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Ladies/Gentlemen:

Dockets 50-266 and 50-301
Annual 10 CFR 50.59 Summary Report for 2001
Point Beach Nuclear Plant, Units 1 And 2

In accordance with the requirements of 10 CFR 50.59(d)(2), Nuclear Management Company, LLC (NMC), licensee for Point Beach Nuclear Plant (PBNP), is submitting the 2001 10 CFR 50.59 summary report. As required by 10 CFR 50.4(b)(1), the signed original is provided, and copies have been sent to the NRC Region III office and the NRC Resident Inspector.

The report contains descriptions of facility changes, tests and experiments, and commitment change evaluations that occurred during calendar year 2001. Note that these descriptions were previously provided in the Annual Results and Data Report for Point Beach Nuclear Plant that was required by the previous Technical Specifications. The annual report for the year 2000 was submitted on February 28, 2001. With the implementation of Improved Technical Specifications at Point Beach in 2001, this annual report is no longer required.

Please contact us if you have any questions.

Sincerely,

Tom Webb

Thomas J. Webb
Regulatory Affairs Manager

Enclosures

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager

IE47

ATTACHMENT 1

ANNUAL 10 CFR 50.59
SUMMARY REPORT FOR 2001

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10 CFR 50.59 AND 10 CFR 72.48 CHANGES

PROCEDURE CHANGES

AOP-10A, Safe Shutdown in Local Control, Revision 31 (Permanent)

The revision ensures that necessary guidance is provided to safely shut down the plant following a fire. The change applies to the control room, cable spreading room, vital switchgear room, and for less severe areas such as the computer and instrument room, and the El. 26' or 46' of the PAB.

Summary of Safety Evaluation: This evaluation only applies to hot shutdown equipment and components that may be affected by a fire, due to the equipment, its associated cables, its power supplies or support systems being located in the area.

As part of this evaluation, an extensive search of revisions made to AOP 10A was performed to determine the reason for the statement, "any other event requiring a control room evacuation." This statement first appeared in Revision 2 of this procedure, however, its purpose was not clearly defined. A search of FSAR Sections 9.8.14, Appendix A.1 and A.2 and other CLB documentation revealed known and analyzed conditions is addressed in other procedures. Consequently, the presence of this entry condition in AOP 10A does not aversely affect an appropriate response to an accident condition beyond Appendix R. Therefore, this statement does not introduce an unanalyzed condition. However, this statement is being retained to give Operations flexibility to apply this procedure should conditions warrant.

This revision will improve the effectiveness in maintaining the Appendix R performance goals during a postulated fire event. This will also ensure that the margin of safety inherent in maintaining the Appendix R requirements is maintained during a postulated fire event. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0032)

AOP-10B, Safe to Cold shutdown in Local Control, Revision 3 (Permanent)

The revision ensures necessary guidance is provided to respond to a specific fire area fire event and to rely on the credited equipment shown to be available (by the Safe Shutdown Analysis Report) in a given fire area. Although, fires can not be accurately predicted and their effects on equipment will vary, manual actions specified by the AOP for specific fires will maintain the Appendix R performance goals at times during the fire event. The revised procedure improves the effectiveness in maintaining the Appendix R performance goals at times during a postulated fire event. This also ensures that the margin of safety inherent in maintaining the Appendix R requirements is adhered to during the fire event.

Summary of Safety Evaluation: This evaluation only applies to cold shutdown equipment and components that may be affected by a fire, due to the equipment, its associated cables, its power supplies or support systems being located in the area.

According to the Appendix R analysis the power, supplies for the pressurizer heaters cannot be ensured therefore, a natural circulation cooldown method is applied. Calculation results were applied to the natural circulation cooldown curve being added to the procedure to demonstrate that the cooldown can be performed within the operating region without the use of heaters.

The revision improves the effectiveness of operations in maintaining the Appendix R performance goals during a postulated fire event. Ensures that the margin of safety inherent in maintaining the Appendix R requirements is maintained during a postulated fire event. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0033)

CAMP 110, Addition of Hydrogen Peroxide to RCS (Reactor Coolant System), Revision 16 (Permanent)

The revision oxygenates the RCS at cold shutdown prior to opening the primary system to facilitate enhanced corrosion product cleanup to reduce refueling cavity dose rates during fuel motion. It also provides a method for corrosion product source term reduction. This procedure is recommended by Westinghouse for PWR's and as an industry practice accepted by EPRI. This procedure is performed under conditions that are compatible with system equipment and utilizing existing plant equipment as designed. The procedure places the RCS in a condition normally encountered during a refueling operation. It does not require the plant to be in an unanalyzed configuration.

Summary of Safety Evaluation: Adding hydrogen peroxide to the RCS under the conditions described does not increase the probability of occurrence or consequence of an accident previously evaluated in the FSAR. Forced RCS oxygenation at cold shutdown conditions does not increase the probability of occurrence or consequence of a malfunction of equipment important to safety or reduce margin of safety a SSC provides. This activity does not create the possibility of an accident or possible equipment malfunction important to safety that was not previously evaluated. CAMP 110 performance does not reduce margin of safety defined or provided for in the PBNP TS. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0019)

FOP 1.2, Potential Fire Affected Safe Shutdown Components, Revision 0 (New Procedure)

The procedure provides guidance to determine the potential impact of a fire on the plant systems or components necessary to achieve and maintain hot shutdown. Potentially affected equipment and manual actions that may be required are identified for each fire area in the plant containing Appendix R safe shutdown equipment.

Summary of Safety Evaluation: The procedure ensures guidance is provided to understand the potential impact on equipment because of a fire event in the plant, and ensures a fire event has no adverse consequences to the plant and that a safe shutdown of the affected unit(s) can be accomplished.

The procedure will improve maintenance of Appendix R performance goals during a postulated fire event, and ensures the margin of safety inherent in maintaining the Appendix R requirements is maintained during a postulated fire event. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0047)

ORT 61, Sump “A” Drain to Auxiliary Building Sump, Unit 1 Revision 11, Unit 2 Revision 14, (Temporary)

The temporary change provides for the installation of a jumper in panel C01 to maintain containment sump "A" isolation valve WL-1723 open as described in the test. The jumper is removed prior to closing out ORT 61.

Summary of Safety Evaluation: WL-1723 is not the initiator of an evaluated accident. Incorrect operation of the valve would not create a new type of accident. Existing plant procedures reduces the probability of accidental actuation of equipment that controls the work inside the control boards. The use of a jumper emulates the operation of the normal valve control switch. The switch is spring return to the close position. The valve must be held open for leak rate testing and would inefficiently occupy the control board operator. Removing the jumper simulates releasing the normal control switch. An automatic containment isolation signal would close valve WL-1723. The jumper is placed in control room panel C01 and is easily accessible. The valve control return to the normal control switch is procedurally directed. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0026)

PBTP 101, Main Generator Voltage Regulator Checkout and Testing, Unit 1 Revision 0 (Temporary)

The procedure provides instructions for the checkout and testing of the Unit 1 main generator voltage regulator following voltage regulator replacement per modification MR 96-046*A. Testing consists of checks for proper connection and set up of the voltage regulator to the exciter and 19 kV system; establishing the proper dampening for stable operation of the closed loop voltage regulator; collection of step response data for system stability studies; and operational testing of limiter functions on the new voltage regulator.

Summary of Safety Evaluation: The function of the Unit 1 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 and has no safety functions.

