 2 Main Generator Voltage Regulator Checkout and Testing Revision 0 (Permanent)

PBTP 103 provides instructions for the checkout and testing of the Unit 2 main generator voltage regulator following voltage regulator replacement per MR 96-046*B. This testing consists of checks for proper connection and setup of the voltage regulator to the exciter and 19 KV system; establishing proper dampening for stable operation of the closed loop voltage regulator; collection of step response data for system stability studies; and operational testing of limited functions on the new voltage regulator.

Summary of Safety Evaluation: The function of the Unit 2 main generator is described in FSAR 8.3 as the normal source of power for the main (X01) and auxiliary (X02) transformers. It is classified as non-safety related, has no safety functions and is not described in the CLB.

This testing is a one time evolution performed to ensure proper operation of the equipment. The testing of the Unit 2 main generator voltage regulator could result in a turbine-generator trip. However, to minimize this risk, prior to placing the voltage regulator in service, testing was performed as part of the modification installation to ensure proper connections and operation. The on-line testing was conducted to minimize the possibility of a generator trip. Improper functioning of the voltage regulator under the test conditions is no different than the voltage regulator failing to function properly during normal on-line operation. Therefore, the probability of the occurrence of an accident or event previously evaluated is not increased.

Bus voltages were maintained within the allowed band of $\pm10\%$ from nameplate to ensure the availability of equipment to perform its intended normal or emergency function. The operating of electrical motors (pumps, fans, compressors) and other equipment at higher or lower than normal system voltage for the brief times in this test will not have any long-term effect on the reliability of the equipment. The emergency power sources were unaffected by this testing and remained capable of supplying safety-related loads if required. Therefore, the probability of occurrence of a malfunction of equipment important to safety as previously evaluated is not increased.

The testing of the main generator voltage regulator did not directly challenge any of the fission product barriers (clad, RCS boundary, containment boundary). Since failures associated with this testing are bound by a loss of the main generator and the main generator is not credited to mitigate any of the accidents described in the FSAR, this testing did not affect the availability of any structures, systems or components necessary to limit radiological consequences of analyzed accidents and events. Therefore, the radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated are not increased.

The testing of the main generator voltage regulator cannot create the possibility of an accident or event of a different type than any previously evaluated in the CLB.

No new failure modes have been created by the testing of the main generator voltage regulator. Generator protective relaying will trip the turbine-generator off line to protect the generator and transformers from exceeding their thermal limits. This is the expected and designed function of this equipment. Therefore, the testing of the main generator voltage regulator cannot create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated.

TS do not identify a margin of safety associated with the turbine-generator. Therefore, the margin of safety defined in Technical Specifications have not been reduced. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0115/SE 2000-0115-01)

PTBCH-SG-001: Westinghouse Procedure for Steam Generator Mechanical Tube Plugging and Stabilizer Installation, Revision 1.

Steam generator mechanical tube plugging is a repair process that removes a tube from service when degradation is discovered during eddy current examinations. Mechanical plugging involves inserting a plug in each end of the tube and then rolling the plug into the tube. The installed plug effectively removes the tube from service and the plug becomes the new RCS pressure boundary. The installation of a mechanical plug may not include the installation of a stabilizer. A stabilizer would normally only be installed to preclude interaction with adjacent active tubes if the potential exists for the plugged tube to separate at the location of degradation.

Summary of Safety Evaluation: The steam generator tube plug is designed to the same design criteria as outlined in FSAR Chapter 4, "Reactor Coolant System." Qualification test results on the plug design have demonstrated that the plug will not leak during normal operating and accident conditions and will not dislodge and cause damage to steam generator tubing or the RCS. Plug material is compatible with primary and secondary water chemistry. The performance of the installed mechanical plugs is not affected by the installation of flexible stabilizers. The installed configuration of flexible stabilizers resting atop mechanical plugs has been shown to be acceptable with regard to fretting wear of the plug. Should a plugged tube separate at a tube location above the plug, the geometric interference provided by the diameter of the stabilizer in the installed configuration will preclude interaction with adjacent active tubes. The installation of mechanical plugs and stabilizers does not diminish the integrity of the RCS in any way. The change does not pose a USQ nor does it require a change to the TS.
(SE 2000-0111)

RESP 2.3: Defective Removable Top Nozzle Replacement, Revision 3 (Permanent)
WEST MRS-GEN-1013-RRTN-FP: Simplified Replacement Tooling Generic Field Procedure
WEST STD-FP-1998-8251: Fuel Assembly Top Nozzle Clamp Inspection Procedure, Revision 4
WEST STD-OP-1992-6095: Single Locktube Tool Operating Procedure, Revision 4.

RESP 2.3 was revised to attach a bracket on the SFP bridge to hold the spent fuel handling tool during top nozzle replacement. New procedures and a simplified top nozzle removal tool were developed by Westinghouse to improve the replacement process. WEST STD-OP-1992-6095 allows use of an air-operated tool. The replacement top nozzles use a spring screw that is made from peened (bead blasted) Alloy 718 to preclude primary water stress corrosion cracking. RESP 2.3 was also revised to allow qualified Westinghouse technicians to move fuel assemblies in accordance with the plant's fuel handling procedures.

Summary of Safety Evaluation: Fuel assembly top nozzles were replaced to address the potential for broken holddown spring screws, a component for the top nozzle. Broken spring screws have been found in Westinghouse fuel including PBNP Unit 1. The replacement top nozzles use a spring screw that is made from peened (bead blasted) Alloy 718 to preclude primary water stress corrosion cracking. The FSAR describes the spring screw material as Alloy 600.

The peened Alloy 718 holddown spring screw is less susceptible to stress corrosion cracking and has higher tensile strength. The corrosion resistance and mechanical characteristics of design for the spring screw are improved, therefore, the new design does not compromise the performance of any system nor does it affect an analysis.

The new design does not increase the probability of occurrence, or the consequences of an accident evaluated in FSAR. Evaluation of the use of peened Alloy 718 fuel assembly top nozzle spring screws concludes that it will not result in a USQ nor does it require a change to the TS. (SE 2000-0102)

RMP 9010: Freeze Seal to Support 1SI-00853C Disassembly, Revision 0 (Permanent)

A procedural temporary modification is used to install and remove a freeze plug to support disassembly of low head safety injection (SI) core deluge check valve 1SI-00853C when Unit 1 is in cold shutdown and the reactor is fueled. The freeze plug is installed upstream of low head SI core deluge isolation valve 1SI-00852A. Valve 1SI-00852A is a double disc gate valve, that is maintained shut while on RHR. The plug is required to prevent RHR flow diversion through the RHR side disc of 1SI-00852A and through the bonnet relief path to SI-00853C. The freeze area piping is constructed of 6" seamless A 312 Type 304 SS Schedule 40S.

Summary of Safety Evaluation: The freeze plug is controlled by RMP-9010, "Freeze Seal Installation and Removal," and Freeze Technology International, Inc. Procedure FTII PRO Revision G., NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," refers to Battelle Columbus Laboratories November 1982 report, "Development of Guidelines for the Use of Ice Plugs and Hydrostatic Testing." This report provides guidelines for using ice plugs and supporting documentation for the procedural temporary modification. The applicable potential hazards listed in the Battelle report associated with ice plug use are addressed by this safety evaluation. Based on these discussions, the potential for a failure of the plug or the affected piping is minimal.

During the freeze plug evolution, both loops of RHR remain fully operable as required by TS. Following the freeze plug evolution, full flow testing verifies the plug has completely dissipated and that ECCS is capable of performing its design function. The change does not pose a USQ nor does it require a change to the TS. (SE 2000-025)

0-TS-RE-001: Power Level Determination Revision 4 (Permanent)

0-TS-RE-001 specifies methods to determine reactor thermal output, and the surveillance requirement of TS Table 15.4.1-1, Item 1, Line 2 (the daily power range nuclear instrumentation [NI] heat balance calibration). CR 00-1089 identified that only a secondary calorimetric calculation should be used for performance of this surveillance, however, the procedure did not make this distinction clear. The change to 0-TS-RE-001 clarifies the methods to be used for performing reactor thermal output. The method of normal power level monitoring is unchanged by the revision.

Summary of Safety Evaluation: There are no accidents initiated by the power range NIS. As long as the NIS is properly calibrated, it performs its functions as analyzed. The procedure change ensures the calibrations meet the intent of the CLB. The ability of the NIS to mitigate accidents is preserved by ensuring they are properly calibrated. There are no new accidents or malfunctions introduced by the changes. The margin of safety is unchanged by the procedure revision. The change does not pose a USQ nor does it require a change to the TS.
(SE 2000-051)

1 & 2-SOP-19KV-001: 1/2X-01 Main Feed Power Transformer Backfeed Operation, Revision 1 (Permanent)

These procedures permit a backfeed to initiate a power supply to a shutdown unit's A01 and A02 buses from X02 via X01. This allows greater flexibility in powering a shutdown unit's 480 V and 4 KV buses during X04, A03 and/or A04 maintenance outages or equipment failure.

Summary of Safety Evaluation: Backfeed is initiated only when the associated unit is shut down. Loading of the X02 transformer is procedurally limited to well within the capability of the transformer. Procedural instructions prohibit the loading of a dry fuel cask when the backfeed is in service. This eliminates the possibility of a fuel handling event. No additional electrical distribution equipment is required. The addition of ground sensing relay protection on the 19 KV system is a duplication of the protection disconnected with the main generator. The temporary relaying provides alarm protection to allow operators to correct a ground on the 19 KV system.

Since various switching configurations are possible on the 4.16 KV system, A03 or A04 out of service for example, loss of backfeed to the A01 and A02 buses would be the most restrictive situation. The unit to be back fed is required to shut down all safety related functions associated with the A01 and A02 buses. Under-voltage, under-frequency, feedwater isolation, protection, are not required to be in service since the affected equipment is not in service. This relaying does remain functional for the backfeed. Existing protective relaying provides protection for plant equipment. Decay heat removal continues to be supplied from safety-related buses supplied by the existing offsite power 4.16 KV distribution system. This distribution may be subject to existing LCOs based on equipment alignment. Quality of power from X02 is equivalent to the quality of power from X-04 for non-vital bus loads.

There is a remote possibility of a problem with sympathetic inrush when energizing the X01 transformers from the 345 KV system. This condition could result in transformer damage and/or the trip of an operating unit. Precautions were included to minimize the likelihood of such an occurrence. This change does not pose a USQ nor does it require a change to the TS.
(SE 2000-0071)

MODIFICATIONS

The following modifications were implemented in 2000:

MR 94-082*A: Meteorological Tower Recorders Replacement

MR 94-082*A replaced and relocated the meteorological (MET) chart recorders and surge chamber (SC)/forebay level recorder with one single electronic recorder capable of displaying multiple parameters at the same time. The circulating water (CW) inlet temperature indicator was also relocated and replaced with two indicators.

Summary of Safety Evaluation: The two indicators concurrently display temperatures for Unit 1 and Unit 2. Relocating the MET recorders and CW temperature indicator from panel 1C103 to panels 2C103 and C01, respectively makes room for an installation of the new radiation monitor system control terminal.

No new failure modes are being introduced that could cause a malfunction of equipment important to safety. The new instrumentation is for indication and provides no control functions to plant equipment, nor will it increase the probability of a malfunction of any other equipment associated with the MET or CW systems. The MET recorders, CW temperature indicators and forebay level recorder do not affect the radiological consequences of an accident or event previously evaluated in the CLB.

Removal of the MET recorders results in a decrease in electrical loading at the 1C103 panel. The recorders at the 2C103 panel replaced with a single recorder decreases electrical loading at this panel. This change does not pose a USQ nor does it require a change to TS.
(SE 1999-0126)

MR 96-037: Relocation of Inverter DY-0A

The purpose of MR 96-037 is to resolve NRC Generic Letter 87-02 seismic interaction issue concerning inverter DY-0A and the neighboring engineered safeguards relay rack 1C-167. The Seismic Qualification Utility Group (SQUG) inspection criteria requires that essential relay cabinets are to be free from impact. This modification resolves this issue by relocating Inverter DY-0A away from 1C-167 such that they will not impact during a design basis seismic event. The existing Inverter DY-0A is QA and safety related. The inverter base is supported by structural channels which are seismically mounted via welds to a baseplate bolted into the concrete slab.

Summary of Safety Evaluation: MR 96-037 was installed under an Installation Work Plan. Inverter DY-0A was taken out of service and isolated. Red channel 120 V AC was supplied by 1DY-01 and 2DY-01 during this modification. The function of the inverter remained the same as before the modification. The inverter was thoroughly tested upon reconnection of the external wiring to ensure reliability.

The red instrument busses were supplied in the normal configuration during the relocation of the Inverter DY-0A. Inverters 1DY-01 and 2DY-01 were operating in the normal configuration. There was no change to the function, operation or components form the previous condition. Compensatory measures, such as a fire watch, were taken to ensure that no conditions were created that take the plant out of the CLB. In addition, the welding machine was located outside the cable spreading room to mitigate any potential EMF interference. Therefore, this modification did not increase the probability of occurrence of an accident, event or malfunction of equipment important to safety previously evaluated in the CLB.

This modification did not increase the radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB since inverter DY-0A was returned to its original function and configuration. The possibility of an accident or event of a different type than any previously evaluated in the CLB has not increased. No possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the CLB was created. There are no Technical Specifications that require changing due to this modification. The effect of the seismic upgrade was to increase overall plant safety by ensuring that DY-0A and 1C-167 do not impact during a design basis seismic event. This change does not pose a USQ nor does it require a change to the current TS. (SE 99-023)

<u>MR 96-037*A</u>: Relocation of Inverter DY-0B

The purpose of MR 96-037*A is to resolve NRC Generic Letter 87-02 seismic interaction issue concerning existing inverter DY-0B and the neighboring engineered safeguards relay rack 2C-156. The Seismic Qualification Utility Group (SQUG) inspection criteria requires that essential relay cabinets are to be free from impact. This modification resolves this issue by relocating Inverter DY-0B away from 2C-156 such that they will not impact during a design basis seismic event. The existing Inverter DY-0B manufactured by Solidstate Controls, Inc. (SCI) is QA and safety related. The inverter base is supported by structural channels which are seismically mounted via welds to a baseplate bolted into the concrete slab.

<u>Summary of Safety Evaluation</u>: MR 96-037*A was installed under an Installation Work Plan. Inverter DY-0B was taken out of service and isolated. Blue channel 120 V AC was supplied by 1DY-02 and 2DY-02 during this modification. Incidental impact between DY-0B and 2C-156 during DY-0B movement was prevented by precautionary notes and bracing of DY-0B at midsection as directed in IWP 96-037*A. The function of the inverter remained the same as before the modification. The inverter was thoroughly tested upon reconnection of the external wiring to ensure reliability.

The Blue instrument busses were supplied in the normal configuration during the relocation of the inverter DY-0B. Inverters 1DY-02 and 2DY-02 were operating in the normal configuration. There was no change to the function, operation, or components from the previous condition. Compensatory measures, such as fire watch, were taken to ensure that no conditions were created that take the plant out of the CLB. In addition, the welding machine was located outside the cable spreading room to mitigate any potential EMF interference. Therefore, this modification did not increase the probability of occurrence of an accident, event or malfunction of equipment important to safety previously evaluated in the CLB.

This modification did not increase the radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB since Inverter DY-0B was returned to its original function and configuration. The possibility of an accident or event of a different type than any previously evaluated in the CLB has not increased. No possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the CLB was created. There are no Technical Specifications that require changing due to this modification. The effect of the seismic upgrade was to increase overall plant safety by ensuring that DY-0B and 2C-156 do not impact during a design basis seismic event. This change does not pose a USQ nor does it require a change to the current TS. (SE 99-057)

<u>MR 96-067*A</u>: Capping of the Boric Acid Gas Stripper Vent Lines

Condition Report 99-1478 addresses an unplanned discharge of two gas decay tanks through the boric acid gas strippers. On June 2, 1999, the gas analyzer was tagged out for maintenance isolating instrument air to the boric acid gas strippers. Isolating air to the gas strippers caused 1BS-CV-004A and 2BS-CV-004B (boric acid gas strippers to the vent header isolation valves) to fail open. This resulted in the vent header feeding back into the boron recycle process gas stripper towers and releasing potentially radioactive gas to the PAB. The boric acid gas strippers are not used and have not been maintained.

<u>Summary of Safety Evaluation</u>: Modification 96-067, addresses abandoning the boric acid gas strippers. The boric acid gas stripper systems are not used except for several inlet valves (BS-CS-009A/B, BS-CS-010A/B and BS-CS-011A/B) that are used to redistribute/concentrate holdup tank effluent. Effluent does not enter the preheaters or the gas stripper towers (HX-6A/B). The 96-067*A package caps the ¾" stainless steel piping near 1BS-CV-004A and 2BS-CV-004B and 3/8" stainless steel tubing near 1BS-CV-005A and 2BS-CV-005B. Both ends of each line were capped. The 1BS-CV-004A and 2BS-CV-004B valves have fail open operators and although the 1BS-CV-005A and 2BS-CV-005B valves have fail close operators, these lines were also capped. The system drawings were reviewed and capping these lines prevents a release from the gas waste gas system through the boric acid gas strippers. During installation of the caps, a portion of the waste gas header was isolated, including the Unit 1 and Unit 2 VCT discharges to the waste gas system. The VCTs are normally isolated by the CV-258-S valves from the waste gas system. The CV-258-S valves are opened to vent hydrogen gas from the VCTs during tank level changes.

Capping the unused lines from the boric acid gas strippers to the waste gas header prevents the release of radioactive gas from the system in the event that Instrument Air fails or if the BS-0CV-004/5A/B valves are mis-positioned. The boric acid strippers are not used, and therefore the abandonment (capping) of the gas stripper vent lines does not increase the probability of an accidental release for the systems or an event different than any previously evaluated. The caps, installation and post-maintenance testing met the appropriate Code requirements and design guidelines. The lines were cut and capped near supports so additional supports are not needed and the seismic classification of the lines are not affected. Capping the lines are more reliable than fail open valves, and therefore the probability of a malfunction of equipment important to safety is not increased.

With the lines capped, the activity does not increase the radiological consequences of a malfunction of equipment. During installation, the portion of the waste gas system was isolated. The waste gas system contains hydrogen. Both a pre and post maintenance nitrogen purge was used to ensure that a flammable gas mixture is not created. Mechanical connections were also used to ensure that there is no possibility of fire/explosion. The post-maintenance purge ensured that no significant oxygen level is introduced in the waste gas system. These installation precautions ensure the risk of fire is minimal. This change does not pose a USQ nor does it require a change to the TS. (SE 99-112)

MR 97-014*E: Transfer DC supplies for Inverters 1DY-01 and 2DY-02, install Panel D-26 and transfer loads to D-26. The modification transferred the DC power supply for inverter 1DY-01 from D-12 to D-11; transferred the DC power supply for inverter 2DY-02 from D-14 to D-13; installed new 125 V DC distribution panel D-26 in the cable spreading room near existing panel D-12; and supplied power to panel D-26 from existing spare switch (D72-01-03) in main 125 V DC distribution panel D-01.

The following loads were transferred during the modification: common Train “A” loads to D-26, including DC power to EDG G-01; inverter DY-0A, 1C-20 and 2C-20 annunciators; RMWT common Train “A” loads to D-26, including DC power supplies.

EDG G-01 was out of service. This does not require an LCO entry, provided EDG G-02 is operable. During the transfer of DC supplies to loads in 1C-20 and 2C-20, the DC supplies were temporarily paralleled to maintain power to all affected loads during the transfer process. During the transfer of the DC to emergency lighting panel EB, the AC supply to the panel remained energized.

Summary of Safety Evaluation During installation of this modification, several loads supplied from the 125 V DC system were de-energized. None of these loads were associated with the initiators of accidents described in the CLB. Installation controls in place during work in energized panels ensured that in-service loads were not affected. Following installation of this modification all affected loads were restored to full operability. No safety-related loads are being added, removed, or modified within the scope of this change. Only the DC power supplies for the affected loads are being reconfigured. New DC panel D-26 was manufactured for use in QA, safety-related, Seismic Class 1 applications. The new panel contains fused switches instead of molded case circuit breakers. The fused switches improve DC system selective coordination and increase fault clearing capability. Mounting of panel D-26, including conduit rework, is seismically qualified.

Rerouting of cables did not violate train separation criteria including Appendix R requirements. During installation of this modification the equipment supplied from the 125 V DC system which is needed to mitigate the consequences of an accident remained operable as required by TS.

Following installation of this change, Appendix R timelines for achieving safe shutdown has been improved due to improved selective coordination. For these reasons, this activity does not increase the probability of occurrence or radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB. This modification does not introduce any 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 have been previously evaluated. Therefore, the activity does not create the possibility of an accident, event, or malfunction of equipment important to safety of a different type than any previously evaluated in the CLB. Operability of all affected systems and equipment was maintained in accordance with TS requirements. This modification does to affect the degree of independence of safety-related DC system trains. This modification does not change any control functions or setpoints. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0063)

MR 97-014*F: Replace Panels D-12, D-13, D-14, D-18, D-19, D-22 and Transfer Loads

MR 97-014*F replaces 125 V DC panel D-14 and transfers power supplies for common Train "B" loads; D-19, resupplies panel D-19 from panel D-14 and transfers common "B" train loads; D-13 and D-18 and transfer Unit 2 Train "B" loads; D-12 and transfers Unit 2 Train "A" loads; D-22, resupplies panel D-22 from panel D-12 and transfers Unit 2 Train "A" loads to D-22. This change also redesignated normal and alternate switchgear DC control power supplies for 4.16 KV buses 2A-01, 2A-02, 2A-03, 2A-04 and 480 V buses 2B-01 and 2B-02.

Summary of Safety Evaluation: The old DC panels were replaced with new panels containing fused switches. These switches improve DC system selective coordination and increase fault clearing capabilities. The new panels were installed in the same location as the old ones. All buswork, switches and fuses on the new panels are rated to supply the maximum expected loading during normal and accident conditions. This change does not affect heat load or ventilation requirements for the areas in which the panels were replaced. During panelboard replacement and transfer of loads, loads which are not required to be operable were de-energized. LCOs were entered for SW, AF and emergency power in accordance with TS.

Rerouting of cables does not violate train separation criteria, including Appendix R requirements. Appendix R timelines for achieving safe shutdown have been improved as a result of selective coordination and removal of non-safety related loads from safety-related DC panels. This activity does not increase the probability of occurrence of radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB. This change does not create the possibility of an accident, event or malfunction of equipment important to safety of a different type than previously evaluated in the CLB. This change does not change control functions or setpoints, nor does it affect the degree of independence of safety-related DC system trains. This change does not pose a USQ, nor does it require a change to TS. (SE 2000-0056)

MR 98-024*D: Unit 1 Auxiliary Feedwater Pump Area Service Water Return

MR 98-024*D adds additional isolation valves, an ultrasonic flow meter, and flush connections to the auxiliary feed pump (AFP), control room (CR), cable spreading room (CSR) HVAC and turbine hall SW headers. The additional isolation valves are: (1) Between the Unit 1 and Unit 2 SW return headers on the 3" AFP/(CR) and (CSR) HVAC SW return header between each auxiliary feedwater pump cooling return line; (2) Between the 4" Unit 1 turbine hall SW return and the 10" Unit 1 SW return header; and (3) On a new flow path installed between the SW return from CR and CSR HVAC condenser (immediately upstream of SW-119) and the 10" Unit 1 return. A section of 4" stainless steel (SS) piping and removable insulation are also installed in the Unit 1 turbine hall return piping to allow use of an ultrasonic flow meter to measure flow rates. Flushing improvements are made by additional flush connections hard piped between the P-38A and 1P-29 SW supply lines and the SW return header.

Summary of Evaluation: Because of necessary locations of the new components, portions of MR 98-024*D were installed and tested under a 7-day Unit 2 P-38A LCO per TS 15.3.4.C.2. Line stops were installed between P-38A and P-38B and immediately south of the 4" Unit 1 turbine hall equipment return in the SW return header to establish the isolation boundaries required to perform work.

All accidents, events or malfunctions of equipment important to safety which could credibly result from the installation of MR 98-024*D were previously evaluated by the CLB or supporting procedures. The new configuration does not interface with or include equipment unique to the original configuration or high energy sources capable of initiating a new accident. No fire penetrations (permanent or temporary) were degraded as a result of the modification. Flooding or loss of SW pressure boundary is not increased above the CLB limits. This change does not pose a USQ nor does it require a change to TS. (SE 99-082)

MR 98-024*H: Water Treatment Area Redundant Isolation Valve Motor Operator

The modification converts existing gate valve SW-527, on the service water supply to the water treatment area, into a motor-operated valve. (Note: "Water Treatment Area" describes all loads supplied by the header containing this isolation valve, including the water treatment area cooler, the Unit 2 electrical equipment room cooler (which includes the Unit 2 control rod drive coolers), dechlorination and water pretreatment equipment, and the deaerator vacuum pumps). The addition of a Limitorque SMB-00 motor operator and associated hardware will model the configuration of existing isolation valve SW-2817, which is powered from Train "B" MCC 2B42 compartment. 10F. The modified valve, to be relabeled SW-4478, will be powered from Train A MCC 2B32 compartment 12D to ensure adequate train separation for these redundant valves. A local FVR starter with 480-120 V control transformer and nonfusible disconnect switch will be installed near the valve. The local control panel consists of a three-position spring-return-to-center control switch for local operation, and valve position indication lights. Additional components (terminal blocks and control cables) will be installed to facilitate installation of future modifications.

Summary of Safety Evaluation: MR 98-024*H converted service water manual valve SW-527 to a motor-operated valve (MOV) and relabeled SW-4478. The MOV is powered from MCC 2B32 (Unit 2 Train A) and is redundant to MOV SW-2817 (Unit 2 Train "B"). A local control panel was also installed containing a control switch and position indication. The design change also facilitates implementation of future modifications to provide control room valve control and position indication and automatic valve actuation signals. This evaluation also includes descriptions of the effects of this modification on the FSAR and FPER. These effects include changing figures which show SW-527 as manual valve.

Systems affected by this modification (service water and the site electrical distributions system) are not identified as initiators of accidents or events evaluated in the Current Licensing Basis. Thus, the probability of such events is not increased. The mechanical effects of the new equipment have been evaluated and will not result in increased probability of equipment malfunction since the new actuator, motor and hardware meet the same quality standards as existing equipment and piping analysis has demonstrated that design limits are not exceeded. Radiological consequences of events and malfunctions are not increased since the ability of the affected systems to perform their functions was not decreased either during or after installation. The creation of a new accident, event or malfunction of equipment important to safety is not possible because any possible failures of affected components are bounded by system design features or higher level failures previously considered. No TS or their bases are affected by this change since branch line isolation valve SW-2817 (required by 15.3.3.D) will remain in service to isolate as designed, and none of the other equipment is required to meet TS requirements.

The FSAR and FPER changes were necessary to reflect the new equipment design, and do not raise any issues not addressed during design preparation. Electrical train separation is maintained in accordance with guidance in existing 10 CFR 50 Appendix R exemptions for the affected fire zones and approved station procedures. This change does not pose a USQ nor does it require a change to the TS. (SE 99-065)

MR 98-0024*I: PAB Cooler Redundant Isolation Motor Operator for SW-502 (SW-4479)

The modification converts SW-502 (6" manual gate valve which isolates the service water supply to the PAB HVAC coolers) to a motor-operated valve by adding a Limitorque SMB-00 motor operator. The body of valve SW-502 (relabeled SW-4479) is cut out and rotated away from the wall to provide clearance between the new operator and the 6" service water riser. The body of SW-2816, in-line motor-operated valve, is also cut out and rotated in the same direction in order to facilitate a future modification to replace its Limitorque SMB-00 operator with a larger SMB-00.

MR 98-024*I also installs a dedicated cable for MOV SW-2927A (inlet to SFP HX-13A) from MCC 2B-42 to C-01; a dedicated cable for MOV SW-2927B (inlet to SFP HX-13B) from MCC 2B-32 to C-01; a dedicated cable for MOV SW-4478 (inlet to water treatment area coolers) from MCC 2B-32 to C-01; remote control and indication for SW-4478 on C-01; a Unit 1 Train "A" automatic SW isolation SI signal to the control circuit of SW-4478. A contact was wired out to a terminal strip in 1C-157 from service water isolation relay 1-SW-AX and tested by MR 98-024*U.

Summary of Safety Evaluation: The SW supply line to the PAB coolers, water treatment area and spent fuel pool cooling system, and the valves in these lines, are not the initiators of accidents analyzed in the CLB. These lines serve non-essential loads that are isolated in the event of an accident. The 480 V electrical distribution system serves loads that are required for accident mitigation, but it does not have the potential to cause an analyzed accident. Therefore, design changes to these systems cannot influence the probability of occurrence of analyzed accidents. The modification to the piping system and supports including the effects of additional weight and rotating the two valves involved has been seismically qualified via S&L calculation WE-300027. Materials added to the piping system are designed to the same quality standards and codes. Operation of original MOV SW-2816, which has the safety function of isolating nonessential PAB cooler loads during an accident, is not changed. New MOV SW-4479 is operated from the control room and is interlocked with the Unit 1 SI initiation logic. Operation of this valve (if manual action is needed) is no more difficult and takes no more time than before adding the motor operator. The new operator is equipped with a hand-wheel for use in the event of motor failure or loss of AC power. Control logic changes for the other MOVs affected by the modification do not adversely impact the operation of the valves or applicable power supplies and control circuits. The fire hazard for the affected zone does not change significantly. The work does not invalidate the existing Appendix R exemptions nor does it violate the conditions imposed for granting the exemption requests. Testing consists of stroke testing SW-2816, setting up, stroke timing and testing the automatic function of SW-4479, stroke testing SW-2927A and SW-2927B, and stroke timing and testing the automatic function of SW-447. Testing is performed by train and is performed such that no LCOs are entered (e.g., SW-2816 is verified operable prior to removing the line stop machine.) Changes to the FSAR and FPER are required to reflect this design change but do not affect the probability of occurrence of an accident or event. The design change enhances our ability to respond to an accident. The modification does not pose a USQ nor does it require a change in the TS. (SE 2000-002)

MR 98-024*K: Install Ultrasonic Flow Meter Transducers on CFC Motor Cooler Piping in Unit 2 CFC and Accident Fan Motor Cooler (AFMC) Replacement.

This is the electrical portion of MR 98-024*K which installed permanent ultrasonic flow transducers on the 2" SW supply piping to each of the four CFC motor coolers (2HX-15A, B, C, D).

Summary of Safety Evaluation: The ultrasonic transducers were mounted on new stainless steel pipe sections installed as part of MR 98-024*K. The old differential pressure flow indicators were completely removed. Root isolation valves on the 2" SW return piping and one on the supply piping remain to provide a convenient draining point for the motor coolers. The installation took place during U2R24 with Unit 2 in cold shutdown. The transducers and couplant are compatible with SS piping and will not degrade the piping and associated containment penetration boundary. The ultrasonic flow instrumentation is accurate enough to verify operability of the CFC motor coolers. Radio frequency interference generated by the ultrasonic flow meter will not affect plant control systems. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0092)

MR 98-024*K replaced the Unit 2 A and B CFC cooling coils (2HX-15A1-8 and 2HX-15B1-8) and the A, B, C, and D AFMC cooling coils (2HX-15A, 2HX-15B, 2HX-15C and 2HX-15D) with coils that are fabricated of material better suited for the water conditions of the SW system.

Summary of Safety Evaluation: MR 98-024*K improved the material condition, reliability and maintainability of the Unit 2 CFCs and AFMCs. This modification replaced the A and B CFC cooling coils; A, B, C, and D AFMC cooling coils; the SW system supply/return piping and isolation valves to the A and B CFC cooling coils; the AFMC SW flow instrumentation and upgrading the platforms that support the A, B and C CFC plenums. This was installed during the U2R24, while the unit was in cold shutdown. This condition did not require CFC operation for accident mitigation.

The new CFC and AFMC coils have been designed to satisfy the heat removal requirements under the design limiting conditions for heat exchanger fouling, tube plugging and SW flow. The new design will not result in SW boiling at the outlet of the CFCs. Post-modification testing verified that the SW flow to the new coils and the air flow in the containment air recirculation cooling system satisfies the requirements for design basis heat removal. Pressure boundary components of the new coils and SW piping have been analyzed and designed for the applicable hydraulic transient and seismic loading conditions. The support structure modifications restored the required design margins for the applicable loading conditions. Since the new components installed are passive, no new failure modes in the containment air recirculation cooling or SW systems have been created. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0099)

MR 98-024*M: G01/G02 Air Compressor Room Service Water Return Header Isolation Valves.

This modification installs three stainless steel 10" manually-operated isolation valves (SW-730/731/732) and a pressure gauge with dual sensing lines in the 10" SW return header; a 3" alternate return header and isolation valves (SW-770/771/772/773/774) for the four instrument air (IA) and service air (SA) compressors; and to hard-pipe the four air compressor SW supply line flush connections. SW-730 is south of the emergency diesel generator (EDG) G01 return line, SW-731 is between the return lines for the G01 and G02 EDGs, and SW-732 is north of the air compressor return lines. The two taps for the pressure gauge tie into the 10" SW return header on both sides of SW-731. The 3" alternate return header ties into the 10" SW return header, north of SW-732.

Summary of Safety Evaluation: Systems affected by this modification are not identified as initiators of accidents evaluated in the CLB, thus the probability of such accidents is not increased. Installation required that EDGs G01 and G02 (one at a time) and the SA/IA compressors (one set at a time) be taken out of service and temporary equipment installed for pressure boundary isolation. The mechanical effects of this temporary equipment were fully evaluated and determined not to impact operability of the associated systems. The reduced redundancy of this equipment is mitigated by adherence to TS 15.3.7.B.1 for the EDGs and be the design features of the SA/IA systems, including backup nitrogen bottles and air accumulators.

Radiological consequences of accidents and events considered in the CLB are not increased, since the affected systems are not sources of radiological contamination and failure of this equipment is bounded by the most limiting single failure, e.g., loss of an entire electrical train. Because such failures are so bounded, there is no potential of creating an accident or event not previously evaluated. nor does the potential for a malfunction of equipment not previously considered. No TS margin of safety is affected since the SW system remains fully operable during and after installation, and the EDGs were only taken out of service one at a time, while following the limitations of TS 15.3.7.B.1. This change does not pose a USQ nor does it require a change to TS. (SE 99-027-01)

MR 98-024*R: Unit 2 Auxiliary Feed Pump (AFP) Service Water (SW) Flush and Return Isolation

MR 98-024*R installed a manual isolation valve on the 3" SW return header between the P-38B and 2P-29 AFP cooling return lines. The change allows isolation of either unit's return headers without requiring a dual unit outage or substantial freeze seal. Flushing connections were hard piped between the AFP supply lines and SW return line to improve line flushing capabilities.

Summary of Safety Evaluation: Portions of MR 98-024*R were installed and tested under a 7-day Unit 1/P-38B LCO. A line stop was installed in the 6" SW return line north of the AFP returns to establish the isolation boundaries required to perform work.

Accidents, events and malfunctions of equipment important to safety that could credibly result from the installation were previously evaluated by the CLB or supporting procedures. Engineering calculations demonstrated SW flows to and from critical components (including AFP bearing cooling and net positive suction head [NPSH] available) were not adversely affected by the interim or final modification configurations. Because the SW and AF systems are not sources of radioactive materials and are not initiators of accidents and operation of equipment important to safety remained as described in the CLB, analyzed accidents or events with radiological consequences as described in the CLB remained unchanged. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0038)

MR 98-024*V: Modify Safety Injection (SI) Logic for Non-Essential Service Water (SW) Load Isolation Valves.

MR 98-024*V completed revision of the Unit 2 A and B train SI SW isolation logic started by MR 98-024*W (SE 2000-0033) described below. The Unit 1 A and B train SI signal connection was performed by the previously described MR 98-024*I (SE 2000-0002).

Summary of Safety Evaluation: MR 98-024*V completely and electronically isolates the 4/6 matrix circuitry and removes the 30-second time delay, completing the final portion of the SW upgrade project associated with revising isolation logic and installing redundant automatic motor-operated isolation valves to non-essential SW loads.

This change affects SW-2816 (inlet to PAB/SSB coolers), SW-2817 (inlet to WT area coolers) SW-2930A (outlet from SFP HX-13A), SW-2930B (outlet from SFP HX-13B), SW-LW61 and SW-LW62 (inlet to and outlet from the radwaste coolers), SW-2927A (inlet to SFP HX-13A), SW-2927B (inlet to SFP HX-13B), SW-4478 (inlet to WT area coolers) and SW-4479 (inlet to PAB/SSB coolers). Adding SI logic to new valves SW-2927A/B, SW-4478, and SW-4479 increases SW margin to essential loads (.e.g., containment fan coolers and EDG coolers) during design basis accidents because these valves are fully redundant to SW-2930A/B, SW-2817 and SW-2816, respectively, in power supply and Unit 1 and Unit 2 SI signals. This redundancy also simplifies the SW flow model since non-essential SW loads will be assumed to be isolated under any design basis condition.

The installation and testing took place with only one train of safeguards and valves de-energized at a time. There were no new changes made that would cause failure of the SW and SI systems and their ability to perform their design functions. The SW system nor any of its components is not identified as an initiator of an accidents or events as described in the CLB. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0052)

<u>MR 98-024*W</u>: Disable Unit 2 Four-out-of-Six Service Water Isolation Logic, IWP 98-024*W

MR 98-024*W revises the Unit 2 Train "A" and "B" SI SW isolation logic such that the valves shut on any Unit 2 SI signal within its associated train. This consists of abandoning in place the four-out-of-six (4/6) SW logic that isolates the SW from non-essential loads if less than four SW pumps start following a SI signal. The change in logic affects the following Train "A" valves: SW-2816 (inlet to PAB coolers) and SW-LW62 (outlet from radwaste coolers). The isolation signal to 2SW-2880 is removed by the modification.

<u>Summary of Safety Evaluation</u>: One of the goals of the SW upgrade is to improve the margin to essential loads, (e.g., the CFCs during design basis accident conditions.) The most limiting design basis accident is a five operable SW pump scenario. Under this scenario, SW to non-essential loads is not isolated. Removing the four-out-of-six SW isolation logic and installing redundant motor operated valves (through other SW modifications) allows SW to be positively isolated to these loads on SI signal and improves margin in the system. SW isolation capability is removed from 2SW-2880 to allow the turbine to be shutdown and secured in a safe manner (e.g. cooling remains to the turbine lube oil coolers).

The margin gained by revising the isolation logic and installing MOVs is much greater than the margin lost by allowing 2SW-2880 to remain open. Testing involves taking voltage readings at strategic points to ensure the control circuit is correctly revised. No valves are stroked during testing to avoid unnecessary plant transients. Applicable procedures are followed to ensure proper control is maintained throughout installation and testing.

The modification does not cause a failure of the SW and SI systems nor does it affect their ability to perform their design functions. The SW system is assumed to be available during accidents or events described in the CLB. In addition, neither the SW system nor its components are identified as an initiator of accidents or events as described in the CLB. During the installation process, procedures are utilized to ensure proper control is maintained throughout the installation. The procedures ensure that the final installation is in accordance with the design. In addition, testing is performed in accordance with approved procedures and with Operations input. Equipment associated with the modification important to safety is QA for safety-related applications. No new equipment is installed. The control circuits are simplified requiring less equipment to function to achieve the desired results. Therefore, the probability of a malfunction of equipment is not increased. This modification does not pose a USQ nor does it require a change to the TS. (SE 2000-0033)

MR 98-032: P-35B-E Diesel Fire Pump HX Water Supply Line Strainer

MR 98-032 upgrades the cooling water piping to P-35B-E Diesel Fire Pump Engine.