Testing of the Unit 1 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 is performed as part of the modification installation to ensure proper connections and operation of voltage regulator components. This checkout includes calibration and testing of limiters and protective relaying located at the voltage regulator components. The on-line testing is conducted to minimize the possibility of a generator trip. Improper functioning of the voltage regulator under the test conditions is no different from 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 are maintained within the allowed band of $\pm$ 10% 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 does not have a long term effect on the reliability of the equipment. The emergency power sources are not affected by the testing and remain capable of supplying safety-related loads. Therefore, the probability of occurrence of a malfunction of equipment important to safety as previously evaluated in the CLB is not increased.

The testing of the main generator voltage regulator cannot directly challenge the fission product barriers (clad, RCS boundary, containment boundary). Since failures associated with this testing are not credited to mitigate accidents described in the FSAR, this testing does not affect the availability of 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 in the CLB are not increase.

The worse case scenario would be a near full power trip during the test of the Unit 1 Turbine-generator due to human error. The turbine-generator trip would result in a reactor trip. A plant trip is not considered a new accident and is within the design capabilities of the plant. Therefore, the testing of the main generator voltage regulator cannot create the possibility of a malfunction of equipment important to safety of a differently type than previously evaluated in the CLB.

The proposed testing of the main generator voltage regulator creates no new failure modes. The failure modes of the voltage regulator remain fail excitation, high fail excitation low, or erratic operation. 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 previously evaluated in the CLB. TS do not identify a margin of safety associated with the turbine-generator. Therefore, the margins of safety defined in TS have not been reduced. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0015)

PBTP 104, Emergency Diesel Generator G-04 Governor Replacement Retest, Revision 0 (Temporary)

The procedure is used to demonstrate that the ability of G-04 to respond as required during design basis accident conditions has not been degraded as a result of the replacement of it's EGB-13-P hydraulic governor actuator, booster pump and booster pump motor. This change is necessitated by the discovery of foreign material in the actuator oil. The procedure includes the following tests: manual start (including idle run), fast start, loaded run (including 5 minutes above 2951 kW), normal shutdown, paralleling G-04 to 2A-06, and a full load rejection. This evaluation will only address the full load rejection portion. This evaluation examines the safety significance of performing a full-load (>2951 kW) rejection to prove G-04's ability to control engine speed without tripping on over-speed. This particular test has been performed previously on numerous occasions in the domestic nuclear power industry as part of post-governor maintenance testing, to demonstrate operability of the affected emergency diesel generator (EDG) following governor maintenance/replacement. Regulatory Guide 1.9 Revision 3 and IEEE 387-1984 recommend this test.

Summary of Safety Evaluation: None of the affected equipment is credited with initiating a design basis accident. This test only affects G-04 and bus 2A-06. The loss of G-04 and/or 2A-06 has been evaluated in the CLB. The test is bounded by an existing evaluation, it does not increase the consequences of a scenario evaluated in the CLB. No margin of safety as defined in the TS Bases is affected by this test. This test has been successfully demonstrated on numerous occasions at PBNP at cold shutdown. This test is performed under bus loading and relay coordination similar to that of previous tests. This test was successfully demonstrated during the factory acceptance test of G-04. This test has been performed on many occasions by other domestic nuclear plants as part of post-governor maintenance testing. Furthermore, Regulatory Guide 1.9 Revision 3 and IEEE 387-1984 recommend this test. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0016)

PBTP 105, Test of Bearing Cooling Requirements for Electrically-Driven Auxiliary Feed Pump, Revision 0 (Temporary)

The auxiliary feedwater (AF) pumps were originally installed with service water (SW) connected to the inboard and outboard bearing housings and stuffing boxes. Based upon information from the AF pump vendor, it was determined that the AF pumps do not require cooling water flow to the stuffing boxes or the bearing housings since the temperature of the process fluid is relatively low for this type of pump design. This test was performed to support the vendor's conclusion. This test evaluates operation of the AF pumps without cooling water. This evaluation does not address physical changes to the plant. This test removes the AF pumps from the list of essential service water loads shown in FSAR Table 9.6-1.

Summary of Safety Evaluation: Based on the vendor information, the accident analysis and normal operating conditions of the AF system, the use of cooling water for the AF pump bearings and stuffing boxes is not required. This has been verified by the performance of this test and vendor evaluation of the test conditions and results. The pressure class of the piping has not been reduced and the piping will maintain its seismic qualification. The cooling water supply to the AF turbine bearings or the SW suction piping to the AF pumps is not degraded. The ability of AF to provide 200 gpm to a unit is not degraded. Based upon vendor evaluation of the pump application and test data, the AF pumps continue to perform their licensing basis functions. The test does not introduce new accidents or events since flooding of the room has already been evaluated and the physical configuration of the plant has not been altered. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0059)

PBTP 107, Emergency Diesel Generator G-04 Test, Revision 0 (Temporary)

The test is used to demonstrate the ability of G-04 to respond during a design basis accident has not been degraded because of the refurbishment. G-04 is tested using safety-related bus 2A-06 as well as an independent load bank before it is declared fully operable. G-04 is OOS during the testing and either appropriately isolated from 2A-06, or protective circuitry is functional for the bus and diesel.

The major tests include: manual start (including idle run), fast start, loaded runs, endurance and margin, hot restart and reliability, full load rejection and simulated single largest load pick up/rejection. Tests have been performed previously on the emergency diesel generators either during manufacturing or on site. During the simulated single largest load pick up/rejection portion of the test, the electric output from G-04 is isolated from 2A-06 and 1A-06 (breakers 2A52-93 and 1A52-86 are racked out) and a temporary load bank installed outside the G-03/G-04 building. The simulated single largest load pick-up test will require installing a temporary vacuum re-closer and cabling in the 2A-06 switchgear room. The cabling will run though the G-04 room and out the emergency fuel oil fill connection penetration in the G-04 room. The emergency fuel oil connection is out of service during some of the testing. The vacuum re-closer and cabling is positioned so as not to interfere with 2A-06, and a review for the fuel oil fill connection in the CLB confirmed that this connection is not credited in the CLB.

Summary of Safety Evaluation: G-04 was refurbished to industry standards under QA supervision and augmented quality AQ oversight. The generator was restored to its original specifications with some minor changes to allow for newer materials and enhanced reliability. The changes have been evaluated. Some factory post-refurbishment testing was completed including dielectric testing and motoring test runs with temperature checks. Before load testing on 2A-06, the generator's insulation is re-checked and unloaded runs are performed. These conditions and the 2A-06 and G-04 protective circuitry provide reasonable assurance that the 2A06 bus is not degraded by this test. In addition, testing only affects 2A-06 and G-04. The G-04 breaker in 1A-06 (1A52-86) is racked out during testing. The loss of 2A-06 bounds a postulated failure of G-04 or 2A-06. The loss of 2A-06 is evaluated in the CLB. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0054)

SLP-1, Items Lifted by Containment Polar Crane, Unit 1 Revision 10 (Permanent)

The revision incorporates a change in the safe load path for the reactor head ventilation ducts. The SLP currently shows a specific storage location for each reactor head ventilation duct. Some alternate storage locations are required to simplify storage when various maintenance activities are being performed during an outage. The SLPs are required to meet the requirements of NUREG-0612. The plant specific SLPs were submitted to the NRC in order to meet these requirements. NUREG -0612 requires that, " the path should follow, to the extent practical, structural floor members, beams, etc, such that if the load was dropped, the structure is more likely to withstand the impact." Because of congestion in the containment building, the load paths were developed to reduce carrying loads over the reactor vessel or safety-related equipment. The NRC reviewed the Point Beach safe load path submittal and stated, "due to the congestion of equipment inside containment, the Licensee reports that priority was given to developing load paths around safe shutdown equipment as opposed to over structural members. Assigning a higher priority to protection of safety-related equipment than to following structural members is in keeping with the intent of this guideline." Based upon the submittal and its review by the NRC, it can be seen that the paths were developed to avoid movement of loads over safety-related equipment, including the reactor vessel. The revised load path uses the same criteria as the original submittal because it avoids safety-related equipment and minimizes movement over the reactor vessel.