Summary of Safety Evaluation: The new configuration fully conforms to NFPA 20-1999 requirements and consists of the addition of two Y-strainers, a pressure regulating valves, and manual valves to provide an identical main and bypass cooling water path to the diesel fire pump engine, and the P-35B bearing oil cooler. P-35B was be taken out of service to perform this modification, and P-35A (electric fire pump) was tested daily to verify operability.

P-35B is credited in the Appendix R scenario to provide fire suppression water to any safe shutdown equipment areas, and also to provide backup cooling to the turbine driven auxiliary feedwater pumps in the SBO event. Failure of P-35B is not an initiator to any design or licensing basis accident or event. MR 98-032 does not increase the probability of occurrence of an accident or event previously evaluated in the CLB, or of a different type than any previously evaluated. This change did not result in a reduction in cooling ability and increases reliability by preventing debris that might be present in the lake water from clogging the pressure regulating valves. All components were selected with pressure ratings that meet or exceed the design ratings of the fire protection system. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0068)

MR 98-050*F: Access Control System Replacement.

MR 98-050*F installs the field-mounted access control equipment, file server consoles and equipment, the main security consoles, and connects the doors, intrusion detection equipment and cameras onto the new access control system. The installation includes power to the console equipment and connection of the communications cables to the field multiplexer chassis and router/repeater.

Summary of Safety Evaluation: MR 98-050*F routed conduits between new and existing equipment. During installation security was notified so t compensatory measures could be taken in accordance with the Security Plan and the Transition Plan. New circuit breakers were added in the security multiplexers or local circuit breaker boxes to provide independent protection for each power circuit. Each door installation, camera and intrusion detection zone was transferred onto the new security computer system and tested.

Work on the security access control system was performed in accordance with 10 CFR 50.54(p), governing changes to the Security Plan that do not reduce security effectiveness. A letter was submitted to the Commission that described the interim configuration of the security system and the compensatory measures that were implemented to ensure the security of the facility was not degraded during the renovation. This change does not pose a USQ nor does it require a change to TS. (SE 99-099)

MR 98-116: 4160 V AC Air Magnetic Breaker Replacement

MR 98-116 replaces Westinghouse 4160 V AC 50DH350 air magnetic breakers with Westinghouse 4160 V AC 50DH-VR 350 vacuum breakers.

Summary of Safety Evaluation: MR 98-116 replaced the air magnetic breakers in buses 2A01, 2A02, 2A03, 2A04, and 2A05 which were not previously replaced by SPEED 97-019. Non-safety related vacuum breakers were installed in 2A01, 2A02, 2A03, and 2A04 and qualified vacuum breakers were installed in bus 2A05. The breaker replacements were controlled by IWPs. Installation of the vacuum breakers required some re-wiring to remove the x and y relays which were used by the air magnetic breakers. These coils are not required for the proper operation of the vacuum breakers since the vacuum breakers have x and y coils installed in the breaker truck.

The new vacuum breakers are an improvement in the design and technology over the air magnetic breakers. The new breakers are less subject to wear and are more easily maintained than the old breakers. The failure rate is expected to be as low or lower than that of the old breakers. Installation of the vacuum breakers does not introduce any new accidents nor do they increase the consequences of an accident. The vacuum breakers are manufactured to specifications in IEEE C37.59-1991 and the breakers are seismically qualified by testing and analysis in accordance with IEEE 344-1987. The new breakers are as reliable as the original breakers in performing their design function under normal and accident conditions evaluated in the CLB. This change does not pose a USQ nor does it require a change to TS. (SE 98-111-01)

MR 99-005: Instrument Air containment Isolation Valve Trim Pack Replacement

MR 99-005 replaces the valve trim in the instrument air (IA) containment isolation valves and slows the opening time of the valves. The valves had quick opening trim. The modification installs equal percentage valve trim and slows the opening time of the valves. The changes decrease initial demand on the instrument air system header allowing for a smooth draw from the IA system. The opening logic is changed via addition of an auxiliary relay to allow the valve to continue to come open after the selector switch is returned to auto.

Summary of Safety Evaluation: Installation of the valve trim was done during a refueling outage in accordance with a work order work plan. Other work may be done at full power or during a refueling outage. Work done at power requires one of the redundant valves to be shut. A failure of the remaining open IA containment isolation causes the valve to fail safe (shut). If both IA isolation valves are shut, the unit needs to be shut down. Therefore a plan is in place to provide a dedicated operator to obtain control of one IA isolation valve. The dedicated operator opens IA containment isolation valve by gagging the valve open locally. A dedicated operator remains at the valve until it is operable or shut. The dedicated operator maintains radio contact with the control room. The dedicated operator shuts the IA containment isolation valve within one minute of a CI signal. This modification does not pose a USQ nor does it require a change to the TS. (SE 99-075-02)

MR 99-029*B: Auxiliary Feedwater Pump (AFP) P-38A and P-38B Minimum Flow Recirculation Line Flow Orifice Replacement.

MR 99-029*B replaced orifices RO-04008 and RO-04015 in the auxiliary feedwater (AF) system with improved design orifices and installed oversized socket welds in the associated piping. This should minimize piping line noise and vibration and preclude socket weld failure when operating these lines.

Summary of Evaluation: During the installation of MR 99-029*B one motor-driven AFP was taken out of service at a time where we entered a 7-day LCO according to TS 15.3.4.C. The orifices RO-04008 and RO-04015 are not discussed in the CLB as a contributor or initiator to an accident or event scenarios already evaluated in the CLB. The replacement orifices meet design, material, construction and testing standards of the previous installation and do not degrade the overall performance, operation or function of the AFPs. The AF system does not participate in radiological release mechanisms, and no new radiological release mechanisms or paths are created. Therefore, the installation does not increase the probability of occurrence of an accident or the radiological consequences of an accident or malfunction as previously evaluated in the CLB.

The new ROs provide improved flow characteristics and prevent cavitation damage, thus minimizing pipe vibration under liquid application and associated socket weld failure. The recirculation line flow path is not required to support the AF system in its response to the design basis accidents since the AFP discharge valves automatically open fully in response to the accident and provide a flow path for the pump. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0055)

MR 99-043: Modify Unit 2 Pressurizer Power-Operated Relief Valve (PORV) Solenoid Cables to Satisfy Appendix R Requirements.

MR 99-043 installed new cables for each Unit 2 PORV (2RC-430 and 2RC-431C) in dedicated conduits from the containment penetration to the control room, bringing these valves into compliance with the requirements of 10 CFR 50 Appendix R.

Summary of Safety Evaluation: Previous routing of 2RC-430 and 2RC-431C cables, coupled with the occurrence of an Appendix R fire event on PAB El. 26' could have caused these valves to spuriously open as a result of a hot smart short from another source. This would prevent the pressurizer pressure relief line from being isolated, which could cause a loss of RCS integrity and the pressurizer relief tank (PRT) rupture disc to fail before the fire could be extinguished and PORVs fail shut by removing their instrument air supply. Installing a new cable for each PORV in a dedicated conduit ensures that a hot smart short from a postulated fire in the PAB does not spuriously open these valves and allow them to be returned to a fully operable status.

Final wiring connections and post-maintenance testing was performed with Unit 2 in refueling shutdown with the reactor head detensioned and all bolts removed. Under this condition, the PORVs are not required for low temperature overpressure protection and did not require an LCO to be entered. There are no new changes being made that would cause a failure of the PORVs or the RCS and their ability to perform their design functions. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0082-01, which supersedes SE 2000-0082 in its entirety)

MR 99-045: Re-route Unit 2 Steam Generator (SG) Pressure Transmitter 2PT-483 Cable.

MR 99-045 re-routes cable ZP2I483F for red channel "B" SG pressure transmitter 2PT-483 so it does not enter on PAB El. 26'. The cable was re-routed off PAB El. 26' by entering PAB El. 8' from the Unit 2 turbine hall instead of entering at PAB El. 26'. This prevents redundant SG pressure transmitters for Unit 2 (2PT-469 and 2PT-483) from becoming unavailable in the event of an Appendix R fire where both cables previously traveled together through the PAB.

Summary of Safety Evaluation: The new route travels through the water treatment area and the Unit 2 charging area instead of through the north wing and middle section of PAB El. 26'. Since the cable previously passed through PAB El. 8' to get to the 2C-205 panel in the AFP room, the new route eliminates PAB El. 26' from the route and introduces additional fire areas on PAB El. 8'. The additional fire areas were selected as acceptable for compliance with Appendix R. This brings the circuit for 2PT-483 into full compliance with Appendix R. The cable re-route was performed while Unit 2 was in cold shutdown to avoid a potential impact on operation of Unit 2. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0067)

MR 99-060*B: Unit 1 Control Circuit Modification of CV-112C Valve to Preclude Spurious Operation.

MR 99-061*B: Unit 2 Control Circuit Modification of CV-112C Valve to Preclude Spurious Operation.

CR 99-2341 identified a condition where MOV CV-112C, normal supply to the charging pumps, could spuriously shut due to a postulated Appendix R fire. To correct the condition, the control circuit conductors that can cause the spurious operation were rerouted to prevent the possibility of a hot short.

Summary of Safety Evaluation: The function of the circuit was not changed. The change eliminated a possible malfunction method for the valves resulting in an improved circuit design that supports the 10 CFR 50 Appendix R requirements. Appropriate precautions were placed in the implementation procedure to ensure that operating equipment cannot be inadvertently operated while the work was in progress. Operations was alerted to the need to manually position the valves should it be required while work was in progress. Design and installation controls, along with testing following the installation to ensure valve operability prior to return to service. The changes do not increase the probability of an existing accident or create new accidents or events or malfunctions of equipment important to safety.. This change does not pose a USQ nor does it require a change to TS. (SE 1999-0122)

MR 99-062: Removal of Contaminated Clothes Washer/Dryer Units

MR 99-062 Removes the contaminated clothes washer/dryer units (Z-037A1, Z-037A2, Z-037A3)/(Z-037B1, Z-037B2, Z-037B3).

Summary of Safety Evaluation: Prior to removal, the electrical supply was de-energized from the power panel and the deionized water (DI) water supply was isolated upstream of the washers and capped. The DI water isolation only affected the downstream laundry sink and its operation with the respirator cleaning station, requiring minor coordination since the non-outage frequency of operation is monthly. Since Z-037B2 and Z-037B3 are auxiliary steam heating dryers, the auxiliary heating and steam supply and return lines were capped. The washer trench and dryer lint traps were decontaminated before work since they pose a slightly higher worker contamination risk. Conformance to the vendor's radioactive materials transportation license requirements was ensured.

The laundry units are not part of TS and do not have a reasonable ability to affect an existing TS. The units are not safety-related or involved with safety related equipment. The disconnecting and capping of the systems (HVAC, DI water, electrical, and auxiliary heating steam) do not negatively affect the systems, and lighten their overall load. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0011)

MR 99-069*A: Reactor Fuel Upgrade to 422V+ Fuel

Starting with Unit 1 Cycle 27 and Unit 2 Cycle 25, Westinghouse 14x14 0.422" diameter VANTAGE+ fuel with PERFORMANCE+ features (referred to as 422V + fuel) will be loaded in the reactor cores as feed assemblies. The improved features will permit higher burnups, future operation at an uprated power level of 1650 MWt, a reduction in the number of feed assembly components, and will support extended 18-month fuel cycles. Modification MR 99-069*A (Unit 2) document this upgrade to 422V+ fuel.

Summary of Safety Evaluation: MR 99-069*A (Unit 2) will increase in the low-low steam generator trip analytical setpoint in accident analysis; increase RCS operating pressure to a nominal 2235 psig; revise procedures that operate the RCS at 2235 psig, test the RCS at a nominal 2335 psig, require a 161-hour delay for refueling, address the weight of the 422V+ fuel, and place the PORVs in manual mode during RCS testing; and add each unit's steam generator design differential pressure data to the FSAR.

These changes will not adversely affect any equipment important to safety, nor will it potentially initiate or create the possibility of an accident or event or increase the radiological consequence of an accident or event, nor will it constitute a reduction in a margin of safety for the following reasons:

- All components exposed to RCS pressure have retained the required design rating of 2485 psig.
- A review of all components exposed to RCS pressure has concluded that they will function at the higher operating pressure, with the exception of the CV-1298 valves, that are replaced by MR 99-069*B and MR 99-070*B.
- Any slight increase in RCS leakage or increased maintenance on the charging pumps is acceptable and is expected to be minimal.
- The change in the steam generator low-low trip analytical setpoint does not affect any plant systems or equipment, and does not constitute a reduction in margin of safety.
- The required procedure revisions will not affect equipment important to safety nor can they cause any accident or event.
- The additional steam generator design differential pressure data does not effect any plant systems or components.

This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0070-01, which supersedes SE 2000-0070 in its entirety.)

MR 00-011: Replace Speed Controller on P-226 (dry fuel storage system)

Speed controller SC-09725 for P-226 cask dewatering moisture separator condensate pump has failed because of the speed controller being left energized when not in use. CR 98-3894 documents this problem. The manufacturer of the current speed controller admitted to a high level of failures in recent history, but has been unable to formulate a permanent design fix. MR 00-011 replaces the speed controller on P-226. This provides more reliable control of P-226, which is used during dry cask loading activities. MR 00-011 removes the electrically operated pump speed controller and its associated equipment. A mechanically operated controller (a manual valve) is installed in its place. The globe valve is installed on a 1" carbon steel service air line located in the truck access area on PAB El. 26'. Approximately 4-6' of service air piping is also replaced.

Summary of Safety Evaluation: Replacement of the current pump speed controller with a manual globe valve on the service air line does not initiate evaluated accidents. Compared to the current speed controller, the globe valve is equally capable of throttling service air to control P-226; however, the possibility of leaving the controller plugged in and causing a failure in this manner is eliminated. The piping is designed and installed to meet or exceed the original piping code. The valve installed meets design class requirements. During installation, the portion of the system modified is completely insoluble. Appropriate installation precautions are taken and post-modification testing is performed to ensure that the new valve functions as intended. The modification does not increase the amount of contaminated fluid in the cask de-watering system nor does it increase the activity level of the service air line. The new valve and piping is not called upon to mitigate accidents, events or equipment malfunctions important to safety in the ISFSI licensing basis. No new failure modes are added as a result of the modification. Removing the current speed controller actually eliminates the documented failure mode associated with it. Therefore, the modification does not increase the probability of occurrence, possibility, or radiological consequences of an accident, off-normal event, or malfunction of equipment important to safety previously evaluated in the ISFSI licensing basis or otherwise.

No USQ, significant increase in occupational radiation exposure, significant unreviewed environmental impact, or conflict with license conditions as contained in the ISFSI Certificate of Compliance is involved by the modification. (SE 2000-0049)

MR 00-020: Unit 2 Condenser Steam Dump Valve Positioner Control Replacement

MR 00-020 replaced the condenser steam dump valve positioner with an improved higher capacity, double-acting positioner.

Summary of Safety Evaluation: The valve positioner replacement allowed removal of the volume booster and the current to pneumatic (I/P) converter for each valve. Improved feedback linkage was also installed and the restrictive 3-way solenoid was replaced with a 2-way solenoid. The solenoid valve added to the closing air line provides for normal blow open operation. A supply air pressure regulator and new air filter were added to ensure clean supply air within the manufacturers recommendations for the equipment in the pneumatic control scheme and valve actuator.

These upgrade do not impact or decrease the ability of the steam dump valves to operate, nor are they credited in any accident or radiological scenario per the FSAR or TS. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0069-02 replaces SE 2000-0069-01 and SE 2000-0069 in their entirety)

TEMPORARY MODIFICATIONS

The following temporary modifications (TMs) were implemented in 2000:

TM 99-060: Furmanite of 1SI-0853C.

The TM applies on-line leak sealant to the body-to-bonnet joint of 1SI-00853C, low head SI core deluge check valve

Summary of Safety Evaluation: This SE revision changes the procedure number. Per Furmanite procedure N-99290 and Furmanite re-injection procedure N-2000004, the final unvented Furmanite leak sealant compound injection pressure cannot exceed 3193 psig. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-133-02)

Summary of Safety Evaluation: The SE revision addresses another injection of 1SI-00853C. The leak sealant process initially injects sealant at 4393 psig (Furmanite procedures N-99290 Revision 0 and N-99290 Revision 1) and 5000 psig (Furmanite procedure N-2000016) with the clamp vented. Since the clamp is vented, no significant forces are transmitted to the body-to-bonnet studs. The final injection of the leak sealant is performed with the clamp unvented. Per Furmanite injection procedures N-99290, N-2000004, and N-2000016, the final unvented Furmanite leak sealant compound injection pressure cannot exceed 3193 psig. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-133-03)

TM 00-012: 480 V AC Power Supply for B-07, B-08/B-09 During H-01 Outage

TM 00-12 provides 480 V AC to load centers B-07 (north service building) and B-08/B-09 (alternate shutdown supply). The TM supports the scheduled H-01 bus outage. A temporary diesel generator provides 480 V AC. Temporary cables are routed through the Unit 2 turbine building and terminated at the respective load centers. Additionally, a 480 V AC supply is provided to the electronic document management system (EDMS) drives in fifth floor computer room. This supply is routed from 2B-11, power receptacle 2PR-31. The TM is in place for the duration of the H-01 bus outage.

Summary of Safety Evaluation: The TM to power B-07, B-08/B-09 loads does not increase the probability of occurrence of an accident or event evaluated in the CLB. The B-08/B-09 load center does not normally supply safety-related equipment and is not relied upon for the mitigation of a design basis accident or event. Alternate shutdown loads (B-08/B-09) and north service building loads (B-07) are shed in the event of a loss of offsite power. Load center B-07, Appendix R, alternate 480 V AC supplies to safe shutdown components are not available for the duration of this TM. Components identified on FPER Table 6.7-1, 4 KV vital switchgear alternate shutdown components for hot shutdown, which does not have its alternate 480 V AC supplies are: 1P2A, (Unit 1 charging pump), 2P 2A, (Unit 2 charging pump), P-32C & E, P-32 B & F (SW pumps), and D-109, (station swing battery charger). Compensatory measures are put in place in accordance with OM 3.27. The TM does not pose a USQ nor does it require a change in the TS. (SE 2000-0039)

TM 00-027: Temporary Modification for Containment Penetration 56 Unit 2

TM 00-027 provides service air to the Unit 2 containment building (temporarily) during the CFC replacement project, through Penetration 56 utilizing an external air compressor.

Summary of Safety Evaluation: A temporary ball valve assembly was fabricated and substituted for containment penetration 56 outside flange. This assembly functions as a containment closure device. Except for the Unit 2 containment, TM 00-027 implementation is independent of and has no interface with plant systems, structures or components. Procedural controls were in place since there was fuel in the reactor vessel to ensure that trained personnel are available to perform containment closure if required. Replacement gaskets and existing flange bolting were used to restore the existing flanges following TM 00-027 removal.

No permanent effects of containment penetration 56 were experienced. At the conclusion of the CFC replacement project and prior to exiting cold shutdown, the penetration was restored and verified by leakage testing to be fully capable of performing its safety related function as a fission product barrier. Since TM 00-027 does not adversely affect any equipment failure mechanisms, the probability of a malfunction of equipment important to safety previously evaluated in the CLB is not increased.

All affected equipment was capable of performing its safety function as described in the Basis for the TS for the conditions specified. Following restoration of the penetration and prior to Unit 2 exiting cold shutdown, TS-required leak rate testing was successfully performed in accordance with ORT 58 to verify the as-left penetration is capable of performing its safety-related containment integrity function. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0100)

MISCELLANEOUS EVALUATIONS

The following evaluations were implemented in 2000:

CR 97-0968: Valve Lineup Changes for Instrument Air and Service Air to Fuel Transfer System

The change evaluates leaving the service air supply valves to the Unit 1 and Unit 2 fuel transfer carts (SA-226 and SA-227) and shutting the instrument air supply valves (IA-250 and IA-251) during normal operation.

Summary of Safety Evaluation: The supply of air to the fuel transfer system is used to drive an air motor that moves the fuel transfer cart into and out of containment during fuel handling. Lining up the valves so the fuel transfer system is supplied by service air reduces the probability that instrument air is valved in when the fuel transfer cart is moved. Instrument air is normally valved in, but the lineup is switched over to service air prior to moving the transfer cart. No equipment is installed via this change. The only change made is a change in the normal valve lineup for air to the fuel transfer system.

The fuel transfer system was initially constructed to operate using the instrument air system. In 1974 the system was modified per modification requests M-67 and M-68 to use service air instead of instrument air. The modification was to reduce the potential for a unit trip that might have resulted from a drop in instrument air system pressure when instrument air was used to drive the fuel transfer cart motor. After the modification, instrument air was lined up to the fuel transfer cart drive system during normal operations and procedural controls were implemented to shift the supply over to service air prior to moving the transfer cart. However, this scheme resulted in at least one instance of both air systems being lined up simultaneously to the Unit 2 air drive motor.

Since the change in the valve lineup reduces the potential for instrument air to incorrectly be left lined up and used to move the transfer cart, there is a reduction in the probability for causing a reduction in instrument air header pressure and a plant trip on loss of air pressure to the main steam isolation valves. There are no changes to the design, operation, or maintenance of the fuel transfer system. Service air is used to move the transfer cart as was the case before this change. Therefore, there is no increase in the probability or consequences of malfunctions or events. Plant safety is enhanced by reducing the probability of a plant trip. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0045)

CR 00-1201: WVSC-24-09 Dry Cask Storage Lid Separation Issue.

This evaluation is to determine if accepting the "as found" configuration of the structural and shield lids associated with WVSC-24-09 entails a USQ. The "as-found" configuration of the lids differs from the "as-described" configuration in the ISFSI licensing basis in that the structural and shield lids are assumed to be slightly separated from each other, and the licensing basis states that they are in contact.

Summary of Safety Evaluation: The separation led to a concern with respect to one of the closure welds placed during loading operations-the ¼" seal weld between the structural lid and the shield lid, which was postulated to become loaded (due to the lid separation) in a vertical drop scenario. Analysis has shown that the VSC-24 design and licensing basis features for WVSC-24-09, have been maintained even though a literal non-conformance with a statement in the FSAR was discovered.

As a result of this examination, it was determined that the as-found (and as-left) configuration of WVSC-24-09 does not involve a 10 CFR 72.48 USQ, significantly increase occupational radiation exposure, create a significant unreviewed environmental impact or require a change to TS. (SE 2000-0060)

CR 00-1644: PAB Electrical Equipment Room Loss of Ventilation Test

The test measures the change in ambient air temperature in electrical equipment rooms A and B after a complete loss of forced ventilation (Fans W-85 and W-86). The test determines operability of equipment in the rooms in light of Appendix R concerns raised in CR 00-1644, where a loss of ventilation to the rooms questioned whether the equipment located within the rooms would remain operable.

Summary of Safety Evaluation: The test maintained temperatures within the operable capabilities of the equipment in the room at all times. The W-85 and W-86 fans were available for restart at any time should plant conditions warrant it. The test equipment did not have an effect upon the equipment in the rooms and was properly secured to prevent movement while placed in the room. Portable ventilation was also staged and ready for use in the unlikely event that neither the W-85 nor W-86 fan could be restarted as a result of a station blackout (SBO) or Appendix R event. This change did not pose a USQ nor does it require a change to the TS. (SE 2000-0062)

DCN 83-135: WEST 110E018 Sheet 4, Spent Fuel Pool Cooling System.

A spent fuel cooling system walkdown revealed several inconsistencies between the installed field equipment and the FSAR drawing WEST 110E018, Sheet 4. This change revises the drawing to reflect the actual plant configuration.

Summary of Safety Evaluation: There were several valves that were shown as the wrong type and some valves that were shown normally shut that are actually normally open. Also, there were some valves and instrumentation tubing that were not shown on the drawing that should be and was installed under modification or original design. There were additional problems such as nozzles shown in the wrong direction or not included, valves without labels, and pipe runs without proper designation or size. Finally, there is a cooling line for the P-13 skimmer pump mechanical seal that was not shown correctly and the priming connection that connects the Z-402 priming chamber and the pump casing.

The changes to the drawing have been evaluated and installed as original plant equipment, under an approved modification, or provide a clarification to the drawing. This was determined from a review of old modification packages, age of equipment, proper use of valve and pipes, and review of other documents such as DCNs. All of the equipment not shown on the drawing or shown as a different type meets the design requirements for the spent fuel cooling system for temperature, pressure, materials, seismic, and quality. The changes do not change the risk model for the system. The drawing change does not increase the probability of an event already described in the CLB. The drawing change does not increase the probability of a malfunction of equipment important to safety. The drawing change does not increase the radiological consequences of a malfunction of equipment. The drawing change does not increase the possibility of an event not already evaluated in the CLB. The drawing change does not increase the probability of a malfunction of equipment not already evaluated in the CLB. Finally, this change does not pose a USQ, nor require a change to the TS. (SE 2000-0089-01, which supersedes SE 2000-0089 in its entirety.)

<u>DCN 2000-1012</u>: Bechtel 6118M-201, Sheet 3, Main Steam.

This change deletes the low point ¼-inch capped drain line, between valve CS-38D and the steam generator blowdown (SGBD) heat exchanger (HX) cooling water header, from drawing Bechtel 6118 M-201, Sheet 3 to reflect actual plant configuration.

<u>Summary of Safety Evaluation</u>: The change was initiated by CR 99-2822. The engineering evaluation of that CR, determined that this ¼-inch capped drain line is not required and does not fulfill a design requirement for the affected piping. In addition, since this ¼-inch pipe drain line is not installed, it should be deleted from the drawing Bechtel 6118 M-201 Sheet 3. This pipe drain line does not fulfill a design requirement. In addition, modification MR 99-021*A re-routed the affected piping in a way which eliminated any low point drain for this portion of the piping. The modified portion of the affected cooling water piping is now self-draining. Therefore, a low point for this portion of the affected piping is not needed and was deleted from the drawing Bechtel 6118 M-201, Sheet 3.

The SGBD HX cooling water system is not discussed in the CLB as an accident initiator or contributor to an accident or event scenarios previously evaluated in the CLB. In addition, the PBNP CLB does not specifically discuss the design details of the SGBD HX cooling water piping, thus, making this line a non-critical component of the SGBD system. The SGBD HX cooling water system does not participate in radiological release mechanisms for the PBNP, and no new radiological release mechanisms or path are created. Based on that, this activity did not increase the probability of occurrence of an accident or the radiological consequences of an accident or malfunction as previously evaluated in the CLB. The low point capped drain line is a non-safety related, non-seismic portion of the SGDB HX cooling water system, and its deletion from the drawing does not have any impact on SGBD system or any other plant equipment. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB. The activity does not degrade the ability to provide cooling water to the SGBD heat exchangers. The change does not pose a USQ nor does it require a change to the TS. (SE 2000-0059)

DCN 2000-1807: Relocation of Blowdown Evaporator (BDE) Bottoms Loop Valves on Drawing PBM-225

Drawing PBM-225 shows a tee that can allow bottoms from the BDE to go to the shipping container or back to the BDE feed. Two isolation valves and a flush line isolation valve are shown to be downstream of the tee in the portion of piping leading back to the BDE feed. The three valves are actually upstream of the tee.

Summary of Safety Evaluation: The blowdown evaporator (BDE) process liquid wastes. The distillate is stored in tanks for eventual discharge. The bottoms that are left behind in the evaporator are occasionally batched out for processing and disposal offsite. The bottoms are moved from the BDE to the shipping container via the bottoms loop. The drawing was corrected to reflect the actual plant configuration. Drawing PBM-225 is used in the FSAR as Figure 11.1-2, "Blowdown Evaporator System."

The change does not involve a USQ since the BDE system and its components are not safety related, are not important to safety, and do not have an impact on safety related equipment. Any possible accident or event related to the BDE system and its components are bounded by existing analyses described in the CLB. In addition, no TS are applicable to this change, so no TS will be changed or affected. (SE 2000-0083)

DCN 2000-1871: Addition of pipe cap downstream of CC-775B on WEST 110E029, Sheet 3

The change to the plant is the installation of a pipe cap downstream of CC-775B. Additionally WEST 110E029, Sheet 3 is changed to add this cap.

Summary of Safety Evaluation: This connection is labeled as a local sample point and is not typically used. The addition of the pipe cap establishes a double barrier on this system penetration and the effect of the addition is limited to the approximately 6" tail piece of the system isolation valve. Therefore this change is minor in scope, does not affect other systems structures or component not associated directly with the change and conservatively provides an additional boundary for this closed system. Westinghouse drawing 110E029 Sh 3 was changed to reflect this change in plant configuration. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0084)

DCN 2000-2056: Change FSAR Figure 11.2.1, Waste Gas System

The gas analyzer sample pump is part of the waste gas (WG) system and helps to measure hydrogen and oxygen levels in various primary system tanks. This pump is shown on drawing WEST 684J972 Sheet 3, Gas Analyzer. The drawing is also shown in the FSAR as Figure 11.2-1 Sheet 3, "Waste Gas Disposal System Process Flow Diagram." The pump is shown as a centrifugal pump on these drawings. This is incorrect. This pump is not and has never been a centrifugal pump. The pump is a positive displacement pump. The drawings were changed to correct the error. No physical change is being made to the plant.

Summary of Safety Evaluation: The gas analyzer sample pump is not safety-related nor is it important to safety. No accidents in the CLB directly pertain to the sample pump and any possible accidents or events not proposed in the CLB are bounded by existing analyses of the WG system.

The drawing change involves the gas analyzer sample pump which is not safety related, is not important to safety, and does not have an impact on safety-related equipment. Any possible accident or event related to the gas analyzer sample pump or the waste gas system is bounded by existing analyses described in the CLB. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0116)

DCN 2000-2267: Addition of 1LIC-3586 & 2LIC-3586 Forebay Screen Auto Start Level Indicator Controllers

The change is to document the as-built field configuration of the instrumentation sensing line routing for 1LIC-3586 & 2LIC-3586, forebay screen auto start level indicator controllers.

Summary of Safety Evaluation: The field routes of the high and low-side differential pressure sensing lines for the level indicators are different than depicted on permanent drawing FSAR Figure 10.1-6B (Bechtel M-212 Sheet 2). The drawing/FSAR figure depict one high-side sensing line for each instrument. The actual condition in the field allows the instruments to be connected to either one of two available high-side sensing lines. The drawing was corrected to depict the actual field routing of the low-side sensing lines to the 1-Z27-1 and 2-Z27-4 traveling screens.

The change to the sensing line locations does not affect the operation of the operation of the traveling screens or the operation of associated instrumentation. The change enhances system maintenance, thus, system performance is improved by increased availability. All materials installed are of the same type and construction. The change does not affect the operation, function, or method of performing a function of a SSC as described in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 2000-0119)

FPER Section 2.3.6: Removal of FPER Inspection Frequency Statement

This safety evaluation removes certain inspection frequencies, nor required by regulatory commitments, from Section 2.3.6 of the FPER.

Summary of Safety Evaluation: Information specific to inspection frequencies performed in accordance with NP 1.9.6, "Removal of FPER Inspection Frequency Statement in Section 2.3.6" is included in the second paragraph of FPER Section 2.3.6, however, these frequencies are not required by regulatory commitments. Specifically, FPER Section 2.3.6 states that plant inspections are conducted on a "weekly basis". The corresponding regulatory commitment statements use the words "frequent" and "regular" to describe plant inspection frequency. Review of the original commitment documentation indicates that there is no basis for requiring "weekly" inspections. Inspections should be conducted "frequently" on a "regular" basis.

Also included in this paragraph is language specific to the conduct of these inspections. The detail included in this additional information is likewise not required by the regulatory commitment but is specific to procedure implementation. Therefore, the second paragraph of FPER Section 2.3.6 is deleted. Removal of this language provides greater latitude in procedure implementation while maintaining the intent of the original commitment.

The change involves the deletion of unnecessarily detailed procedural information from the FPER, which has no basis with respect to the regulatory commitment. The change does not impact the physical plant and therefore has no impact on equipment important to safety. The commitment to perform the inspections discussed in this FPER section stems from statements made in the PBNP 1977 Fire Hazards Analysis and the NRC response as documented in the FPSER, dated August 2, 1979. The commitment made in these documents (e.g. conduct frequent/regular plant tours, by plant management and staff, to ensure plant cleanliness, housekeeping, and that combustible materials are not stored in safety-related areas), is maintained in FPER Sections 2.3.6 and 2.3.2 after this change. Therefore, removal of this information does not remove reference to committed actions in the FPER; it provides the plant latitude to revise implementing procedures to conduct plant inspections based on observed plant conditions. This change does not pose a USQ nor does it require a change to the TS.
(SE 2000-0017)

<u>FPER Section 7.2.2</u>: Operability of Fire Protection Systems

The change revised compensatory measures for inoperable fire pumps contained in FPER Section 7.2.2, "Operability of Fire Protection Systems" and OM 3.27, "Control of Fire Protection & Appendix R Safe Shutdown Equipment".

<u>Summary of Safety Evaluation</u>: The administrative controls in these documents specify that both fire pumps are required to be operable. If one fire pump is inoperable, the second fire pump must be tested to demonstrate operability and must be tested every 24 hours thereafter. Fire pump operability requirements are revised so there are no compensatory measures required when one of the two fire pumps is inoperable. The requirement to test the operable fire pump every 24 hours is a holdover from PBNP TS. The purpose for the specification was to ensure high availability and reliability of the fire pumps. Availability and reliability of the fire pumps is now monitored under the requirements of the maintenance rule (10 CFR 50.65). Under the Maintenance Rule, the number of functional failures of the fire pumps and fire pump availability is monitored against performance criteria established based on probabilistic safety assessment (PSA) assumptions.

Requiring the remaining operable fire pump to be started and run every day increases the amount of wear and tear on the fire pumps and therefore increases the probability for a component failure. Deleting the requirements to start the operable pump every 24 hours will result in less wear on the fire pumps between maintenance overhauls, and will therefore, enhance reliability. The FPER and OM 3.27 were revised to no longer require the operable fire pump to be started when the other pump is inoperable. Each fire pump is tested monthly in order to ensure it remains capable of performing its safety function. When one of the fire pumps is inoperable, only the remaining operable pump is to be tested on its normal monthly frequency

The only event evaluated in the CLB involving either the electric or diesel fire pump is a fuel oil fire originating at the diesel fire pump. After being identified as a concern for the surrounding service water pumps, the pumphouse was modified to reduce the risk from fire. A partial-height wall was installed to separate the service water pump area into two halves. Additionally, a wet pipe sprinkler system was installed in the service water pump area in order to prevent the spread of fire between the north and south pumphouse area.

Since the diesel fire pump will run less often, it will be less likely to initiate a fire in the pumphouse. Therefore, the probability of occurrence of an accident or event is decreased. The TS no longer contains fire pump operability or testing requirements. Fire protection system operability and surveillance requirements were transferred from the TS to the FPER in 1997. This transfer was consistent with the guidance for GL 86-12, "Removal of Fire Protection Requirements from the Technical Specifications." Since the TS no longer contain requirements for the fire protection system, there can be no reduction in the margin of safety defined in the basis for any Technical Specification. This change does not pose a USQ nor does it require a change to TS. (SE 2000-0117)

FPER Section 9: Relocation of FPER Contained Fire Protection Technical Evaluations (FPTE) to the Fire Protection Technical Evaluation Manual

FPER Section 9, contains eight technical evaluations which must be removed in support of the FPER revision efforts via the Appendix R Rebaselining Project

Summary of Safety Evaluation: The eight evaluations are:

- FPTE-002, "Technical Evaluation of Inadvertent Suppression System Actuation at Point Beach Nuclear Plant."
- FPTE-003, "Technical Evaluation of Emergency Lighting Capability at Point Beach Nuclear Plant,"
- FPTE-004, "Technical Evaluation of Fire Damper Tests at Point Beach Nuclear Plant."
- FPTE-005, "Technical Evaluation For Appendix R Hot Shutdown Components Out of Service."
- FPTE-006, "Technical Evaluation of Fire Detector Location Plan at Point Beach Nuclear Plant."
- FPTE-007, "Technical Evaluation of PBNP Point-to-Point Portable Radio Communications for an Appendix R Fire."
- FPTE-008, "Technical Evaluation for Appendix R Cable Separation in the Auxiliary Feedwater Pump Room Fire Zone 304."
- FPTE-009, "Technical Evaluation for Removal of the Foam System on the Outside Fuel Oil Storage Tanks T-32A/B Fire Zone 576."

FPTE-001, "Technical Evaluation of Fire Barrier Penetration Seals, Fire Rated Wrapping and Cable Tray Fire Stops at Point Beach Nuclear Plant," was previously removed from the FPER via SE 2000-0094

The evaluations are being moved from the FPER to a controlled reference document entitled "Fire Protection Technical Evaluation Manual." These documents provide design basis as opposed to licensing basis information; the FPER is a licensing basis document, and as such, should not be a repository for design basis information. Relocation of this information to the new manual allows for its reference, update and retention while preventing confusion of this information with the Current Licensing Basis. These documents will be reformatted and assigned a Technical Evaluation number to facilitate this change.

In each case the evaluations were prepared to support assertions/commitments made in the FPER. Removal of this information from the FPER does not change original commitments. This information becomes a controlled reference document similar to an engineering evaluation or calculation. Because these evaluations were written to validate compliance with commitments made in the CLB they are considered to be design rather than licensing basis. Therefore, relocation of this information does not adversely affect the ability to achieve and maintain safe shutdown of the plant in the event of a fire as required in Facility Operating Licenses DPR-24 and DPR-27 for Unit 1 and Unit 2, respectively) Condition 3.H. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0128)

FPER Section 9, Tab 9.4: Technical Evaluation of Fire Barrier Penetration Seals, Fire Rated Wrapping and Cable Tray Fire Stops at Point Beach Nuclear Plant.

FPER Section 9, Tab 9.4 is being moved from the FPER to a new controlled reference manual titled "Fire Protection Technical Evaluation Manual."

Summary of Safety Evaluation: This change is being made because the document provides design basis as to opposed to licensing basis information. The FPER is a licensing basis document and as such should not be a repository for design basis information. Relocation of this information to the new manual allows for its reference, update, and retention while preventing confusion of this information with the Current Licensing Basis. The document will be reformatted and assigned a technical evaluation number.

Relocation and revision of the information contained in the Technical evaluation does not change the current licensing basis concerning fire barrier penetration seals. The change removes design basis information from the CLB and appropriately relocates it to a controlled reference document that facilitates its reference, update and retention. Removal of this information from the FPER ensures that it is not confused with licensing basis information. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0094)

FSAR: Deletion of Red Lock Indications from FSAR and Plant Drawings

The change removes red lock indications from plant drawings and FSAR figures. The red lock is an administrative control that prevents mis-positioning of plant components. Red lock use and control is administered by NP 2.1.3, "Administrative Control of Valves, Locks, and Switches," and OM 3.12, "Control of Equipment and Equipment Status." This change does not result in a physical change in component positions, but removes red lock indication from plant drawings and associated FSAR figures.

Summary of Safety Evaluation: This change removes red lock indications from PBNP drawings and FSAR figures. A red lock is an administrative control. This change does not result in physically changing any component position, but removes red lock indications from all drawings and associated FSAR figures. Red locks are not being physically removed as a result of this change. Plant alignment and operational procedures are written in accordance with the guidance provided in the procedures for removal and installation of red locks when required.