Summary of Safety Evaluation: The same criteria are used in the development of the revised safe load paths as was used in the original development. Therefore, the change does not increase the probability of the occurrence of a load drop. The change does not affect other accident or event described in the CLB. Therefore, the change does not result in an increase occurrence of an accident or event previously evaluated in the CLB. The revised safe load path does not cause loads to be moved over new equipment important to safety and does not cause loads to be moved over the reactor vessel. Therefore, the change does not increase the probability of an occurrence of a malfunction of equipment important to safety and does not increase the radiological consequences of an accident, event or malfunction of equipment important to safety previously evaluated in the CLB. No new accidents or events are introduced due to this change. The proposed change does not create the possibility of an accident or event of a different type than was previously evaluated in the CLB. Revisions to the safe load paths do not reduce the margin of safety as defined in the Basis. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0022)

MODIFICATIONS

The following modifications were implemented in 2001:

MR 94-056, Boric Acid Storage Tank Level Indication

The modification replaces the existing nitrogen bubbler level measuring instrumentation for the BAST T-6A, T-6B and T-6C with a more reliable and accurate system having an expanded operating range.

Summary of Safety Evaluation: The BAST has three channels of instrumentation. Since the BAST levels are no longer monitored for the operation of the safety injection (SI) system, they are no longer under equipment qualification (EQ) or Regulatory Guide 1.97 commitments. Since the BASTs are still used as an available flow path for reactivity control but are no longer required for RG 1.97, the modification reduces the number of channels of level indication from 3 to 2. The third channel is disconnected and removed.

No margin of safety is reduced since the new equipment operates in the same manner and adheres to the same design standards as the previously installed equipment. During the installation of the transmitters, the level indication is gathered from the tygon tubing. Each system is tested following installation by comparing the output of the transmitters with the tygon tubing. The comparisons are done in addition to performing normal calibration on the components. The channel redundancy is maintained. TS require that when a reactor is critical there must be two available flow paths from either refueling water storage tank (RWST) and/or BAST. Each tank is completed one at a time so a flow path is available from the BAST to provide sufficient boric acid to a reactor in case of an accident or required shutdown. This change does not pose a USQ nor does it require a change to the TS. (SER 97-037-01)

MR 95-071, Fire Protection System Upgrade in the G-01/G-02 Rooms

There are several sprinkler heads located in the G-01 and G-02 emergency diesel generator rooms (fire zones 308 and 309) that have their sprays blocked by miscellaneous obstructions. The obstructions prevent the spray pattern of the sprinklers from reaching the designed coverage area, which is in violation of NFPA 13 for the installation of sprinkler systems. This modification corrects this situation by modifying the sprinkler configuration in the G-01 and G-02 emergency diesel generator rooms.

Summary of Safety Evaluation: The new configuration conforms to NFPA 13-1999 requirements, and corrects the problem of nine obstructed sprinkler heads. Nine new sprinkler heads are installed at the obstructed locations with the existing sprinkler heads remaining and the second sprinkler head mounted lower. The sprinkler system is taken out of service to perform this modification, and a fire watch is performed hourly to ensure safety and fire protection.

The G-01 and G-02 diesel generator room sprinkler systems are not credited in the Appendix R scenario. Failure of the sprinkler system is not an initiator to a design or licensing basis accident or event. Taking the sprinkler system out of service to perform the modification cannot cause an accident or event. Therefore, the modification does not increase the probability of occurrence of an accident or event previously evaluated in the CLB. The modification increases reliability by installing new heads mounted lower that are not obstructed. The hydraulic analysis is not affected per NFPA 13-1999. New sprinkler heads are essentially identical to the existing sprinklers, and have the same actuation temperatures as the existing sprinkler heads. Components are selected with pressure rating that meet or exceed the design ratings of the fire protection system. The piping was analyzed and the new configuration meets seismic 2/1 requirements. Taking the G-01 and G-02 diesel generator room sprinkler systems out of service is allowed per the FPER, as long as hourly fire watches are performed. The system is tagged and drained prior to work to ensure that flooding does not occur, and that operating equipment does not get wet. The G-01 and G-02 emergency diesel generators remain fully operable during installation. The probability of occurrence of a malfunction of equipment important to safety, evaluated in the CLB is not increased. The modification does not increase the radiological consequences of an accident, event or malfunction of equipment important to safety evaluated in the CLB. The G-01 and G-02 diesel generator room sprinkler systems do not have margins of safety discussed in the CLB. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0036)

MR 97-014*G, Replace Panels D-11, D-16, and D-17, Install Connection Points in D-03 and D-04, Replace D-302 Interlock, Install Fuses for D-01 and D-02 Shunt Cables.

The modification replaced 125 VDC panel D-17, re-supplied Panel D-17 from panel D-26, and transferred common A-train loads to D-17, replaced 125 VDC panels D-11 and D-16, and transferred the Unit 1 A-train loads to Panel D-16, installed battery discharge test connection points in D-03 and D-04 to resolve a personnel safety issue, replaced interlock between switches D72-302 and D72-302-03 on D-302 to improve operation, and fused shunt cables in D-01 and D-02 to provide isolation in the event of a fault on these cables.

Temporary power is supplied to loads that must remain energized during the installation. During transfer of direct current (DC) control power supplies to non-safety-related switchgear, control power is transferred to the alternate supplies.

Summary of Safety Evaluation: The existing direct current panels are replaced with new panels containing fused switches. The fused switches improve direct current system selective coordination and increase fault-clearing capability. The replacement direct current panels are manufactured, tested and qualified for use in safety-related, seismic applications. The new panels are mounted in the same locations as the existing panels. Bus work, switches and fuses on the new panels are rated to supply the maximum expected loading during normal and accident conditions. The panels to which safety-related loads are transferred belong to the same direct current train and are supplied from the same main direct current bus (D-01) as the original panels. Safety train separation criteria are met for cables. Cables are adequately sized for ampacity and voltage drop.

This change does not affect heat load or ventilation requirements for the areas in which the panels are replaced, including the control room and cable spreading room. During panel board replacement and transfer of load to new supply panels loads that are not required to be operable, are de-energized. LCOs are entered for service water, auxiliary feedwater, emergency power, and control room emergency filtration in accordance with TS. During installation several loads supplied from the 125-volt direct current system (VDC) are de-energized. The affected loads by this modification are required primarily for accident mitigation. Since Unit 1 is in a refueling outage during installation of this modification, most accidents for which the affected loads are required to be operable are not possible. Direct current system malfunctions are not initiators of the accidents described in the FSAR accident analysis. No safety-related loads are being added removed or modified within the scope of this change. The new panels have ratings equivalent to those of the existing panels. Panel loading and voltage drop for the new configuration meet the requirements documented in the direct current system master calculations. Installation performed in accordance with requirements for safety-related operating equipment. Installation controls and post-maintenance testing ensure that the reliability of the new panels and cabling is not reduced.