Since there is no change in any system alignment or operation, there is no affect on existing accident or incident analysis. Normal valve positions remain indicated on the Drawings and FSAR figures with only locked position indications removed. Note that FSAR Drawings do not reflect actual plant configuration under different operating modes. This change does not pose an USQ nor does it require a change to the TS. (SE 2000-0007-01, which replaces SE 2000-0007 entirely)

FSAR Sections A.5.2 and 7.4: Seismic Design Classification of PAB Ventilation and Main Feedwater Equipment That Isolates the Supply of Main Feedwater to the Steam Generators

The change revises FSAR Section A.5.2. The seismic class of the PAB ventilation system and the main feedwater equipment that isolates the supply of main feedwater to the steam generators is revised from Class I to III. In addition, FSAR Section 7.4.1.3.k is revised to remove the statement that the limit switches on the main feedwater regulating valves are seismically qualified.

Summary of Safety Evaluation: The main feedwater equipment that isolates the supply of main feedwater to the steam generators was determined to have a safety function of isolating main feedwater flow to the steam generators after a main steam line break. It is not required to consider a steam line break as a result of a seismic event. Therefore, during and after a seismic event, the safety function of the equipment relied upon in the accident analysis is not required. It was therefore concluded that the change in seismic design classification for the valves would not affect the probability or consequences of an accident and would not increase the probability or consequences of a malfunction of equipment or consequences of an accident and would not increase the probability or consequences of a malfunction of safety equipment. FSAR Section 9.5.1 states that the PAB ventilation system is not required to perform a safety function and that no credit is taken for the system in accident analyses or habitability studies for the filtration capability. Based on this, it was concluded that the seismic design classification change for the PAB ventilation system does not affect the probability of an accident, malfunction, or equipment or consequences of an accident. The change in seismic design classification does not create accidents not previously analyzed in the CLB. The PAB HVAC equipment is not relied upon to perform a safety function or mitigate an accident. The change in design classification does not create a possibility of a malfunction of equipment important to safety different that that previously evaluated in the CLB. The change does not add new equipment to the plant nor does it revise the safety of the equipment. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0014)

FSAR Section 3 and 14: Reactor and Safety Analysis Concerning Unit 2 Cycle 25 Reload

This safety evaluation covers all modes of operation for the Unit 2 Cycle 25 (U2C25) core loading pattern. Unit 2 was recently operating in Cycle 24 with 121 Westinghouse 14x14 upgraded optimized fuel assemblies (OFAs). For Cycle 25, Westinghouse 14x14, 422 VANTAGE+ (422V+) fuel assemblies were introduced into the Unit 2 core. Forty OFAs were replaced with 20 fresh Region 27A (4.20 w/o U-235), and 20 fresh Region 27B (4.60 w/o U-235) 422V+ fuel assemblies. .

Summary of Safety Evaluation: The 422V+ fuel product was licensed under the Fuel Criteria Evaluation Process (WCAP—12488-A) and is supported by new analyses which have been accepted by the NRC. License Amendments 193 (Unit 1) and198 (Unit 2) approved the use of the 422V+ fuel product and cover required TS changes. A separate 10 CFR 50.59 evaluation (SE 2000-0070, under MR 99-069*A) has been approved that describes the use of the 422V+ fuel product, and evaluates some additional changes related to the 422V+ fuel upgrade program. Therefore, evaluations that are related specifically to the use of the new fuel upgrade program, and not to the core reload, are not addressed by this safety evaluation.

Westinghouse reported the results of the cycle design and safety analyses in the Point Beach Nuclear Plant Unit 2 Cycle 25 Final Reload Safety Evaluation. Westinghouse concluded the Cycle 25 design results in no unreviewed safety questions or TS changes provided the following conditions are met:

- The End-Of-Cycle 24 burnup is bounded by 19,370 to 20,870 MWD/MTU;
- The U2C25 burnup will 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 Cycle 25 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 the TS;
- The safety aspects on the reactor internals of using peripheral power suppression assemblies (PPSA) have been assumed by Wisconsin Electric (Nuclear Management Company, LLC)

The effect of the U2C25 reload core design on the Boron Dilution Event in Cold Shutdown is assessed to be acceptable in calculations 99-0033 and 00-0033.

The U2C25 reload core design meets all 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 all pertinent licensing basis acceptance criteria are met. Though fuel and core design are not directly related to the probability of any previously evaluated accident or event, 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. This calculation is valid as long as the core does not return to critical following a SLB. Westinghouse has performed an evaluation that confirms Unit 2 Cycle 25 core will not return to critical following a SLB. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0113)

FSAR Sections 4.0-4.4: "Reactor Coolant System"

FSAR Sections 4.1, 4.2, 4.3, and 4.4 changes provide editorial changes, clarification of wording, revising technical information to agree with plant modifications, and deletion of extraneous information. The change does not involve a physical change to the plant.

Summary of Safety Evaluation: A summary of some of the technical revisions to these sections were:

- Tables 4.1-3, 4.1-5 & 4.1-6 added the current operating pressure of 1985 psig.
- Section 4.1, Page 4.1-10 Heatup and cooldown #2, Revise to "slower initial heatup rates attainable from pump energy and pressurizer heaters only."
- Table 4.1-1, Table 4.1-2/Table 4.1-1 gives two operating pressures 1985 or 2235 $\pm$ 100 and two low pressure trip points for the different operating pressures. Spray valves start to open at either 2010 or 2260 psig depending on the normal operating pressure. Table 4.1-2 should also give both operating pressures of 1985 or 2235 since there are two possible operating pressures.
- Table 4.1-3, Table 4.1-3 Maximum heatup rate of RCS using heaters only (approximately 55 degrees/hour is more accurate for pumps and heaters, from plant operating history).
- Section 4.2, page 4.2-4 Revise to "During a positive surge caused by an increase in RCS temperature, the pressurizer spray system..."
- Section 4.2, Page 4.2-4 Pressurizer discussion to "... the pressurizer spray system, which is fed from the cold leg of each coolant loop, operates to condensate steam..."
- Section 4.2, Page 4.2-4, Third paragraph, First sentence. Change to "During a negative pressure surge caused by decreasing RCS temperature, water in the pressurizer flashes to steam to mitigate the pressure drop, and heaters automatically actuate to restore RCS pressure to normal."
- Section 4.2, Page 4.2-24 Added brief descriptions of reactor vessel level indication system and isolated piping thermal relief protection. Added cross reference to section 7.2.3.2 for RTD bypass loops
- Section 4.2, Update references Pages 4.2-23 #5 and #6 relate to turbine discs. Added ("disc type" LP turbine spindles have been replaced with spindles of mono block design).
- Section 4.4, Update Table 4.4-3, Reactor Vessel Surveillance Capsule Removal Schedule. Capsule P was removed June 8, 1997.
- Section 4.3. Page 4.3-1 Reactor Vessel. Second paragraph, first sentence. Deleted "...now in service..." because although the 'operating cycles estimates' were obtained from Yankee Rowe, which has since been taken out of service.
- Section 4.2. Page 4.2-24. Added, "The system can also be used to reduce primary system pressure at hot shutdown allowing boration of the RCS using high head safety injection pumps."

The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0031-01, which supersedes SE 2000-0031 in its entirety.)

FSAR Section 5.1: "Containment System Structure"

The change provides editorial changes and clarification or wording, revises technical information to agree with the design bases, and deletes extraneous and historical information. The change does not involve a physical change to the plant.

Summary of Safety Evaluation: The following summarizes the technical changes:

- Deleted a sentence because the stated BTU's do not agree with those listed in FSAR Table 14.3.4-2, and because it is not necessary to state accident energies in Section 5.1.1.3.
- Added the code years for the IEEE an ASME Codes to clarify what codes were used during design and construction. FSAR Section 5.1.1.5, Item 2 lists IEEE-317, April 1971 version. However, the 1976 version was also used as specified in PBNP Electrical Penetration Specification PB-102 (November 1980) and 257 (May 1986). For ASME Section III, FCR 99-008 and DBD-33, Section 2.2.7, state that the 1965 Edition was the code of record. In addition, FSAR Page 5.1-63 states that the 1968 Edition of ASME III and all Addenda were used for the design, fabrication, inspection, and testing of the containment penetration head fittings.
- Clarified that expansion bellow seals are not required on containment barriers inside containment. Expansion bellow seals are used on containment penetrations outside containment per drawings M-82 and C-124 and FSAR Figure 5.1-2, and as described in Section 2.2.3.b of the Containment Leak Rate Testing Program Basis Document. This change conforms the FSAR to the plant configuration.
- Revised the external pressure load design condition discussion to be consistent with Westinghouse letter PBW-WMP-105 dated December 23, 1966.

This change provides editorial and technical changes to FSAR Section 5.1 The design of the plant systems is not affected by the change. Also, the change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-005-01, which supersedes SE 2000-005 in its entirety.)

FSAR Section 5.3: "Containment Ventilating System"

The FSAR Section 5.3 changes provide editorial changes and clarification of wording, revising technical information to agree with the design bases, and deletion of a figure. The change does not involve a physical change to the plant

Summary of Safety Evaluation: The following summarizes the technical changes:

- Clarified the containment fan cooler post-accident heat rate to include the limitation "without boiling of the service water." Deleted the cooling water (SW) flow rate and temperature for the limiting design basis accident condition, and reworded the sentence describing the original heat rate to delete the cooling water flow rate and temperature. It is not necessary to state the flow rate or temperature in connection with the design heat rate. The change makes the discussion FSAR Section 5.3 consistent with the containment air recirculation cooling system post-accident performance requirements described in FSAR Section 6.3.1.
- Revised a sentence to state that the purge supply and exhaust ducts' butterfly valves, both inside and outside containment, are locked closed during normal operation. This is according to TS Section 15.3.6.A.1.c and OP 9C, Section C, Paragraph 2.1. The change makes the FSAR consistent with the plant configuration.
- Changed "30,000 cfm" to "34,150 cfm" for the capacity of the auxiliary building exhaust ventilation system filters. The basis for this change is Farr Component Instruction Manual No. 370 which gives the filter capacity as 34,150 cfm.
- Changed the heat removal rate for the reactor cavity cooling coils from "342,000 BTU/hr" to "368,000 BTU/hr" in accordance with Bechtel Specification M-37, Revision 0.
- Changed the containment cleanup fan capacity from "5,000 cfm" to "5,400 cfm" per WEST Component Instruction Manual, Control #130.

The change conforms the FSAR to the plant configuration. This change provides editorial changes and clarification of wording, revision of technical information to agree with the design bases, and deletion of a Figure in FSAR Section 5.3. The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0020-01 which supersedes SE 2000-0020 in its entirety.)

FSAR Section 5.5: "Minimum Operating Conditions"

FSAR Section 5.5 was revised to agree with the design basis, and delete unverifiable information. The change does not involve a physical change of the plant

Summary of Safety Evaluation: The following summarizes the changes.

- Revised the external pressure load design condition discussion to be consistent with Westinghouse Letter PBW-WMP-105 dated December 23, 1966.
- Deleted a statement that the containment 2 psig vacuum condition is specified as an operating limit to avoid any difficulties with motor cooling. The 2 psig vacuum condition is a containment structural design limit only. It is not specified in the design or licensing basis as an operating limit to avoid any difficulties with motor cooling.

The design of the plant systems is not affected by the change. Also, the change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ, nor does it require a change to the TS. (SE 2000—0006-01, which supersedes SE 2000-0006 in its entirety.)

FSAR Table 5.6-2: Inventory of Aluminum in Containment, Reactor Coolant Pump Cubicle Smoke Detectors

This change deletes references to RCP cubicle smoke detectors in FSAR Table 5.6-2.

Summary of Safety Evaluation: The RCP cubicle smoke detectors listed as containing aluminum were installed in 1976 (E-122 & E-123). Later they were removed and new detectors were installed throughout the Unit 1 and Unit 2 containments (E-255). The new detectors did not contain aluminum in their components or composition. MR 97-110*E2 (Unit 1) and MR 97-110*B (Unit 2) subsequently replaced the smoke detectors located throughout the containments, including the RCP cubicle smoke detectors. None of the new detectors contain aluminum products.

This activity involves changing the FSAR documentation to reflect the current plant equipment configuration. The RCP cubicle smoke detectors' function will not change as a result of the update to the FSAR. The RCP cubicle smoke detectors are not required for or assumed in any CLB accident scenario. No Appendix R safe shutdown equipment is affected. Changes will not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0127)

FSAR Section 5.7: "Tests and Inspections"

The FSAR Section 5.7 changes provide editorial changes and clarification of wording, and revising technical information. The change does not involve a physical change to the plant

Summary of Safety Evaluation: The following summarizes the technical changes:

- Added a sentence to state that the pressure boundary integrity of the emergency core cooling systems outside containment is monitored by the leakage reduction and preventive maintenance program (FSAR Section 6.2.3). Revised a sentence to state that the measured deflections were compared with only one other containment structure, in accordance with Containment Structural Test Report B-SIT-5 for PBNP Unit 2. The change clarifies the FSAR and makes it consistent with the test reports.
- Corrected a sentence by replacing "two different and large loading conditions" with "test pressure." The only load applied to the containment during the Structural Integrity Test (SIT) was test pressure. No other load was added to the pressure.
- Revised the quantity and description of tendons inspected to agree with Technical Specification 15.4.4.II.
- Revised the description of the tendon surveillance program to agree with Technical Specifications 15.4.4.II.B and 15.4.4.II.C.
- Changed the paragraph before the description of the inspection intervals, to include the other tendons that are selected, not just the vertical tendons. The paragraph is revised to read: "The inspection of the randomly selected tendons is sufficient to indicate any tendon corrosion that could possibly appear." The basis for this statement is RG 1.35, Revision 3, and TS 15.4.4.II which is based on RG. 1.25, Revision 3 as stated in the Basis section of the TS.

The design of the plant systems is not affected by the change. Also, the change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0021-01, which supersedes SE 2000-0021 in its entirety.)

<u>FSAR Figure 6.2-4</u>: "Safety Injection Pump Performance Characteristics"

Figure 6.2-4 of the FSAR shows the high head SI (HHSI) pump performance curve and the 5% pump head degradation curve (labeled as "Reduced Performance Used in Analysis"). The change revises Figure 6.2-4 to change the 5% degraded curve to 10% pump head degradation curve that was developed for the fuel upgrade (422V+) analyses. In addition, the note indicating the maximum runout flow will be removed from the figure.

<u>Summary of Safety Evaluation</u>: The change revises FSAR Figure 6.2-4 "Safety Injection Pump Performance Characteristics" and associated text in FSAR Section 14.3.1, "Small Break Loss of Coolant Accident." The change also allows the IST program to begin testing the SI pumps using acceptance criteria based on 10% pump head degradation.

The 10% pump head degradation curve is used in the new fuel upgrade/uprating analyses, along with a PBNP-specific SI system flow model, to generate a new minimum SI flow versus RCS pressure curve used in the accident analyses. As a result of using the PBNP-specific SI system flow model, the resulting SI flow versus RCS pressure curve is bounded by the curve used in the existing analyses for the fuel. Therefore, revising the "Reduced Performance" curve in the FSAR Figure 6.2-4, and allowing a revised IST acceptance criteria based o the 10% pump head degradation remains analyzed by the current accident analyses for OFA fuel, and is acceptable. Note that the full implementation of new safety analyses based on the 422V+ fuel upgrade program will become effective as of U1C27 and U2C25. The revised FSAR figure is applicable for OFA fuel, 422V+ fuel, and transition cores. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0095)

<u>FSAR Section 6.2</u>: "Safety Injection System (SI)"

The FSAR Section 6.2 changes provide editorial changes and clarification of wording, revising technical information to agree with the design bases and plant configuration, and deletion of extraneous and historical information. The changes do not involve a physical change to the plant.

<u>Summary of Safety Evaluation:</u> The changes are summarized as follows:

- Revised a sentence to delete the 6'-6" depth of water in containment after an accident and the gallons of water that contribute to that depth because the depth of water is not conservative and is not substantiated by calculation. CR 99-1581 substantiates via calculation the depth of water that would accumulate in containment sump after a LOCA.
- Added sentences to describe the most severe single failures assumed for the SI system which are described in the Accident Analysis Basis Document, Section 15.4.7 (SBLOCA) and 16.4.7 (LBLOCA).

- Changed the maximum RHR operating temperature during recirculation from 210°F to 250°F to agree with the Thermal Modes Evaluation Report (S&L Calculation M-89992-02.TM), Calculation N-90-067, and Westinghouse letter WEP-90-162, dated August 23, 1990.
- Changed the "Max. Flow Rate" to "Runout Flow Rate", and "1230 gpm" to "1233 gpm" for the safety injection pump. The pump was tested by Byron Jackson at a maximum of 1233 gpm per Westinghouse memo PA-EME-680, dated October 1, 1970.
- Deleted the NPSH curve on the SI pump curve figure because this information is not needed or referred to the FSAR. The curve on the figure does not match the curve in the pump component instruction manual (Byron Jackson Manual 262, Revision 19).

The design of the plant system is not affected by the changes. Also, the changes do not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000—013-01, which supersedes SE 2000-0013 in its entirety.)

<u>FSAR Section 6.3</u>: "Containment Air Recirculation Cooling System (VNCC)"

The FSAR changes provide editorial changes, clarification of wording and revising technical information. The changes do not involve a physical change to the plant.

<u>Summary of Safety Evaluation:</u> The changes are summarized as follows:

- Clarified the containment fan cooler post-accident heat rate to include the limitation "without boiling of the service water."
- Deleted the cooling water (SW) flow rates, fouling factors and temperature.
- Added a paragraph to describe the functional testing performed per TS 15.4.5.1.C that each fan cooler unit is tested periodically to verify proper operation of the accident fans, backdraft dampers and service water bypass valves. The statement was added for completeness because the subsection, "System Testing" only described the testing that was done after the original installation of the fans.

The design of the plant systems is not affected by the change. Also, the change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 000-0019-01, which supersedes SE 2000-0019 in its entirety.)

<u>FSAR Section 6.4</u>: "Containment Spray System"

The changes to FSAR Section 6.4 provide editorial changes, clarification of wording, revising technical information to agree with the plant procedures and as-built plant configuration, and deletes information in FSAR Section 6.4. The changes do not involve a physical change to the plant.

Summary of Safety Evaluations: The changes are summarized as follows:

- Revised the initial conditions for the use of the containment spray pumps during the recirculation phase to be consistent with the initial conditions listed in Procedure CSP-Z.1, Revision 13. These conditions first appeared in Revision 13 of CSP-Z.1. SCR 99-0549 stated that Revision 13 was based on Revision 1C of the WOG ERGs, dated September 30, 1997. Therefore, the justification for this change is Revision 1C of the WOG ERGs.
- Changed the design head and the shutoff head for the containment spray pumps to be consistent with the values in the pump data sheets and pump curves provided in Ingersoll Rand containment spray pump component instruction manual, Revision 11.
- Changed the material listed of r the spray additive tank to be consistent with the tank material provided in Westinghouse component instruction manual, Revision 1, and drawing 685J119.
- Replaced the containment spray pump curve figure to show the pump performance curve of a containment spray pump in the Ingersoll Rand component instruction manual, control #143. The original FSAR Figure showed a shutoff head of 575 ft vs. the instruction manual curve that shows a shutoff head of 550 ft.

The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0009-01, which supersedes SE 2000-0009 in its entirety.)

FSAR Section 6.5: "Leakage Detection Systems"

The change revised the dimension of the auxiliary building (air ejector exhaust only) vent from 8" to 4" to agree with the plant configuration.

Summary of Safety Evaluation: The change does not involve a physical change to the plant. The change makes the FSAR consistent with the plant configuration. The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve new system interactions or connections and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0124)

FSAR Section 7.5.1.4: "Plant Process Computer System"

The changes to the FSAR Section 7.5.1.4 remove wording from the FSAR that the display keyboard stations be CRT based, and add additional data available in the control room.

Summary of Safety Evaluation: The design of the plant systems is not affected by the changes. The changes do not affect plant procedures, and do not create or revise tests or experiments. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0118)

FSAR Section 7.6.3: "Incore Instrumentation"

This change revises various technical information to FSAR Section 7.6.3.

Summary of Safety Evaluation: The changes are summarized as follows:

- Change "maximum chamber dimensions" to "approximate chamber dimensions" for the movable flux detectors to be consistent with plant documentation.
- Correct the time it takes to perform a core flux map.
- Corrects a statement about the measurement of hot channel factors to be consistent with TS.
- Corrects a statement about the measured nuclear peaking factors to be consistent with TS.

The design of the plant systems is not affected by the changes. The changes do not involve a physical change to the plant. The changes do not affect plant procedures, and no test or experiment is revised or crated. The changes do not involve new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0123)

FSAR Sections 8.7 and 9.5: "PAB Ventilation" and "125 V DC" (VNPAB and VNBI Systems)

FSAR Sections 8.7 and 9.5 changes provide editorial changes, clarification of wording, revising technical information to agree with plant procedures and as-built plant configurations, and deletion of information which has been considered to be excessive detail (as defined in NEI 98-03), "Guidelines for Updating Final Safety Analysis Reports" that was endorsed by RG 1.181, " Content of the Updated Final Safety Analysis Report in accordance with 10 CFR 50.71(e)." The change does not involve a physical change to the plant.

Summary of Safety Evaluation: The design of the plant system is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve new system interactions or connections and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0015)

FSAR Section 8.8: Diesel Generating System (DG) Figure 8-6and Westinghouse Drawing 541F152, Sh 4

This change is to correct Westinghouse drawing 541F152, Sheet 4 and FSAR Figure8-6 to agree with Bechtel drawing. E-1 (FSAR Figure 8-1) for the cable size and type shown for the cable from G01 to bus 2A05 and the cable from G02 to bus 2A05.

Summary of Safety Evaluation: The Westinghouse drawing shows the cable as 2-350 MCM cables per phase and the Bechtel drawing shows the same cable as 1-1000 MCM cable per phase. The correct configuration is 1-1000 MCM cable per phase and has been verified in the field as the original and technically correct plant configuration. The corrected cable size shown in Figure 8-6 of the FSAR is the proper size conductor for the main feed from generator G01 (or G02) to the emergency AC switchgear bus 2A-05. This is the originally installed cable that the plant was tested with during original pre-operational testing.

The 1-1/C-1000 MCM cable per phase that is installed has the proper voltage rating (5000 V) and adequate ampacity to carry the full continuous load output of the emergency generator. Also, the correct configuration is shown in Figure 8-1 of the same FSAR. There is no impact to the operation, or function of the emergency diesel generator G01 or G02. The cable is the same one that was installed when the plant was built and successfully passed the original pre-operational tests. The malfunction or failure modes for the emergency diesel generator and the 4160 V emergency AC power system are unchanged by this correction. This is true since the cable is adequately sized to allow the generator to provide continuous full output when needed. Also, the only failure modes of the cable itself (open circuit or short circuit) are identical to the ones that exist for the 2—350 MCM cables per phase as shown presently in the FSAR. This change does not pose an USQ nor does it require a change to the TS. (SE 2000-0088)

FSAR Section 8.8: "Diesel Generator System"

The changes to FSAR Section 8.8 incorporate technical information that is supported by design basis documentation and site procedures; and several editorial changes.

Summary of Safety Evaluation: These changes improve the contents of the PBNP in its description of the design, operating attributes, and capabilities of the emergency diesel generators and their associated auxiliary support systems found in Section 8.8. No additional commitments nor license requirements are created by the incorporation of these changes. No additional tests, experiments, or malfunctions are created; nor is there any increase in probability of any accidents or malfunction. No radiological consequences are increased. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0022-01, which supersedes SE 2000-0022 in its entirety.)

FSAR Figure 9.11-1: Change to Drawing 541F092 Sheet 1

The changes indicate the normal configuration of the Unit 1 primary sample system and adds more information to enhance system status control. The changes are consistent with current procedures dictating the use of the Unit 1 primary sample system.

Summary of Safety Evaluation: No changes in normal valve alignment occur. Changing the drawing to reflect normal conditions improved the status control of the sampling system, making system misalignments less likely. Status control is still accomplished by the same approved plant procedures. Accidents or events involving the primary sample system and the changes are bounded by the waste gas and waste liquid accidents evaluated in FSAR Sections 14.2.2 and 14.2.3, respectively. Because the operation of the sample system is not changed, the probability of an equipment malfunction, accident, or event is not increased. No new accidents or equipment malfunctions occurred. Changing the primary sampling drawing to reflect actual plant configuration does not increase the source term or radiological consequences of a previously evaluated accident. The valve position changes on the drawing do not affect containment isolation or RHR loop isolation valves. Because system operation is not changed, the ability to meet TS sample frequencies for sampling is not affected. No other TS is affected. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0004)

FSAR Section 9.2: "Heat Removal System"

Section 9.2 was revised to provide editorial changes, revising technical information to agree with the plant procedures and as-built plant configuration, and deletes information in FSAR Section 9.2. The change does not involve a physical change to the plant.

Summary of Safety Evaluation: The changes are summarized as follows:

- Changed "to 350°F and approximately 425 psig" to: "to less than or equal to 350°F and less than 400 psig," which are the RCS temperature and pressure conditions for placing the RHR system in service per OP-3C (Step 2.3.4), Revision 79, 10 CFR 50.59 screening 97-1251, and the Basis for TS 15.3.1.A.
- Revised a sentence to state that relief valve RH-861C discharges to the containment atmosphere, not to the containment floor drains, to reflect plant configuration per MR 89-179(Unit 2) and MR 89-191 (Unit 1) which made the changes.
- Changed the decay heat generation at 20 hours after shutdown from $31.7x10^6$ to $30.4x10^6$ to agree with Westinghouse Calculation RFS-W-19.
- Added "(WL System)" after "residual heat removal pump room sump pumps" to clarify that these components are part of the WL System, not the RH System. Also, deleted "per unit" because there are a total of two sump pumps for both units per CHAMPS.

The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0008-01, which supersedes SE 2000-0008 in its entirety.)

FSAR Section 9.3: "Chemical and Volume Control System"

Section 9.3 changes provide clarification of wording, revising technical information to agree with plant procedures and as-built plant configurations, and deletion of information. The changes do not involve physical change to the plant.

Summary of safety Evaluation: The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0043)

FSAR Section 9.4: Fuel Handling System

The change consists of revising a sentence to change "water" to "unborated water" in the description of the prevention of fuel storage criticality in FSAR Section 9.4. The change does not involve a physical change to the plant.

Summary of Safety Evaluation: The design of plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0122)

FSAR Section 9.6: "Service Water System"

Section 9.6 changes provide editorial changes, clarification of wording, revising technical information to agree with the plant calculations, and deletion of information. The changes do not involve a physical change to the plant.

Summary of Safety Evaluation: The changes are summarized as follows:

- Page 9.6-6. Section 9.6.3. second paragraph, Add the following as the new first sentence: "Service water pumps are normally controlled from main control room panel C-01."
- Page 9.6-6. Section 9.6.3. second paragraph, second sentence (old first sentence). Revise to "The service water pumps additionally have a local/remote switch located on the East wall in the East portion of the Auxiliary Feedwater (AF) pump area.
- Page 9.6-6. Section 9.6.3. second paragraph, third sentence (old second sentence). Change to "When the service water pump local/remote switches are placed in the local position; remote operation is bypassed, the auto starting features are disabled, and an alarm is received in the control room."
- Page 9.6-3. Section 9.6.2. Fifth paragraph. Change the third and fourth sentences to "normally, two of the six pumps are capable of carrying the required normal cooling load for the two units. During periods of higher lake temperatures or when RHR cooling is in service, operation of three pumps is normally required. During an accident, three pumps are required unless a service water TS Allowed Outage Time (AOT) is in effect at the time of the accident in which case, additional service water pumps running, the system valve lineup and positioning. Typical flow rates for the system in accident conditions vary from about 3,000 gpm to 21,000 gpm.
- Page 9.6-4. Section 9.6.2. ninth paragraph. second sentence to, "Following a loss-of-coolant accident, the service water system supply and return pressure for the ventilation coolers is normally below the containment design pressure of 60 psig."

The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS (SE 2000-0023-01, which supersedes SE 2000-0023 in its entirety.)

FSAR Section 9.6: "Service Water System"

The section was revised to describe the SW piping in the control room HVAC room as being Seismic Category 1. In so doing, Paragraph 3(a) in the NRC SER dated November 20, 1975, is removed from the CLB because the SER no longer applies to this piping. The change does not involve a physical change to the plant.

Summary of Safety Evaluation: The change is based on Piping Analysis Stress Report WE 30024 that analyzed the SW piping in the control room HVAC room and showed the piping to meet the criteria for Seismic Category 1. The change does not affect plant procedures or plant operations. No test or experiment is revised or created by the change. There are no new system interactions as a result of the change. System integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0048)

FSAR Section 9.6 and 9.9: Service Water System and Spent Fuel Cooling

The changes include controlling all redundant automatic isolation valves on non-essential SW loads to ensure automatic isolation capability is continuously available; increasing the maximum allowable SW temperature to 79°F for the Train A emergency diesel generators G-01 and G-02 (80°F for containment fan coolers and other supplied components); and relaxation of unnecessarily restrictive procedural controls (e.g. removing the prohibition against having two electro-hydraulic (EH) or turbine lube oil coolers in operation in parallel, restrictions against AFW supply line flushes, etc.)

Summary of Safety Evaluation: The changes are justified by and supported by the most recent analyses (Calcs 96-0059 Revision 5; 98-0172 Revision 1; and 97-0054 Revision 4) evaluating SW system performance under limiting accident (design basis) conditions. The completed installations of modifications to ensure positive automatic isolation of non-essential loads (additional modifications are pending but have not been credited in the referenced analyses) has afforded sufficient system margin to relax the itemized alignment restrictions while at the same time increasing the maximum allowable temperature.

A TS change was required because the TS requirements are not sufficiently conservative with respect to the analyses that form the bases for the activities. However, the activities are not in conflict with the TS. The activities do not represent a USQ. (SE 2000-0077)

FSAR Section 9.8: "Control Room Ventilation System"

The references to ANSI N510-1980 and ASTM D3803-1989 in FSAR Sections 9.8.4.2.b(1), 9.8.4.2.b(2), and 9.8.4.2.b(3) are clarified.

Summary of Safety Evaluation: The FSAR change indicates that control room emergency filter (F-16) testing is done in accordance with the methodology of ANSI N510-1980 Sections 10, 12 and 13, excluding sections 10.3 and 12.3. A clarification is also made that the tolerances of ASTM D3803-1989 will be applied for test temperature and humidity for laboratory analysis of a representative carbon sample in FSAR Section 9.8.4.2.b(3).

F-16 is not an initiator of an accident or event no matter what the effectiveness of the filter. The acceptance criteria in FSAR Section 9.8.4 will not be changed. The change will continue compliance with ASTM D3803. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0098)

FSAR Section 9.8: "Control Room Ventilation System"

This change provides editorial changes and clarification of wording, revision of technical information to agree with the design bases, and deletion of information in FSAR Section 9.8. The change does not involve a physical change to the plant.

Summary of Safety Evaluation: The changes are summarized as follows:

- Changed "HEPA" to "roughing" for filter F-43. F-43 is a roughing filter not a HEPA filter per the Farr component instruction manual and as confirmed by the System Engineer.
- Revised the description of flow switches to be consistent with M-144, Sheet 2, Revision 9, which shows a flow switch downstream of each fan, for W-14A,B and W-107A,B, vs. each set of fans.
- Deleted ±10% which was the temperature margin on 75°F because the design specification for the control room ventilation system, Bechtel Specification M-40, specified no margin on 75°F This change conforms the FSAR to the plant design basis.
- Deleted the heading "Control Room Ventilation Following Loss of Offsite Power" and the text under it (except for first paragraph) in accordance with the resolution of CR 99-2280 ("Potential FSAR Discrepancy on Control Room Habitability and LOOP vs. LOOP/LOCA Assumptions").
- Revised a sentence to add the word "significant" in the phrase, "following significant painting, fire or chemical release in the control room envelope." to be consistent with TS 15.4.11.4.a and HPIP 11.54.
- Changed "ASTM D3803" to "ASTM D3803-89" because TS 15.4.11 and the NRC SER for TSCR 192 dated July 9, 1997 state that the Code year for ASTM D3803 is 1989.
- Revised Figure 9.8-1 to clearly show the flow paths (darkened lines) for each mode of operation (some were hand drawn and illegible); to change Filter F-43 from a HEPA filter to a roughing filter; and to correct the FO/FC positions of two dampers to be consistent with positions shown on P&ID M-144, Sheet 2, Revision 9. Damper VNCOMP-4849A has a "FC" position and damper VNCOMP-4849G has a "FO" position per the P&ID. The changes clarify the figure and make it consistent with the plant configuration.

The design of the plant systems is not affected by the change. Also, the change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0026-01, which supersedes SE 2000-0026 in its entirety.)

FSAR Section 9.9: "Spent Fuel Cooling"

Section 9.9 was revised to change the quantity of the spent fuel pool skimmers in the FSAR from 2 to 1 and the spent fuel pool skimmer design flow rate in the FSAR from 50 gpm each to 100 gpm total. The changes do not involve a physical change to the plant.

Summary of Safety Evaluation: The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or crated. The change does not involve new system interactions or connections, and system integrity is maintained. Revising FSAR Section 9.9 does not result in a significant increase to occupational radiation exposure, or a significant unreviewed environmental impact, and no conflicts with a license condition as contained in the C of C is involved. This change does not pose a USQ nor require a change to the TS. (SE 2000-0125)

FSAR Sections 9.9 and 11.6: "Spent Fuel Cooling" and "Shielding Systems"

The FSAR descriptions for three items associated with the spent fuel pool (SFP), SFP cooling system were revised to make the FSAR description consistent with the as-built plant.

Summary of Safety Evaluation: The changes are summarized as follows:

- The refueling water circulating pumps (RWCP) total developed head, in FSAR Table 9.9.1, is changed from 50' to 150' water. Changing the RWCP total developed head does not increase the probability of malfunction of equipment important to safety that interfaces with the RWCP, because the piping systems supplied by the RWCP were designed for a RWCP capacity of 100 gpm at 150'. The only accidents, events or malfunctions that have radiological consequences where operation of the RWCP might be called upon is for "Loss of Decay Heat Removal" (Generic Letter 88-17) events. Changing the RWCP total developed head makes the FSAR consistent with our response to GL 88-17.
- The SFP low level alarm as described on FSAR Page 11.6-6 is changed from 59'-10" to 62'-8" to be consistent with the level setting described in the PBNP Setpoint Document Section 8.1. Raising the SFP low level alarm does not increase the probability of malfunction of equipment important to safety, because the higher setpoint alerts operators to low SFP levels sooner. This decreases the probability that a low SFP level leads to a loss of suction to the SFP cooling pumps, and decreases the probability for a fuel assembly to be excessively withdrawn
- The SFP volume as described in FSAR Table 9.9-1 was changed from a water volume of "47300 ft^3 Approx." to a pool physical volume of "48283 ft^3." The SFP water volume currently listed in FSAR Table 9.9-1 was not explicitly addressed as part of the NRC safety evaluation for the second SFP rerack (NRC SER dated April 4, 1979). This is consistent with the SER evaluation that considered a loss of a single train of SFP cooling as the evaluated event. SFP cooling can be temporarily interrupted when SW is isolated to the SFP heat exchangers, or if SFP water, fuel, racks and other equipment slows the heatup rate until SW and/or SFP cooling system flow can be reestablished. Changing the SFP volume from water volume to a descriptive physical volume does not change the fact that these SFP heat sinks exist to adequately slow the pool heatup rate when cooling is temporarily interrupted. The change therefore does not increase the probability of a malfunction of equipment important to safety previously evaluated in the CLB, because SFP water volume associated with the current SFP configuration was determined to be acceptable in NRC SER dated April 4, 1979, and a descriptive physical volume adequately describes the contents of the SFP.

The RWCP, SFP low level alarm annunciator, and SFP volume are not initiators to accidents or events evaluated in the CLB; therefore, the change does not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the CLB because no new failure modes are created. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0018)

FSAR Section 9.11: "Sampling System"

Technical information in Section 9.11 was revised.

Summary of Safety Evaluations: The change does not involve a physical change to the plant. The change merely makes the FSAR consistent with the plant configuration. The number of sample heat exchangers was changed from "5" to "5 per unit." The number of sample vessels was changed from "8" to "5 per unit." The word "approximate" was added to the sample vessel volumes. The radiation sample vessels and their approximate volumes were added to the list of sample vessels.

The basis for the change is CHAMPS and discussions with the system engineer. The design of the plant systems is not affected by the change. The change does not affect plant procedures, and no test or experiment is revised or created. The change does not involve new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0097)

FSAR Section 10.1: "Steam and Power Conversion"

The Unit 1 & 2 priming air ejector drain orifices appear in a different orientation on the drawing than they appear in the plant. Bechtel drawing M-201 Sheet 2 and M-2201 Sheet 2 do not have numbering on these orifices. Therefore, the drawings were changed to reflect actual plant configuration.

Summary of Safety Evaluation: The function of the priming air ejectors and their associated equipment is unaffected by the change. There is no change to the facility as described in the CLB. The changes have no effect on administrative controls or activities associated with the priming air ejectors or its instrumentation. The changes do not impose restrictions on system availability or administration of setpoint activities. The drawing change does not change procedures described in the CLB. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0028)

FSAR Section 10.1: "Steam and Power Conversion"

Unit 1 & common vacuum priming pump service water flow switch (FS-4305) bypass valves appear on Bechtel drawing M-207 Sheet 2 as gate valves when they are actually globe valves.

Summary of Safety Evaluation: The function of the SW system, the vacuum priming pumps and associated equipment are unaffected by the change. There is no change to the facility as described in the CLB. The drawing changes have no effect on administrative controls or activities associated with the SW system, the vacuum priming pumps, or its 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. Neither the structural integrity of the SW system nor the system capability is affected. The drawing change simply and accurately reflect the current material condition of the plant. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0029)

FSAR Section 10.1: "Steam and Power Conversion"

Unit 1 air removal system (AR) pressure regulator isolation valves CW-180 & 181, and 1CW-160 and 1CW-161 do not appear on Bechtel drawing M-212 Sheet 1, but are numbered in the plant.

Summary of Safety Evaluation: The function of the air removal system and its associated equipment is unaffected by the change. Therefore the addition of two normally open valves to the drawing does not constitute a change to the facility as described in the CLB. The changes have no effect on administrative controls or activities associated with the air removal system or its instrumentation. It does not impose restrictions on system availability or administration of set point activities. The drawing change does not change procedures described in the CLB.

The drawing change does not affect the operation of the air removal system or its instrumentation. The change more accurately reflects the current material condition of the plant. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0036)

FSAR Section 10.1: "Steam and Power Conversion"

This change provides clarification of wording in FSAR Section 10.1.

Summary of Safety Evaluation: The change does not involve a physical change to the plant. A sentence was revised that describes ice blockage of the intake crib during the wintertime. The word "prevent" is replaced by "reduce the likelihood of." The basis for this change is RCE 00-007, "Unit 1 Manual Trip Due to Decreasing Forebay Level," dated February 29, 2000, which showed that ice formation and blockage can occur in the 30" pipes and in the larger openings in the intake crib (CR 00-0213). The design of the plant systems is not affected by the proposed change. The change does not affect plant procedures. No test or experiment is revised or created. The change does not involve any new system interactions or connections, and system integrity is maintained. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0065)

FSAR Figure 11.1-2: "Blowdown Evaporator System"

The blowdown surge tank sample valve is shown on FSAR Figure 11.1-2 as a normally shut sample valve. The blowdown surge tank sample valve does not exist in the field and the sample line is capped. The drawing was changed to replace the valve with a cap. Another method of sampling the blowdown surge tank is available via the P-131B booster pump casing sample.