Following installation of this modification, affected loads are restored to full operability. Rerouting of cables does not violate train separation criteria, including Appendix R requirements. Following installation, Appendix R timelines for achieving safe shutdown are improved due to improved selective coordination. Therefore, 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 new failure modes or reduce the independence of the main direct current buses, including interim conditions. Accidents or events associated with the loss of a single direct current train have been previously evaluated. The only failure modes associated with the new direct current panels and associated cabling are open or short circuits. These failure modes exist with the existing equipment and have been previously evaluated. Therefore, the proposed activity 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. Operability of affected systems and equipment is maintained in accordance with TS requirements. This modification does not affect the degree of independence of safety-related direct current system trains. This modification does not change control functions or setpoints. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0004-01)

MR 97-049*B, Replacement of Cable Spreading Room Ventilation System Chiller Unit HX-38A

The modification replaces the existing cable spreading room ventilation (VNCSR) system chiller unit, HX-038A, with a new chiller unit. The new chiller unit has a greater cooling capacity and provides greater operational flexibility when maintenance activities are performed on the chiller unit. This modification also contains various other equipment and piping changes required to support the new chiller unit and improve the overall operation and maintenance of the VNCSR system. These other changes include: SW system and VNCSR system piping reroutes and new piping supports to accommodate different nozzle locations on the new chiller unit; new chilled water circulating pumps and motors specifically sized for the new chiller unit; upgraded power supplies to the new chiller unit and the new chilled water circulating pumps; replacement of the automatic strainer in the chiller unit's SW inlet piping with a manual Y-type strainer; relocation of the SW regulating valves from the chiller unit's inlet piping to the chiller unit's outlet piping; addition of a throttle valve in the chilled water system supply piping; addition of SW regulating valve bypass valves; new flow indication in the chilled water loop; additional control switches for the chiller unit and chiller water circulating pumps for future "maintenance mode" operation; and reduction of the chilled water supply temperature to 34°F (nominal).

Summary of Safety Evaluation: The modification improved the reliability and maintainability of the VNCSR system. The new equipment and components do not introduce new failure modes or increase the consequences of equipment malfunctions, accidents or events. The design of the new SW system components maintains the integrity of the SW system during a design basis earthquake, and the design of the new VNCSR components ensures that they do not degrade the integrity of the adjacent SW system components during a design basis earthquake.

The VNCSR and SW systems will remain operable during installation of the modification. A temporary chiller unit is placed in service via TM 99-023 to provide cooling to the cable spreading room, and line stops and piping supports are installed in the SW and chilled water system piping to provide the necessary isolation between the modified piping and the operable portions of the systems. Neither a failure of the temporary chiller unit nor a failure of a temporary line stop results in a new malfunction or increases the consequences of existing equipment malfunctions, accidents or events. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0106)

MR 97-049*C, Replacement of Control and Computer Room Ventilation System Chiller Unit HX-38B

The modification replaces the existing control room/computer room ventilation (VNCR) system chiller unit, HX-038B, with a new chiller unit. The new chiller unit has a greater cooling capacity and provides greater operational flexibility when maintenance activities are performed. This modification also contains various other equipment and piping changes that are required to support the new chiller unit and improve the overall operation and maintenance of the VNCR system. The other changes include: SW system and VNCR system piping reroutes and new piping supports to accommodate different nozzle locations on the new chiller unit; new chilled water circulating pumps and motors specifically sized for the new chiller unit; upgraded power supplies to the new chiller unit and the new chilled water circulating pumps; replacement of the automatic strainer in the chiller unit's SW inlet piping with a manual Y-type strainer; relocation of the SW regulating valves from the chiller unit's inlet piping to the chiller unit's outlet piping; addition of SW regulating valve bypass valves; new flow indication in the chilled water loop; additional control switches for the chiller unit and chiller water circulating pumps for future "maintenance mode" operation; replacement of the support for computer room humidifier Z-078; reduction of the chilled water supply temperature of 34°F (nominal); and installation of chilled water supply and return cross ties to allow more maintenance flexibility.

Summary of Safety Evaluation: The modification improved the reliability and maintainability of the VNCR system. The new equipment and components do not introduce new failure modes or increase the consequences of equipment malfunction, accidents or events. The design of the new SW system components maintains the integrity of the SW system during a design basis earthquake, and the design of the new VNCR components ensures they do not degrade the integrity of the adjacent SW system components during a design basis earthquake.

The VNCR and SW system remain operable during installation of the modification. A temporary chiller unit is placed in service via TM 99-023 to provide cooling to the control and computer rooms. Line stops and piping supports are installed in the SW and chilled water system piping to provide the necessary isolation between the modified piping and the operable portions of the systems. Neither a failure of the temporary chiller unit nor a failure of a temporary line stop results in a new malfunction or increases the consequences of existing equipment malfunctions, accidents or events. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-121-01)

MR 97-049*D, Upgrade Control Room Envelope Boundary

The change consists of four separate packages that are directed toward upgrading the pressure boundary formed by the control and computer rooms. Three ventilation systems are affected, CR HVAC, the CSR HVAC, and the smoke and heat removal system. These systems share common ductwork communicating to outside the control room envelope. To prevent in-leakage of unfiltered outside air, during emergency modes of operations, these airflow paths are fitted with automatically operated isolation dampers. Five of these components are replaced with new bubble-tight dampers. New sections of duct are of leak tight construction and are supported to meet seismic 2/1 criteria. New manual balancing dampers are installed where needed to permit post-modification adjustments to system airflow. Existing duct penetration fire dampers are fitted with new fusible links, intended to enhance the performance of the smoke and heat ventilation system. The existing control room toilet exhaust fan is replaced with a new direct drive model and provided with a new downstream motor operated bubble-tight isolation damper. Since implementation of the modification requires taking the cable spreading room heating, ventilation and air conditioning system out of service, a temporary cooling system capable of meeting the heat load demands of the CSR, is provided. Temporary components are installed and powered in compliance with existing station procedures. Pipe and conduit penetration associated with the temporary system is made a permanent feature of the HELB and fire boundaries in which the new check valve and bypass manifolds to assure positive isolation between the main and standby compressor sources. During the short period of time required to complete this step, the CR-HVAC system is manually aligned to Mode 2 operation, and a voluntary LCO is entered.

Summary of Safety Evaluation: The configurations of the affected systems, during and following implementation of these design changes are bounded by the existing accident analysis, and limits of acceptability defined in the TS. The proposed activities do not affect the plant protective boundaries, do not cause release of radioactivity, and do not degrade the performance or margin of safety of SSC required for safe shutdown or to mitigate the effects of accidents. This modification cannot initiate an accident or transient that would challenge safeguards systems function or equipment. This modification does not introduce new failure or single failure modes bounded by the CLB. No changes are made to main control board annunciation. Completion of this design modification results in an improved control room environment consistent with the objective of establishing a high level of confidence that the CR HVAC system will function reliably and at a degree of efficiency equal to, or better then, that assumed in the accident analysis. There are no TS requirements for the CSR HVAC system. Failure of CSR ventilation is compensated for by the contingency actions listed in AOP 10A, Safe Shutdown in Local Control, Attachment E. Implementation of this modification does not place the plant outside the acceptance limits for operability of a SSC related to safety, as defined in the TS, or other document in the CLB. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0049-01)

MR 97-073*C and 97-073*D, SI Test Line Return Valves 1SI-897A & B Conversion to Fail Open

The modifications convert fail closed air operated valves (AOVs) 1SI-897A & B, safety injection recirculation line return, into fail open AOVs. The modifications decrease the amount of time required to establish containment sump recirculation and eliminate an operator casualty work around for the 1SI897A & B valves.