Summary of Safety Evaluation: The blowdown evaporator system and its components are not safety-related nor are they important to safety. No postulated accidents in the CLB apply to the change. No adverse affects are possible from this particular change that essentially is equivalent to the old configuration of a shut sample valve. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0034)

FSAR Figure 11.1-1: "Waste Liquid System"

The change revises the depiction of spent resin cask inlet isolation valve WL-1680 from open to closed on drawing 684J971 Sheet 1 (Figure 11.1-1 in the FSAR).

Summary of Safety Evaluation: This makes the drawing consistent with the configuration established the majority of the time by controlled, approved operating procedures, and reflects the normally expected alignment of the system. Maintaining the isolation valve shut while resin transfers or line flushes are not in progress is appropriate and minimizes the potential for inadvertent leakage of contaminated resins and water into a spent resin cask or onto the PAB floor. There are no negative impacts associated with the alignment, and the alignment has been demonstrated to be acceptable by several years of practical experience. This drawing change is consistent with the licensing basis and prudent operation of the facility, and does not result in a USQ or TS conflict. (SE 2000-0080).

FSAR Figure 11.2-1: "Gaseous Waste Management System"

DT-1041, K-1A&B waste gas compressor suction water drain trap is located in a very high radiation area that is not frequently entered. Since this is the case, a valve that serves as the trap's first-off isolation valve was not included on FSAR Figure 11.2-1 (Westinghouse drawing 684J972 Sheet 1.) To correct this, WG-01644, DT-1041 water drain trap first-off isolation is normally shut per plant procedures.

Summary of Safety Evaluation: The waste gas system and its components, including the drain train isolation valves, are not safety-related nor important to safety. Failures of these valves are bounded by analyses already contained in the CLB. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0047)

FSAR Section 11.5: "Radiation Monitoring System"

The change consists of revisions to technical information in FSAR Section 11.5.

Summary of Safety Evaluation: The changes are summarized as follows:

- Addition of "service water discharge" to the list of monitored discharge paths, for completeness.
- Correction and/or clarification to the name, type, detector range, indication, and control function for many radiation monitors in the tables to be consistent with plant configuration and the design and licensing basis documentation.
- Addition of RE-239, -240, and -243 to the list of area radiation monitors to be consistent with plant configuration.
- Addition of RE-241 and RE-242 to the list of process radiation monitors to be consistent with plant configuration.

This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0129)

FSAR Section 11.5: "Radiation Monitoring System"

The purpose of this safety evaluation is to make changes to FSAR Section 11.5 regarding operation of the radiation monitoring system, to agree with the configuration of the plant.

Summary of Safety Evaluation: The changes are summarized as follows:

- Delete the discussion of remote indication for low range are a monitors from Page 11.5.8.
- Delete meter indication from area monitor alarm unit background monitor channel RE-234B from Tables 11.5-2A and 11.5-B.
- Delete Technical Support Center background monitor channel RE-237B from Tables 11.5-2A and 11.5-2B.
- Delete reference to SPING Channel 2 (alpha particulate), 4 (iodine background) and 8 (low range gas background) from Table 11.5-3.

No changes in hardware or operation were made as a result of these changes. The changes do not affect the indications relied on to diagnose and respond to an accident or event. Deletion of background channels results in conservative radiation readings provided by the SPINGs and iodine monitors. Since the readings are more conservative, there is no adverse impact on the ability to respond to elevated radiation levels.

The radiation monitors associated with this change are not addressed in the TS. Since the radiation monitors will continue to provide their normal and post-accident monitoring functions, the changes do not pose a USQ nor require a change to the TS. (SE 2000-0064)

FSAR Section 14.1.6: "Reduction in Feedwater Enthalpy Incident"

S&L Calculation M-09334-229-FW.1, Revision 0, determined that opening the low pressure feedwater heater bypass valve would result in a feedwater temperature reduction of 31.1°F. FSAR Section 14.1.6, "Reduction in Feedwater Enthalpy Incident," was revised to reflect the new temperature reduction value (rounded up to the next integer value). The feedwater temperature reduction value because of a 10% step load increase is also added (rounded down to the next integer value), so that the results and acceptance criteria for this analysis are both stated in terms of temperature reduction in degrees F.

Summary of Safety Evaluation: 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 are no radiological consequences associated with either the reduction in feedwater enthalpy accident analysis or the excessive load increase incident (FSAR Section 14.1.7). 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. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0010)

FSAR Appendix A: "Seismic Analysis of SW Piping Using Pumphouse Response Spectra"

Seismic design of service water piping inside the pumphouse was originally evaluated using the static "g" method of analysis. Since pumphouse seismic response spectra have been developed, FSAR Appendix A.5, Section 5.5 was modified to allow the Response Spectrum Method of analysis as an alternative to the static "g" methodology.

Summary of Safety Evaluation: The original static "g" method of seismic analysis of service water piping inside the pumphouse was utilized in lieu of developing pumphouse seismic response spectra. The response spectrum methodology in accordance with FSAR Appendix A.5, Section A.5.8 would have been utilized had pumphouse seismic response spectra been developed to support the original design evaluation. Acceptance criteria when using the response spectrum method are the same as the static "g" method of design evaluation. Acceptance criteria when using the response spectrum method are the same as the static "g" method of seismic analysis. Since the static "g" methodology remains a valid methodology, no SSC is adversely impacted by this change. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0104)

MWRs 931624 and 931625: Blowdown Valves 1SW-00812 and 1SW-00813 Added to Service Water Strainers in Unit 1 Condensate Pump Motor Oil Cooler Supply Lines

Maintenance work requests 931624 and 931625 installed blowdown valves on service water strainers 1YS-02990 and 1YS-02994 for the Unit 1 condensate pump motor oil cooling supply lines. The Y-strainers are shown on Bechtel drawings M-207 Sheet 2 and M-202 Sheet 1 and FSAR Figures 9.6-5 and 10.1-2 Sheet 1. The addition of the blowdown valves constitutes a change to the facility as described in the FSAR. A 10 CFR 50.59 evaluation was not performed for the change at the time of installation. Therefore, this SE was prepared to document the acceptability of the existing plant configuration. QCR 99-0130 documents this error and completed corrective actions.

Summary of Safety Evaluation: The blowdown valve addition consisted of adding approximately 6" of 3/8" diameter carbon steel pipe with threaded ends connected to the strainers at one end and 3/8" brass blowdown valves at the other end. In addition, a threaded pipe nipple and pipe cap is installed on the outlet of each blowdown valve. The construction and pressure ratings of these components are acceptable and meet the requirements of the Bechtel JB pipe class. The components were installed in a non-QA, non-seismic line. Based upon a visual inspection of the involved weights and the locations of existing pipe supports, the added weight to the piping and strainers because of blowdown valve additions is considered acceptable.

The addition of the blowdown valves does not affect other service water components and does not change flow rates to service water loads. The service water system design functions as described in the CLB are not affected by the addition of the Y-strainer blowdown valves. No equipment important to safety is impacted by this change. The change does not pose a USQ nor does it require a change to the TS. (SE 2000-0030)

WO 9818976 and WO 9819877: Replace Condensate and Feedwater Chemical Addition Pots

The change replaced the chemical addition pots to the suctions of 1P-025A and 2P-025B condensate pumps.

Summary of Safety Evaluation: The pots and some of the immediate piping (up to 1&2CS-2, first-off isolation valves) will be replaced with Schedule 80 TP316 stainless steel. A drain valve will also be added to the bottom each pot, as well as a lockable lid. The stainless steel has higher tensile strength, internal pressure allowance, and resistance to the chemical additives then the previous carbon steel.

Foreign Material Exclusion improvements reduce the risk of contaminants entering the feedwater system. Therefore this activity will not increase the probability of occurrence of an accident or event previously evaluated in the CLB. Chemistry control is required to maintain secondary piping and components in good working order. The chemical addition pots still provide an injection point for chemicals to the condensate system.

This activity will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB. Radiological consequences due to a failure of the integrity of the piping for the chemical addition pot is not described in the CLB. Therefore, this activity will not create the possibility of an accident or event of a different type than any previously evaluated in the CLB.

This activity does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the CLB. The type of material and FME controls used for these pots are not described in the TS. Therefore this activity will not reduce the margin of safety defined in the basis for any Technical Specifications. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0058)

Work Orders 9931040 and 9914124 (IWP-98-046): Replace Unit 2 Rod Position Indication (RPI) Cards With New Analog Rod Position Indication (NARPI) module.

To ensure that the new rod position modules, being installed under MR 98-046 have sufficient range on the potentiometers to allow the module to be properly zeroed and spanned while installed in the RPI cabinets, a NARPI module was temporarily installed.

Summary of Safety Evaluation: The only way to verify that the modules can be properly spanned and zeroed is to install a module in the RPI cabinet while the control rod is fully withdrawn and while the control rod is fully inserted. This verification needs to be performed while the reactor is at rated temperature. Therefore, WO 9931040 was written to install a NARPI module in the U2 RPI cabinet with the unit at power to verify that the module can be properly spanned and WO 9914124 (IWP 98-046) was revised to install a single module with the unit at hot shutdown to verify that the module can be properly zeroed.

Installing the NARPI module in the RPI system only affected the position indication of one control rod. The installation of the NARPI module does not prevent the rod cluster control assembly (RCCA) from controlling reactivity automatically, nor does it prevent the RCCA from inserting into the core during a reactor scram. The NARPI module is designed to fit into the existing analog RPI card rack using the existing card edge connectors. Installation of the module only affects the position indication of the selected control rod since each control rod has its own analog conditioning card and rod bottom bistable. The power supply for the coil stacks of each control rod is fused. Therefore, upon the unlikely event that the installation of the NARPI module causes a short, the fuse for the control rod that the NARPI module is being installed in should prevent a total loss of the RPI system. In addition, per the FSAR the digital step counter serves as a backup to the analog position indication and a failure of the analog position indication system, in itself, does not lead the operator to take erroneous action in the operation of the reactor. Also, TS require specific actions to be taken upon loss of an analog RPI. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0091)

V. COMMITMENT CHANGE EVALUATIONS

WE Emergency Plan: The plan describes an extensive and reliable system for communications among the plant, near site EOF, JPIC, Alternate EOF, and state and local response organizations. A comprehensive communications system with backup capabilities has been designed to provide reliable communication links between the emergency response facilities and offsite support organizations. The system consists of the plant telephone system, the plant public address system, dedicated microwave links, portable radio systems, fixed radio communication systems, two digit dial select, and standard and dedicated offsite telephones. The primary means of initial notification is the two digit dial select. This is a statewide, dedicated telephone warning system linked directly with the Manitowoc and Kewaunee County Sheriff's Departments and the State Patrol. In addition, radio communications with transmit/receive capability are available between the control room and the Manitowoc County Sheriff Department. These communication systems are manned 24 hours per day.

Justification for Change: The National Warning System is no longer in use at PBNP. The NAWAS system has been replaced with the two-digit dial select system.

The NAWAS was a dedicated communications system controlled by the Federal Emergency Management Agency (FEMA) to transmit emergency information on a wide range of natural and technological hazards. The NAWAS telephone was located in the TSC. NAWAS was a four-wire system so that one can hear every transmission by every sender, and each receiver that has its speaker turned up hears each transmission. The NAWAS network had approximately 2300 direct contact points in the United States. The NAWAS telephone was used by first listening to the speaker to determine if the line was not in use. The handset was then picked up, the push-to-talk button depressed and the message delivered.

The two-digit dial select circuit is a unique, dedicated telephone network and is used as the primary means of notifying the state and counties of events at PBNP. The system allows for conference calling with any or all of the following locations: Manitowoc and Kewaunee County EOCs and Sheriff Dispatch centers, Wisconsin EOC and State Patrol in Madison, Kewaunee Nuclear Power Plant CR, IOF, and TSC, and PBNP TSC, EOF, AEOF and Control Room. To operate the two-digit dial select telephone, pick up the handset. There will not be a dial tone. Key in the applicable two-digit number. If a party fails to answer, a ringing tone will continue. The tone may be silenced by pressing the "#". Additional parties may be added at any time by depressing their number keys. If a conference call with all facilities is desired, dial 22. (CCE 2000-001)

Responsibilities Of the CAS/SAS Supervisor are Being Re-assigned to Another Position

A CAS/SAS Supervisor position was added to provide overall coordination and direction of CAS/SAS activities. The CAS/SAS supervisor also prepares and arranges for security support of plant activities currently handled by the CAS operator.

Justification for Change: The CAS/SAS supervisor position was created as a corrective action to Violation 1 of Inspection Reports 50-266/97007-02; 50-301/97007-02. The purpose of this action was to prevent recurrence of reportable uncompensated protected area alarm zone events. The full-time position provided overall coordination and direction of CAS/SAS activities. The CAS/SAS supervisor prepared and arranged for security support of plant activities previously handled by the CAS operator. Other corrective actions were implemented n response to the initiating event. The corrective action primarily credited with improved performance has been the change in compensatory measures procedures. Unrelated to the CAS/SAS supervisor position, compensatory measure implementation was simplified by reducing the variety of measures. Three compensatory measures procedures were reduced to one procedure. The effectiveness of implemented corrective actions for the initiating event is measured by the consistent and continued good performance of CAS/SAS operators. There has been no security event report required for an uncompensated protected area alarm for over three years.

Oversight of CAS/SAS operations during the past year has been shared with the contract operations captain because the committed position no longer requires full-time resources. The CAS/SAS supervisor has been assigned other duties. As such, all responsibilities of the CAS/SAS supervisor will now be reassigned to the contract operations captain. The CAS/SAS supervisor position will then be eliminated. (CCE 2000-002)

Wisconsin Electric Quality Assurance Policy Manual, Appendix B, Revision 8: The portions of the anticipated transient without scram mitigation system actuation circuit (AMSAC) that interface with existing safety-related portions of the plant are subject to the Nuclear Quality Assurance Program. The remainder of AMSAC is non-QA scope. The controls applied to the non-QA Scope AMSAC system meet the requirements of GL 85-06.

Justification for Change: The function of AMSAC is to start the auxiliary feedwater pumps and trip the turbine upon sensing that both main feedwater pumps have tripped or that both main feedwater regulating valves have shut. This system is diverse from the reactor trip system and is the result of the requirements of 10 CFR 50.62. The NRC did not require licensees to address the operability of this system in the plant TS, nor did the NRC require that this equipment be designated as safety-related. NRC GL 85-06 was issued to give quality assurance guidance for the control of ATWS systems that were not safety related. The guidance structured around the existing 10 CFR 50 Appendix B program. This guidance, however, did not require that utilities include ATWS systems under the 10 CFR 50 Appendix B program in order to met its intent. NRC Information Notice (IN) 92-06 was later issued to address reliability concerns of ATWS systems and other required equipment that was not being controlled by plant Technical Specifications. The IN expressed concern that utilities may not be giving adequate levels of attention to the operability of ATWS systems.

To comply with 10 CFR 50.62 and NRC GL 85-06, PBNP chose to control the AMSAC system under the Augmented Quality (AQ) program. A review of the AMSAC system was conducted and documented in Safety-Related and QA-Scope Reclassification Document (SQRD) 92-02 Action #1. (CCE 2000-003)

OM 4.2.2: Inservice Tests, Section 5.7

Applicable IST program documents were revised to require the performance of an operability determination and NRC notification prior to returning equipment to service in the event that equipment in the Section XI inservice testing program have vibration levels greater than 0.325 ips. This was a corrective action commitment.

Justification for Change: In the integrated inspection report covering the period March 2, 1996, through April 27, 1996, PBNP received a Notice of Violation which described four violations of NRC requirements. One of these cited violations involved the Section XI testing of the P-32E service water pump and dealt with the concern that the pump was returned to service with vibration levels above 0.325 ips when using ASME Section XI, Omb-1989, Part 6, Table 3 and 3a. As the pump was returned to service with vibration levels above 0.325 ips the NRC concluded the condition was in conflict with ASME Code requirements.

In response to the Notice of Violation (documented on VPNPD-96-042, dated July 19, 1996) PBNP stated that returning the service water pump to service did not violate ASME Code requirements. This was based on the position that "acceptable" as used in Paragraph IWP-3111 of ASME 1986 Edition, Section XI, is a general reference to equipment condition, not associated with specific vibration levels above 0.325 ips to be returned to service following maintenance provided the equipment testing frequency is increased and that a technical evaluation substantiates equipment operability. As the difference in opinion between the NRC and PBNP position hinged on the definition of the word "acceptable," PBNP submitted an inquiry to ASME. While the Code inquiry was moving forward, PBNP committed to revise applicable IST program documents to prevent equipment from being returned to service with vibrations in the alert range (above 0.325 ips). This commitment was later modified (NPL 97-0348, dated June 13, 1997) to revise applicable IST program documents to require the performance of an operability determination and NRC notification prior to returning equipment to service in the event that equipment in the Section XI inservice testing program have vibration levels greater than 0.325 ips.

The ASME response to the question, "Do Section 4.3 and 4.4 of OMb-1989, Part 6, prohibit the establishment of reference values above the fixed alert range of Table 3a?" was answered in ASME Inquiry Number OMI 96-01. ASME replied, "No, provided that acceptable operation is demonstrated and actions of Section 6.1 requires that equipment be tested on an increased testing frequency if vibration levels exceed 0.325 ips. Sections 4.3 and 4.4 pertain to establishment of reference values following maintenance. PBNP transmitted the ASME response to the Code Inquiry to the NRC on April 2, 1998.

Based on this, PBNP believes the original position that ASME Code does not prevent the return to service of equipment with vibration levels above 0.325 ips is correct. PBNP continues to recognize that the return to service of equipment under these conditions is contingent upon an evaluation that demonstrates that the equipment is operating in an acceptable manner and that the testing frequency is increased. However, as PBNP is required by TS 15.4.2.B to test ASME Class 1, 2 and 3 components in accordance with Section XI of the ASME Code, no additional commitment to follow ASME Code requirements is needed. Accordingly, the NRC commitment on this issue was retracted. (CCE 2000-004)

NRC Monthly Operating Reports

PBNP shall submit the report to the NRC by the fifteenth of each month, rather than the tenth. Also, the portion of the "Narrative Summary of Operating Experience" that describes the major safety-related maintenance, should be included in the NRC Annual Report and Data Results rather than the monthly report.

Justification: The change in report submittal from the tenth to the fifteenth of each month is in accordance with industry practice and the Standardized Technical Specifications.

The inclusion of major safety-related maintenance on a monthly basis is not necessary. This information is redundant to that reported in the Annual Results and Data Report. Major projects (e.g. steam generator replacement, addition of new emergency diesel generators) require additional correspondence and discussion with the NRC prior to work implementation and would be covered via that avenue. Other operational experience (e.g. shutdowns, significant power reduction, changes in nameplate ratings) will still be reported on a monthly basis. Initial monthly reporting was performed in accordance with NUREG-0020, "Licensed Operating Reactors - Status Summary Report", ("Gray Book"). However, the use of that guidance was discontinued in December 1995. GL 97-02 dated May 15, 1997, contains the current guidance for reporting monthly data to the NRC to assist the agency with the NRC Performance Indicator (PI) Program. Major operating experience will still be included in the monthly report; however, a listing of major safety-related "maintenance" is not needed to support GL 97-02 and NRC PI requirements.

These two changes can be implemented once the PBNP ITS submittal has been approved by the NRC for implementation. Immediate implementation can not occur because these specific requirements contained in the 1977 NRC letter are contained in TS 15.6.9.1.C, Monthly Operating Reports. ITS 5.6.3 states, "Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis..." No specifics of report content exist in our ITS submittal; therefore, the changes noted in this CCE will be upheld by the ITS. The CCE reflects GL 97-02 requirements. (CCE 2000-005)

NP 1.9.6 (old PBNP 3.4.12, and PBNP 3.4.20)

To verify the effectiveness of the requirements contained in PBNP 3.4.12 (NP 1.9.6), we will utilize our existing plant inspection program. Presently, group heads and supervisors perform regular and scheduled plant inspections in accordance with procedure PBNP 3.4.20 (NP 1.9.6) "Plant Cleanliness, Storage, and Inspection Program." Individuals who perform these inspections will be required to verify the proper seismic storage of equipment in accordance with PBNP 3.4.12 (NP 1.9.6). We will evaluate this procedure and make appropriate revisions to support this expectation.