Summary of Safety Evaluation: There are no new changes that would cause failure of the AOVs or the SI system and their ability to perform their design functions. The design prevents the AOVs from having to be "gagged" open, thus allowing a decrease in the amount of time required to establish containment sump recirculation. During installation procedures are utilized to ensure proper control is maintained. These procedures ensure final installation is in accordance with the design. In addition, testing is performed to verify that components perform their design functions before they are placed in service. Equipment important to safety is currently QA for safety-related application. The same or equivalent QA safety-related equipment is already in use throughout the plant and is used to implement the modifications to maintain reliability of equipment and not change the probability of a malfunction of equipment. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0109)

MR 97-110*B, Unit 2 Containment and Façade Fire Detection Smoke Detector Replacement

The modification replaces smoke detectors in Unit 2 containment and Unit 2 façade. These are tied into the new fire detection system by installing panel XLH-8701 and rewiring local fire detection panels FACP-008, D-404, and C-911. None of these components has an effect on fire suppression systems, nor on the balance of the existing fire detection system. During the portion of time that D-400 and D-404 are de-energized, the applicable sections of OM 3.27, "Control of Fire Protection & Appendix R Safe Shutdown Equipment," are be entered and followed for plant fire rounds.

Summary of Safety Evaluation: The Unit 2 containment smoke detectors are replaced one-for-one using existing cable, conduit and fixtures. The Unit 2 façade smoke detectors are being replaced with linear heat detectors to provide better coverage and sensitivity and be less susceptible to environmental effects. Local panel signal rewiring at FACP-008, D-404, and C-911 presents easy access and is limited to the fire detection system. Work is procedurally controlled and includes provisions for pre-job briefs, FME controls, and full functional testing.

This activity is focused on the fire detection system, which is not required or assumed for a CLB accident scenario. New breaker loads are within the breaker limits and are similar to the loads they replace. Work is performed safety without impacting other SSC, and full functional testing assures proper installation. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0057)

MR 97-110*E, Unit 1 Containment, Façade and PAB Fire Detector Replacement

The modification replaces smoke detectors in Unit 1 containment, Unit 1 façade, and PAB El. 26' south. These are tied into the new fire detection system by rewiring local fire detection panels FACP-007, D-402, C-905, and C-906. A PPCS station at 1C-20 is replaced with a new fire alarm station. None of these components has an effect on fire suppression systems or on the balance of the existing fire detection system.

The Unit 1 containment and PAB El. 26' south smoke detectors are replaced one-for-one using existing cable, conduit, and fixtures. The Unit 1 façade smoke detectors are replaced with linear heat detectors to provide better coverage and sensitivity. Three of the PAB detectors cover an area containing safe shutdown cables (fire zone 184), so hourly fire rounds are performed per OM 3.27, Control of Fire Protection and Appendix R Safe Shutdown Equipment, while they are out of service to ensure protection. Local panel signal rewiring at FACP-007, D-402, C-905, and C-906 presents easy access and is limited to the fire detection system. PPCS station removal and fire alarm station installation at 1C-20 is limited to the PPCS and fire detection circuits. Surrounding circuits in 1C-20 remain unaffected and cannot cause or increase the probability of an accident or event. Work is procedurally controlled and includes provisions for pre-job briefs, FME controls, and full functional testing.

Summary of Safety Evaluation: This activity is focused on the fire detection system, which is not required or assumed for CLB accident scenario. The PPCS is also affected, but it remains functional throughout this activity. New breaker loads are within the breakers limits and are similar to the loads they replace. Work can be performed safely without impacting other SSC, and full functional testing and SQUG walkdowns assure proper installation. This change does not pose a USQ nor does it require a change to the TS. (SE 99-079)

MR 98-002*B, Safety Assessment System (SAS)/Plant Process Computer System (PPCS) Replacement

The modification consists of installation of new plant computer equipment in the computer room; installation of Cisco network components and two temporary workstations (consisting of one Sun Ultra CPU, one monitor, one keyboard and one mouse) in the control room; installation of cables to jumper the parallel computer points on terminal strips in the existing multiplexer cabinets to barrier terminal strips in new controller cabinets; installation of fiber optic cable between the computer room and the control room; and installation of fiber optic cable between the computer room and the technical support center (TSC). The new system is set-up to allow future connection of parallel computer point inputs through the I/O controller cabinets and cables are in place to allow the future connection of serial data (as controlled through the site acceptance test) from the PE-3205 computer in the multiplexer cabinets, radiation monitoring system, the ISFSI temperature monitoring system and the secondary performance monitoring system recorder to support a site acceptance test. The new computer components are not powered up during this modification.

Summary of Safety Evaluation: The plant process computer system is used to provide supplementary information to the operator and technical support personnel to assist in the normal operation of the nuclear steam supply system and to inform him of off-normal conditions. The plant design includes adequate instrumentation for the operator to operate the plant in a safe manner at all times, regardless of the availability of the computer system. The plant process computer and its associated display devices are not identified as an initiator of accidents identified in the CLB. The SSPDS functions of the multiplexers (for primary display of the core exit temperature data) are not affected by the installation of cabling between the multiplexers and the new I/O controller cabinets since the core exit temperature inputs are handled in a different portion of the multiplexer cabinets. To ensure against inadvertent termination of the wires in the multiplexer and I/O controller cabinets, the installation work plan includes check and concur steps for terminations. Parallel connection to the parallel computer point inputs in the existing multiplexer cabinets monitor the existing non-safety-related contact inputs that provide signal isolation between the PPCS computer and the associated instrument loop. The new PPCS is not connected to the PE-3205 computer in the multiplexer as part of this modification. As such, the data flow to the 'A' or 'B' PPCS computers and data flow between the multiplexers are not impacted by this modification and SAS/PPCS is 100% available. Power for the new computer server, controller cabinets and workstations is provided from non-vital buses to control the loading on vital buses. Loading on the buses has been evaluated to ensure that the new plant process computer components do not impact the vital instrument buses. Interim equipment locations have been evaluated for potential Seismic 2/1 concerns to ensure that non-safety-related components do not impact safety-related components. Temporary equipment is installed in locations separate from safety-related equipment. Tie downs are provided to prevent non-safety-related equipment from impacting safety-related equipment. The anchorage of the new computer cabinets was evaluated for seismic loading and the installation work plans included steps to ensure that a SQUG evaluation of new raceway, equipment and cabinets are completed. The parallel operation of new and existing computer components (once the new PPCS is powered up during the site acceptance test) will result in higher heat loads on the computer room/control room HVAC system. A calculation has been prepared to access the new heat loads versus the capacity of the computer room/control room HVAC systems. With the components energized the heat load remains below the capacity of the computer room/control room HVAC system. The number of core exit temperature measurement points is not affected by this modification. Provisions in AOP-21, PPCS Malfunction, define the methods for implementing TS monitoring and surveillance when the SAS/PPCS system is not available. No other TS related components are affected by this modification. The new PPCS is not connected to the PE-3205 computer in the multiplexer as part of this modification. As such, the data flow to the 'A' or 'B' PPCS computers and data flow between the multiplexers are not impacted by this modification and SAS/PPCS is 100% available. This change does not pose a USQ nor does it require a change to the TS. (SE 99-083)

MR 98-002*D and MR 98-002*E, Safety Assessment System (SAS)/Plant Process Computer System (PPCS) Replacement – Control Room Remodeling and Rewriting

Control Room revised the room layout (including adding a raised floor in designated areas) replaced control operator and duty operating supervisor consoles, moved existing SAS/PPCS terminals and video display monitors to the consoles, replaced the SAS alarm screen monitor and removed the existing SAS alarm screen pushbuttons, installed new plant computer equipment in the control room to allow parallel operating of the existing and new PPCS, and installed workstations (consisting of one Sun Ultra central processor unit (CPU), monitor (s), keyboard and mouse) in the control room. Modifications to the power cabling for the new PPCS components provided additional power for new components, and installed data cables for the new PPCS components.