PBNP 3.4.20 and PBNP 3.4.12 have been revised and renumbered to the following: NP 1.9.6, "Plant Cleanliness and Storage," contains requirements of proper storage including seismic storage, NP 13.3.1, "KNPP/PBNP Leadership Observation Program," contains requirements for plant inspections.

Justification: At the time of this commitment, PBNP 3.4.20 was the approved procedure for plant inspections which was later renumbered to NP 1.9.6. NP 1.9.6, Revision 5 "Plant Cleanliness, Storage, and Inspection" is presently approved and issued. On February 23, 2000, NP 13.3.1 Revision 0, "KNPP/PBNP Leadership Observation Program," was issued which now contains the requirements for plant inspections. Since this is the case, NP 1.9.6 is being revised to remove inspections from the procedure. Since this in not a simple administrative procedure renumbering, this commitment is affected. The proper storage requirements are contained in NP 1.9.6, but the regular and scheduled plant inspections have been moved to a new procedure NP 13.3.1. (CCE 2000-006)

IR 91-002 #4 and #5: (4) We expect the Safety Evaluation Group (SEG) to screen and prioritze open items and to assure that items of possible safety significance are being properly and promptly addressed. (5) The revision will establish a graded escalation process consistent with the significance (priority) of each respective open item. (i.e. delinquent top priority items will be escalated faster and to higher levels of management.) This commitment should be canceled.

Justification: This commitment stems from a 1989 Notice of Violation (IR 89-032 and IR 89-033) involving the poor timeliness for closure of self-identified items in internal PBNP open item lists. The original process governing the identification, review and resolution of issues (Corrective Action Requests and Non-Conformance Reports) were not well documented or controlled. This resulted in several conditions adverse to quality not being resolved in a timely manner when the threshold for reporting such discrepancies was lowered in 1988. The NRC cited that several outstanding open items were not properly prioritized and flagged for nuclear safety significance (cited violation of 10 CFR 50, Appendix B, Criterion XVI) for timely resolution.

Response to the referenced Inspection Reports resulted in the creation of the Safety Evaluation Group, whose charge was to review all 10 CFR 50.59 Safety Evaluations and to screen open items for safety significance (including the prioritization of open items).

Since making this regulatory commitment, the plant organization and the corrective action program have evolved and changed dramatically. The SEG organization no longer exists. The existing corrective action program is in keeping with industry standards (verified by 2000 Self-assessment versus the INPO developed industry standards document). (CCE 2000-007)

NP 5.4.1, Open Item Tracking: Point Beach Nuclear Plant shall maintain a process which governs the tracking and prioritization or regulatory commitments. This process shall also include a provision for bringing additional management attention to regulatory commitments.

Justification: The previous statement says that PBNP will have procedures for the tracking and prioritization of commitments and for bringing additional management attention to near term and overdue commitments or correction of deficiencies. This wording was vague with regard to the "correction of deficiencies" and was a gratuitous statement not related to corrective actions necessary to prevent recurrence. It was questionable whether this sentence is a commitment or a statement of fact regarding completed corrective actions.

A review of the specific NRC Notice of Violation (NOV) that resulted in this non-licensing basis "commitment" revealed that PBNP deviated from an Obligation contained in 10 CFR 50.48(d)(4) in that we did not show dedicated shutdown capability within 30 months of NRC approval (July 27, 1988). Contrary to this, we deviated from the obligation in that we had not implemented safe shutdown capability until May 21, 1991 which exceeded the 30 months permitted by the regulation by over four months.

The part of the statement in the reply to the NOV, "and correction of deficiencies" is gratuitous, and had nothing to do with corrective actions needed to ensure that we meet scheduler commitments to the NRC or that we do not deviate from NRC regulations containing scheduler details. This change is administrative, and therefore, does not impact the safety function of SSCs. (CCE 2000-008)

QP 16-6 (Cancelled), QP 16-4 (Cancelled), NP 5.4.1: On April 1, 1991 a new department wide procedure, QP 16-6, "Delinquent Open Item Notification System" took effect. This procedure simplifies the escalation process and clarifies applicability to all nuclear power department Priority 1 and 2 open items. This commitment should be canceled.

Justification: The existing commitment cites that a specific procedure and a very specific type of priority scheme will be in place to address the Notice of Violation from IR 91-02. This is a poor practice and a poorly contrived commitment. QP 16-6 was subsequently cancelled and incorporated into QP 16-4. QP 16-4 has been cancelled and replaced by NP 5.4.1. During this period of time the PBNP prioritization scheme has changed several times as well. Each of these process iterations has resulted in non-compliance with this commitment.

In our response to the NOV contained in IR 91-002, dated September 13, 1991, we stated that a commitment tracking system would be employed and that we had revised our procedures for tracking and prioritization of commitments, the process for bringing additional management attention to near-term and overdue commitments, and for the correction of deficiencies. The September 13, 1991, commitment encompasses the intent of the original commitment and should be the commitment of record for the resolution of the NOV from IR 91-002. The September 13, 1991, commitment is reflected in NP 5.4.1.

10 CFR 50, Appendix B, Criterion V requires that activities affecting quality to be prescribed by documented instructions or procedures commensurate importance of the activity. NP 5.4.1, "Open Item Tracking" addresses this requirement for the tracking of open items.
(CCE 2000-009)

ISI Long Term Plans, Third Interval, Units 1 and 2: PBNP is already committed to performing additional ISI examinations on reactor coolant bolting.

Justification: IE Bulletin 82-02 was issued to address and NRC concern on bolting being degraded due to boric acid attack or stress corrosion cracking. There had been numerous incidents throughout the industry where these types of degradation mechanisms had comprised the joint integrity, leading to leakage. Bulletin 82-02 was written to address this issue and to have utilities perform additional examinations beyond those required at the time.

PBNP now examines the bolting of every mechanical connection in systems borated for the purpose of controlling reactivity in accordance with Section XI, IWA-5200. This examination is required by Section XI, and is performed every outage on the reactor coolant system pressure boundary, and once each 40 month period for Class 2 systems. The examinations consist of VT-2 visual examination looking for evidence of leakage. If leakage is detected, then additional measures must be taken to assure the bolting has not been damaged. This can include removal of the affected bolting for a more detailed evaluation. This type of requirement targets those areas where problems are suspected, and ensures a continued high degree of confidence in the installed bolting. The areas examined every outage duplicate those required under IE Bulletin 82-02.

The current ISI requirements exceed those in the PBNP commitment to IE Bulletin 82-02. These examinations have proven effective in locating degraded fasteners before they can become a problem. Continuing the additional examinations per the commitment to IEB 82-02 does not add to the safety of the system. All other commitments for training will remain in place.
(CCE 2000-010)

MR 95-040: Portable shielding is no longer required for the north door to the control room due to the installation of the work control center which acts as permanent shielding that is more effective than the original portable shielding.

Justification: A commitment clarification is necessary due to the installation of MR 95-040 which erected the work control center on the north side of the control room. Subsequent analysis (Calc 95-150) showed that the WCC is a more effective shield for the control room door than the portable shielding. Therefore, the portable shielding was no longer necessary. (CCE 2000-014)

NUMBER OF PERSONNEL AND TOTAL DOSE BY WORK GROUP AND JOB FUNCTION - 2000

Work Group Station Employees	Number of Personnel Greater Than 100 mrem	Total rem for Job Group	Job Function and Total Dose, rem					
			Reactor Operations & Surveillance	Routine Maintenance	Inspections	Special Maintenance	Waste Processing	Refueling
Operations	44	15.364	7.876	-------	3.779	-------	1.224	2.485
Maintenance	68	26.089	-------	7.541	1.121	8.042	0.897	8.488
Chemistry & Radiation Protection	29	15.570	13.898	-------	-------	-------	1.672	-------
Instrumentation & Control	15	3.263	-------	2.598	------	-------	-------	0.665
Administrative groups & Engineering	13	5.362	0.955	-------	2.580	------	1.827	-------
Utility Employees	9	4.110	0.272	3.838	------	------	-------	-------
Contractor Workers & Others	201	69.231	7.722	0.062	2.719	58.557	0.171	-------
GRAND TOTALS	379	138.989	30.723	14.039	10.199	66.599	5.791	11.638

1104 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
TSC	Tubesheet Cold 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)
MBM	Manufacturing Burnish Mark
PVN	Permeability Variation
DEP	Deposit

Unit 1

No eddy current testing was performed on Unit 1 in 2000.

Unit 2

Inspection Plan:

Unit 2 steam generators were inspected in October of 2000. The following table shows the inspection plans performed and the number of tubes inspected per steam generator. There were previously 2 tubes plugged in "B" steam generator.

Inspection plan	"A" SG	"B" SG
100% Full Length Bobbin Coil	3499	3497
40% Tube End to Top of Tubesheet +2" in hot leg MRPC (pluspoint)	1411	1417
20% Row 1 U-bend MRPC (pluspoint)	11	11
Special Interest MRPC (pluspoint)	7	13

The following EPRI Appendix H qualified techniques were utilized for potential damage mechanisms:

EPRI ETSS#	Probe	Application
96001.1	Bobbin	Detection of thinning at TSP and Top of Tubesheet
96004.1	Bobbin	Detection of Wear at AVB Locations
96005.2	Bobbin	Detection of Pitting in the presence of copper
96008.1	Bobbin	Detection of IGA/ODSCC at non-dented eggcrate supports and/or sludge pile regions
96402.2	Pluspoint	Detection of axial and circumferential ODSCC at dented and non dented locations with and without structures and expansion transitions
96511.2	Pluspoint	Detection of axial and circumferential PWSCC in low row u-bend regions
99997.1	Pluspoint HF	Detection of axial and circumferential PWSCC in low row u-bend regions

Inspection Results:

There were no active damage mechanisms or service induced degradation detected in either steam generator. 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
MBM	78 (121)	66 (81)
DNT	8	12
PVN	2	5 (7)
DEP	1	1

All dents greater than 5 volts, permeability variation and deposit indications were inspected using a rotating coil probe (pluspoint). The result in all cases was NDF (no defect found).

The following table shows the location of each indication in the "A" steam generator.

Row	Column	Indication	Location	Inch Mark
12	31	DEP	05C	4.41
13	28	DNT	07H	2.69
13	30	DNT	07H	2.69
21	4	DNT	AV1	5.01
33	42	DNT	03C	42.38
33	88	DNT	AV6	1.06
55	70	DNT	AV6	3.01
70	81	DNT	AV6	20.92
72	73	DNT	AV6	14.38
1	56	MBM	TSH	10.79
1	56	MBM	TSH	14.88
1	56	MBM	TSH	21.73
1	56	MBM	TSH	24.32
1	56	MBM	TSH	25.76
1	56	MBM	TSH	29.76
1	56	MBM	TSH	31.2
1	56	MBM	TSH	32.8
1	56	MBM	TSH	35.17
1	56	MBM	TSH	36.58
1	56	MBM	TSH	38.49
1	56	MBM	TSH	40
1	56	MBM	TSH	42.79
1	102	MBM	01C	3.38
1	102	MBM	01C	6.06
1	102	MBM	01C	23.2
1	102	MBM	01C	24.36
1	102	MBM	01C	25.9
1	102	MBM	01C	27.15
1	102	MBM	01C	32.39
1	102	MBM	01C	34.72
1	102	MBM	01C	36.23
1	102	MBM	01C	39.75
1	102	MBM	FBC	15.61
1	102	MBM	FBC	17.15
1	102	MBM	FBC	18.7
2	5	MBM	FBH	12.64
2	5	MBM	FBH	14.75
2	5	MBM	FBH	15.55
2	5	MBM	FBH	18.32
3	76	MBM	01C	32.31
4	59	MBM	02H	4.36

Row	Column	Indication	Location	Inch Mark
4	103	MBM	01H	38.43
5	32	MBM	04C	24.54
6	51	MBM	TSC	12.86
6	103	MBM	06H	39.1
7	28	MBM	TSC	18.33
8	105	MBM	06H	39.14
10	5	MBM	06C	33.09
10	5	MBM	06C	35.25
11	44	MBM	06H	10.97
12	35	MBM	05H	6.69
14	59	MBM	07H	37.26
15	66	MBM	04C	18.74
15	86	MBM	02C	8.18
15	94	MBM	04H	21.98
16	93	MBM	06H	5.09
17	52	MBM	TSH	11.79
17	52	MBM	TSH	17.04
17	52	MBM	TSH	21.12
17	52	MBM	04H	41.2
17	52	MBM	06H	11.68
17	52	MBM	06C	28.62
17	52	MBM	06C	35.67
17	86	MBM	02H	41.03
18	51	MBM	TSH	22.32
18	51	MBM	TSH	38.92
18	87	MBM	04H	30.26
18	93	MBM	05C	12.27
18	103	MBM	06C	4.35
23	28	MBM	01C	6.81
24	77	MBM	05C	20.05
25	12	MBM	03C	18.29
25	68	MBM	TSH	8.79
26	79	MBM	05C	24.89
27	50	MBM	03C	6.43
28	21	MBM	06C	35.83
30	23	MBM	TSH	10.95
30	31	MBM	02H	8.06
31	14	MBM	06C	29.44
31	48	MBM	05H	23.91
31	48	MBM	05H	32.65
31	48	MBM	05H	35.78
31	48	MBM	05H	39.19
31	48	MBM	05H	42.18

Row	Column	Indication	Location	Inch Mark
31	92	MBM	TSC	6.18
31	96	MBM	TSC	5.82
34	53	MBM	04C	10.86
36	75	MBM	FBC	5.33
37	14	MBM	03H	3.5
37	14	MBM	03H	13.02
38	15	MBM	03H	35.52
38	31	MBM	01H	4.16
38	33	MBM	05C	22.86
38	35	MBM	AV1	9.6
38	37	MBM	AV1	11.15
39	22	MBM	AV2	4.47
39	22	MBM	AV6	11.69
39	28	MBM	02H	21.71
41	72	MBM	AV1	9.92
43	32	MBM	03H	20.4
43	80	MBM	FBC	3.01
44	31	MBM	FBC	6.19
44	79	MBM	06H	2.89
44	83	MBM	01H	40.81
44	83	MBM	05H	35.46
45	24	MBM	02H	36.7
47	76	MBM	05H	28.18
54	21	MBM	TSH	8.93
54	25	MBM	02C	15.66
54	29	MBM	04C	35.17
54	73	MBM	03C	13.42
55	92	MBM	05C	18.81
57	16	MBM	07H	16.44
57	16	MBM	07H	18.42
57	88	MBM	AV1	13.48
59	84	MBM	02H	6.27
61	30	MBM	02H	22.76
61	34	MBM	02C	22.93
63	34	MBM	04C	6.35
64	21	MBM	05C	28.35
64	79	MBM	01H	3.04
67	66	MBM	05H	21.9
69	42	MBM	AV2	2.28
74	31	MBM	01C	5.69
75	40	MBM	FBC	11.43
75	44	MBM	06C	17.84
75	72	MBM	02H	24.72

Row	Column	Indication	Location	Inch Mark
78	75	MBM	AV2	6.3
79	70	MBM	01C	42.96
80	63	MBM	02H	20.15
3	42	PVN	06C	21.39
8	105	PVN	06C	42.26

The following table shows the location of each indication in the "B" steam generator.

Row	Column	Indication	Location	Inch Mark
63	32	DEP	06H	32.26
7	30	DNT	07H	12.88
8	21	DNT	04C	37.92
13	66	DNT	07H	34.69
16	87	DNT	07H	6.23
18	5	DNT	TSH	6.16
18	21	DNT	AV6	17.31
35	62	DNT	AV5	3.69
41	22	DNT	05H	38.1
42	47	DNT	04H	19.07
64	67	DNT	TSH	3.73
71	30	DNT	AV5	0.74
83	66	DNT	03C	31.28
1	42	MBM	TSC	27.47
1	54	MBM	TSH	23.36
1	54	MBM	TSH	24.85
1	54	MBM	TSH	27.01
1	54	MBM	TSH	30.42
1	56	MBM	TSH	32.53
1	58	MBM	TSH	29.62
1	70	MBM	TSH	25.08
1	70	MBM	TSH	27.65
1	76	MBM	TSH	20.48
1	76	MBM	FBH	2.67
1	76	MBM	FBH	13.78
4	95	MBM	03C	40.33
5	4	MBM	06H	41.45
5	10	MBM	05H	41.27
9	2	MBM	05C	39.97
9	62	MBM	04H	9.1
9	90	MBM	02H	31.09
10	3	MBM	05C	31.2
10	39	MBM	05H	38.37
13	50	MBM	06C	2.46
14	53	MBM	04C	14.07
14	91	MBM	05H	36.39
15	104	MBM	04C	16.06
16	103	MBM	06H	36.93
18	13	MBM	06C	15.35
19	38	MBM	02C	42.19
21	34	MBM	05C	40.2

Row	Column	Indication	Location	Inch Mark
21	66	MBM	06C	3.04
22	49	MBM	04H	26.71
23	4	MBM	01H	3.89
23	4	MBM	01H	5.52
23	54	MBM	06C	9.11
26	27	MBM	02C	32.92
26	101	MBM	02C	39.63
27	48	MBM	04H	7.43
27	52	MBM	01C	32.44
28	37	MBM	05H	14.44
33	22	MBM	02H	22.43
33	36	MBM	06H	7.19
36	93	MBM	05C	2.7
38	23	MBM	AV2	10.97
41	64	MBM	01C	36.92
42	9	MBM	03C	8.29
42	9	MBM	03C	15.07
42	9	MBM	03C	17.45
42	9	MBM	03C	20.34
42	9	MBM	03C	27.25
42	9	MBM	03C	29.36
42	9	MBM	03C	30.81
42	13	MBM	04H	32.35
45	62	MBM	AV5	13.12
45	94	MBM	03H	23.36
45	96	MBM	03H	14.82
45	96	MBM	TSC	8.53
46	31	MBM	AV2	5.88
47	18	MBM	TSC	16.59
47	22	MBM	06H	4.13
47	92	MBM	FBH	8.95
53	50	MBM	TSC	14.27
53	50	MBM	TSC	16.07
53	68	MBM	06C	6.14
55	64	MBM	03C	34.65
55	66	MBM	TSH	2.23
57	36	MBM	TSH	21.5
57	78	MBM	04H	3.27
58	89	MBM	06C	41.89
59	58	MBM	07H	12.68
60	55	MBM	04C	10.43
60	59	MBM	TSC	9.76
61	36	MBM	03C	33.71

Row	Column	Indication	Location	Inch Mark
61	50	MBM	AV5	10.41
64	65	MBM	FBC	14.26
65	64	MBM	02C	8.09
73	80	MBM	FBC	9.4
76	65	MBM	06H	18.75
78	73	MBM	TSH	3.69
80	71	MBM	TSH	2.35
84	41	MBM	05H	36.31
84	51	MBM	03H	4.75
85	48	MBM	05H	42.25
7	16	PVN	TSH	12.48
7	20	PVN	FBH	16.47
11	24	PVN	06H	37.18
15	12	PVN	04H	5.94
31	48	PVN	TSH	-3.24
41	98	PVN	05H	2.4
41	98	PVN	05H	3.77

Repaired or Plugged Tubes:

No tubes were plugged in the "A" steam generator. Tube locations R31C48 and R73C58 were mechanically plugged in the "B" steam generator due to excessive tube noise. The noise that was present in these two tubes was present at the same level during the baseline examination. The tubes were conservatively plugged, no actual defect was found in either tube.

To date, 4 tubes have been plugged in "B" steam generator (0.11%).

Integrity Assessments:

A condition monitoring and operational assessment were performed in accordance with NEI 97-06. No corrective actions were necessary since there were no degradation mechanisms present.

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 2000.

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 2000.

IX. REACTOR COOLANT ACTIVITY ANALYSIS

There were no indications in 2000 where reactor coolant activity exceeded that allowed by Technical Specifications in either Unit 1 or Unit 2.