The existing SAS and PPCS displays are on new consoles, new PPCS displays and computer components are installed on the consoles and the two systems are operating in parallel.

Summary of Safety Evaluation: The plant process computer system is used to provide supplementary information to the operator and technical support personnel to assist in the normal operation of the nuclear steam supply system and to inform him of off-normal conditions. The plant design includes adequate instrumentation for the operator to operate the plant in a safe manner at all times, regardless of the availability of the computer system. The plant process computer and its associated display devices are not identified as an initiator of accidents identified in the CLB. The anchorage of the new operator consoles and new raised floor has been evaluated for seismic loading and the installation work plans includes steps to ensure SQUG evaluation of new raceway and equipment. The existing SAS/PPCS components being relocated to the new control room consoles are non safety-related components that receive power from safety-related vital instrument buses. The new PPCS 2000 workstations are non safety-related devices. Modification MR 98-002*E provides power from vital instrument buses to receptacles on the new control room operator consoles. Circuit breakers are installed in the vital buses to protect the new circuits. Per calculation, circuit breakers sized 20A and 15A meet the coordination requirements for 120 volt vital instrument panels and associated inverters. This prevents a fault on a non safety-related load from causing a power interruption to safety-related devices. During completion of this modification and the final phase of the PPCS upgrade, only existing SAS/PPCS loads are powered from the new vital circuits. Most of the new PPCS 2000 loads are temporarily supplied from non-vital computer room instrument panels until modification MR 98-002*C is complete. The modification adds cables to existing raceway in the control room, computer room and cable spreading room and installs new raceway in the control room. New cable meets fire retardancy requirements and does not significantly add to the level of fire hazard in the computer room or control room. The new data cable operates at low voltage and generates little heat. The new 120 volt power cables carry low currents and generate little heat. The new consoles are constructed of steel and the console tops are a solid surface material. The new raised floor is constructed of cementous-filled steel floor tiles. New carpeting is also installed in the control room. The materials of construction have been selected for low flame spread and smoke development properties. The parallel operation of new and existing computer components will result in higher heat loads on the control room HVAC system.

A calculation assesses the new heat loads versus the capacity of the control room HVAC systems. With the components energized, the heat load remains below the capacity of the control room HVAC system. Interim conditions created by this modification result in loss of SAS/PPCS display at their normal locations. Redundant displays are available that can be used by the control operator. During removal of the old consoles and the installation of the new consoles, operator access to the main control boards may be restricted. The installation work plan includes provisions for additional operator staffing during these periods. The number of core exit temperature measurement points is not affected by this modification. Provisions in existing plant procedure AOP-21, PPCS Malfunction, define the methods for implementing TS monitoring and surveillance when the SAP/PPCS system is not available. No TS related components are affected by this modification. Redundant data displays are available within the control room. This change does not pose a USQ nor does it require a change to the TS.
(SE 2000-0056)

MR 98-002*M, Safety Assessment System (SAS)/Plant Process Computer System (PPCS) Replacement – Dedicated Core Exit Temperature Recorders

The modification installs dedicated core exit temperature recorders 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. Installation of the dedicated core exit temperature recorders allows 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, PPCS Malfunction. The dedicated recorders provide the qualified display location for the core exit temperature data.

Summary of Safety Evaluation: The incore thermocouples provide input to the SSPDS, reactor vessel level indication system and the sub-cooling monitor. These indications and displays provide information to the operator and technical support personnel to assist in the normal operation of the nuclear steam supply system and to identify off-normal conditions. The core exit thermocouples are used only for data acquisition and monitoring. None of the equipment affected by this modification performs control function. The SSPDS, reactor vessel level indication system and the sub-cooling monitor are not identified as an initiator of an accident identified in the CLB. Although, the qualified display location for the core exit temperature is shifted from the SSPDS to new recorders, the required core exit temperature indication and one channel of reactor vessel level indication and sub-cooling monitor remain available.

The new core exit temperature recorders are seismically qualified. The purchase requisitions for the new recorders require that software within the recorders be validated and verified as QA Level A software. An evaluation has been performed to demonstrate that the additional cabling for the new recorders does not have an adverse impact on the data feeds to the PPCS. The core exit temperature indication components are powered from vital buses.

The new design for the core exit temperature indication is simpler than the current SSPDS design. The modification adds cables in the computer room and control room. New cable is IEEE-383 approved and does not significantly add to the level of fire hazard in the computer room or control room. The new data cable operates at low voltages and the heat generated is negligible. Installation of the thermocouple cables requires erection of scaffolding in the control room. During the termination of the new cables for the core exit temperature recorders, signals from individual thermocouples to the SSPDS and PPCS are lost for a brief period. Work is sequenced in the installation work plan to ensure that indication for a minimum of two thermocouples per reactor quadrant is available at all times. The thermocouples jumpered to the SPEC200 cabinets are not routed to the new dedicated recorders. As such, the number of channels of sub-cooling margin monitor and reactor vessel level indication are not impacted by this modification. The mounting of the new recorders has been evaluated for seismic loading. The installation work plans include steps to ensure SQUG evaluation of the recorder mounting is completed.

The new recorders replace existing PPCS trend recorders in the 1/2C-020 panels. The existing recorders are powered from vital buses. The power requirements and heat generation of the new recorders is equivalent to that for the existing recorders. As such, there is no impact on the 120 volt vital power systems or control room HVAC system. The number of core exit temperature measurement points is not affected by this modification. With 8 thermocouples connected to each of the 2 recorders per unit, the dedicated recorders meet the TS requirements of a minimum of 2 thermocouples per reactor quadrant. No other TS related components are affected by this modification. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0037)

MR 98-002*O, Safety Assessment System (SAS)/Plant Process Computer system (PPCS) Replacement – Pre-Parallel Run Test on New PPCS

This modification consists of installation of data link software on the new PPCS system; testing the data links from the radiation monitoring system (RMS) CT1, the radiation monitoring system CT2 and the secondary performance monitoring system recorder (YR-04111) for compatibility with the new PPCS; one multiplexer at a time, connecting the new PPCS to the existing PPCS via spare serial ports on the PE-3205 computer in multiplexer cabinet; terminating wires for I/O inputs (sequence of events points; pulse accumulator points; and analog/digital points for the incore flux mapping system) at the I/O cabinets; performing logic tests for the I/O modules; performing functional tests for the I/O modules; and performing software tests. At the completion of this modification, the interconnections between the new system components and the existing PPCS system are installed and tested and the new system is operating in parallel with the existing PPCS.

Summary of Safety Evaluation: The plant process computer system (including data link with the RMSCT1 or RMSCT2) is used to provide supplementary information to the operator and technical support personnel to assist in the normal operation of the nuclear steam supply system and to inform them of off-normal conditions. The plant design includes adequate instrumentation for the operator to operate the plant in a safe manner at all times, regardless of the availability of the computer system. The plant process computer is not identified as an initiator of accidents identified in the CLB. The SSPDS functions of the multiplexers (for the primary display of core exit temperature data) is not affected by the termination of cabling in the new I/O controller cabinets or the testing of the significant operating event points since the core exit temperature inputs are handled in a different portion of the multiplexer cabinets. Disconnecting the RMSCT1 or RMSCT2 input to SAS/PPCS and connecting the new PPCS to RMSCT1 or RMSCT2 does not affect the control functions of the radiation monitoring system.

The secondary performance monitoring system recorder is not addressed in the CLB and does not affect plant operating or control functions. None of the equipment affected by this modification performs a control function. The new PPCS is connected to the PE-3205 computer in the multiplexers as part of the modification. Based on the PPCS System Engineer's observation of data traffic on the existing modems in the multiplexers (for data communication with the "A" computer, "B" PPCS computer, inter-multiplexer communications and, in 2 of the multiplexers, communication with the SSPDS display in the 1/2C-020 panels), addition of 2 new data ports are not degrade data traffic to the existing PPCS components. The IWP includes steps to monitor the performance of the old PPCS after the communication with the new PPCS has been initiated. If the System Engineer detects degradation of the existing SAS/PPCS functions during this initial connection, the procedure requires that the serial links to the new PPCS be disconnected. Therefore, the data flow to the "A" or "B" PPCS computers and data flow between the multiplexers are not impacted by this modification.

Work in the multiplexer cabinet to connect the fiber optic modems to the DC power supplies and to initiate communication to the new PPCS requires that the multiplexer be removed from service for a short period of time. AOP-21, PPCS Malfunction, provides guidance for monitoring TS requirements when the PPCS becomes unreliable or unavailable. The data signal from the PE-3205 computers in the multiplexers to the new PPCS is optically isolated. The data signal from the PPCS and RMSCT2 is electrically isolated so an electrical fault in the new PPCS cannot propagate to permanent plant equipment. During testing of the MRSCT1 and RMSCT2 data links, appropriate contingency actions, in accordance with RMSASRB 2.0, response guidelines for non-routine RMS situations, can be taken to maintain RMS data available to control room personnel. The temporary disconnection of data links exists for a short duration (less than one shift for each data link).

The number of core exit temperature measurement points is not affected by this modification. When a multiplexer cabinet is taken out of service to make the connection to the new PPCS, only half of the thermocouples are affected (since work is being done in one multiplexer at a time). The remaining half of the thermocouples would provide sufficient redundancy to satisfy TS requirements of two operating thermocouples per core quadrant. If the number falls below two per quadrant, AOP-21, PPCS Malfunction, provides guidance for monitoring TS requirements when the PPCS becomes unreliable or unavailable. No other TS related components are affected by this modification. De-energizing either or both system servers (RMS-CT1 and RMS-CT2) does not affect the operation of the individual process monitors that perform alarm/control functions in the RMS. None of the factors used to determine radiological consequences are changed by installation of the modification and the interim conditions created during the installation. No new radiological release paths or mechanisms are created. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0003-02)

MR 98-024*A, Modifications to the West Service Water Header Valves

The modification installs a manually-operated 14" gate valve SW-574 and associated drain valves SW-814 and SW-815 in the west service water header between the SW supply to the spent fuel pool heat exchangers and the supply to the Unit 2 containment fan coolers, and demolishes a section of the 6" service water line to the decon area connected to the 14" west service water header. This modification allows portions of the west header to be isolated for maintenance while maintaining the SW supply available to the spent fuel heat exchangers through either the north or west SW headers. During normal plant operation, the new isolation valve is locked fully open and performs no active safety function. Its only essential function is to passively maintain the pressure boundary integrity of SW system piping. This new isolation valve is completed without a refueling outage. An isolation boundary is established between line stops in the west header on either side of the proposed valve location, and the affected piping is drained. This interrupts ring header continuity for the SW system, resulting in entry into a 7-day LCO pursuant to TS 15.3.3.D.2.b for each operating unit. Prior to inserting the line stops, the spent fuel pool heat exchangers HX-13A and HX-13B are aligned to the north SW supply line so that spent fuel pool cooling is not interrupted. The new north SW supply line must be in place and operable. None of the supply lines to the containment fan coolers are isolated by these line stops. Testing consists of an inservice leak test, NDE of installed welds, and full stroke tests of valve SW-574 to verify proper operation.

The unused 6" line to the decon area is cut and capped approximately 24" from the connection to the 14" west service water header and again approximately 12" from pipe guide JB-2-H35. Once isolated, approximately 30' of 6" pipe including valves SW-227, SW-326, and SW-327 and 4 pipe hangers/supports associated with the line are demolished.

Summary of Safety Evaluation: The service water system is not a source of radioactive materials and is not the initiator of an analyzed accident or event described in the CLB other than plant flooding. Failures of the new valves and associated piping following installation and failure of a line stop during construction are bounded by failures previously identified and mitigated by design features of the service water system as described in FSAR 9.6, and therefore does not increase the probability or consequences of flooding.

Components are designed and installed to Seismic Class 1 requirements in accordance with codes, standards and guidelines established by the PBNP CLB. Line stop fittings and equipment are designed to meet or exceed B31.1 and original service water system design basis requirements, including pressure and temperature ratings. Line stop fittings are installed and tested per B31.1 and service water system requirements prior to breaching the existing service water pressure barrier. The line stops and cutting machines have a pressure rating well in excess of existing service water piping (100 psig at 100°F) and have been hydrostatically tested at the vendor's shop to at least 1.5 times the design pressure of the service water system. The temporary fittings and line stopping/hot tapping equipment are scoped as augmented quality (AQ).

The interim piping configurations have been included in the piping analysis, to ensure the stresses remain within the code. The new header isolation valve is locked open and is classified as QA safety-related and seismic class 1, consistent with the SW system design. Approved plant procedures and checklists ensure the SW system remains operable and unanalyzed conditions are not inadvertently entered controls operation and maintenance of the new valves. There is no increase in the radiological consequences of an event, and no new release paths for radioactive materials are created. This design change and process meets or exceeds the requirements of the installed system. This change does not pose a USQ nor does it require a change to the TS.
(SE 2000-0090)

MR 98-024*E, Component Cooling Water Heat Exchangers Service Water Alternate Return Line

The modification resolves the following design issues: The service water return header in its current configuration cannot be practically isolated for maintenance without either dual unit outages or substantial freeze seal, line stop, and/or hot tap installations. Providing a simplified and permanent means of isolation either the Unit 1 or the Unit 2 return header for maintenance is a key objective of the parent modification, MR 98-024.

At least three component cooling water (CCW) heat exchangers need to be available at all times unless one or both units are defueled. One CCW heat exchanger is needed for each operating unit, and two heat exchangers are needed for a unit being cooled down. For dual unit operation, a swing heat exchanger needs to be available to support cooldown. The current service water return header and outlet isolation valves for the CCW heat exchangers cannot be isolated for maintenance without temporary measures to keep at least 3 of 4 CCW heat exchangers operable.

Summary of Safety Evaluation: Potential accidents or events from the modification have been evaluated by PBNP CLB and/or supporting procedures. Service water and component cooling water are not identified as initiators of accidents or events evaluated in the CLB. Therefore, the probability of such events is not increased. The mechanical and hydraulic affects of the new equipment have been evaluated and do not result in increased probability of equipment malfunction. The new hardware meets the same quality requirements as existing equipment and analysis has demonstrated that design limits are not exceeded.

Radiological consequences of events and malfunctions are not increased since the ability of critical systems and components to perform their functions is not adversely impacted, and no unmonitored release paths are being created. Both outboard radiation monitors (1/2RE-229) is in service to ensure monitoring of releases while the main SW return header is split by line stops. In the event either 1/2RE-229 are not operable, the compensatory action required by TS is implemented.

The creation of a new accident, event or malfunction of equipment important to safety is not possible because failures of affected components are bounded by system design features or higher level failures previously considered. No TS or radiological effluent control manual (or their Basis) are affected by this change. Installation work is done in accordance with applicable Appendix R programs. No fire barriers are degraded as a result of the modification. Components are designed and installed to seismic requirements, in accordance with codes standards and guidelines established by the PBNP CLB. Interim piping configurations as well as the final configuration have been included in the seismic qualification analysis.

Because the service water and component cooling water systems are not sources of radioactive materials or initiators of accidents. The operation of equipment important to safety remains as described in the CLB. Analyzed accidents or events with radiological consequences as described in the CLB remains unchanged. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-012-01)

MR 98-024*J, Unit 1 Containment Fan Cooler and Fan Motor Cooler Replacement

The modification improves the material condition, reliability and maintainability of the Unit 1 "A" and "B" containment fan coolers (CFCs) and the Unit 1 "A" and "B" accident fan motor coolers (AFMCs). This is accomplished by replacing the "A" and "B" CFC cooling coils, replacing the "A" and "B" AFMC cooling coils, replacing the service water (SW) system supply/return piping and isolation valves to the "A" and "B" CFC cooling coils, and replacing portions of the SW system supply/return piping and isolation valves to the "A" and "B" AFMCs. This modification also upgrades the platforms that support the Unit 1 "B" and "D" CFC plenums.

Summary of Safety Evaluation: The capacities and functions of the CFCs, AFMCs and SW system are not adversely affected by this modification. The new CFC coils and AFMC coils have been specifically designed to satisfy the heat removal requirements under the design limiting conditions for heat exchanger fouling, tube plugging, and SW flow. The new CFC coil design does not result in SW boiling at the outlet of the CFCs. Post-modification testing verifies the SW flows to the new coils and the airflow in the VNCC system satisfies the requirements for design basis heat removal. The required SW flows to the replacement CFC coils and AFMC coils do not change from current values. Therefore, flow to all other SW supplied equipment is unaffected by this modification. The new design features of the CFC cooling coils and the AFMC coils (removable waterbox, corrosion resistant material) enhance the overall performance of the coils by reducing their potential for fouling and permitting their internal inspection and cleaning. 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 modifications to the "B" and "D" plenum support structures are required to resolve existing discrepancies between the support structures' as-built configuration and their design drawings. The support structure modifications restore the required design margins for the applicable loading conditions. Finally, since new components installed by this modification are passive, no new failure modes in the containment air recirculation cooling (VNCC) or SW systems have been created.

The SW system isolations results in the isolation of water to the fire hose reels inside containment. The affected hose reels are not required to protect Appendix R safe shutdown equipment. While the hose reels inside containment are isolated, compensatory actions for fire protection is established. The SW piping is not capable of performing its containment boundary function due to required breaching of the SW system pressure boundary. While the SW system pressure boundary is breached, the respective containment penetrations are considered to be in a degraded condition. This condition is tracked in accordance with CL 1E, Containment Closure Checklist, and containment closure for the affected penetrations is established by maintaining the containment isolation valves for the SW system in the closed position. Equipment movements in containment are performed in accordance with the safe load path (SLP) procedures. Demolition, installation and post-installation testing are performed to applicable procedures and standards. Visual inspections and pressure tests required by the PBNP ASME Section XI Repair and Replacement Program verifies the integrity of welds and component connections. CFC coils are transported into and out of containment during refueling operations through the personnel lock. To ensure the consequences of a fuel handling accident is not increased, coil movements are coordinated with the Core Loading Supervisor to ensure that core components are not being handled while the personnel lock third door is open. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0014)

MR 98-050*B, Access Control System Replacement – Modify Secondary Alarm Station

The modification expands the SAS structure to accommodate the new SAS console. This is accomplished by removing the existing bullet resistant panels and door that make up the east wall of SAS, adding new bullet resistant panels along the south wall to extend out approximately 8' to the east, adding new bullet resistant panels to form the east wall, and add a new bullet resistant door to the northeast corner. New bullet resistant ceiling panels are added over the expanded area. A new structural member is added at the seam between the old ceiling panels and the new panels for support. A new access control unit is mounted in the existing door control box to interface with the new security computer. The existing door exit pushbutton is relocated. A new card reader is mounted outside the new door for access control. The floor and ceiling tiles are removed and new tiles installed throughout the SAS. Additional convenience receptacles are installed. New tinting is added to the window panels to prevent anyone outside SAS from viewing the interior while still allowing the SAS operator to see outside. The lighting is replaced and new track lighting is installed. Work is performed under an IWP, written in accordance with approved plant procedures to ensure proper security controls are established prior to renovation.

Summary of Safety Evaluation: The SAS structure is analyzed under 10 CFR 50.54(p), this regulation governs changes to the Security Plan. Prior to performing work under this modification, a transition letter dated May 21, 1999 was submitted to the NRC. The letter describes the interim configuration of the SAS and the compensatory measures that is in force by the security personnel to ensure the security of the facility is not degraded during the renovation. Compensatory measures remain in effect until the integrity of the SAS is restored to the level described in the Security Plan.

The final configuration of SAS maintains the level of protection committed to in the Security Plan. Work is performed within the south gatehouse, the south gatehouse is a non-safety-related, non-seismic building that contains no vital equipment and is totally independent of plant operating systems. The change does not pose a USQ nor does it require a change in the TS. (SE 99-085)

MR 98-050*D, *F, *G, *H, Access Control System Replacement

The modifications installs field mounted access control equipment, the file server consoles and equipment, CAS and SAS main security consoles, connection of the doors, intrusion detection equipment and cameras onto the new system. Installation includes power to the console equipment and the connection of the communications cables to the field multiplexer chassis and router/repeaters. Conduits are routed between new and existing equipment. Security is notified so compensatory measures can be taken, or should doors be required to be held open. New circuit breakers are added in the security multiplexers or local circuit breaker boxes to provide independent protection for each power circuit. The new console equipment is connected onto the security access control computer and tested. Each door installation, camera, and intrusion detection zone are transferred onto the new security computer system and tested. Work performed on Appendix R fire doors, HELB barrier doors and flood barrier doors are done in accordance with the appropriate, approved plant procedures.

Summary of Safety Evaluation: The work on the security access control system is performed under 10 CFR 50.54(p). The regulation governs changes to the Security Plan that do not reduce security effectiveness. Transition letter (NPL 99-0299 dated May 21, 1999) has been filed with the NRC. The letter describes the interim configuration of the security system and the compensatory measures in force by security personnel to ensure the security of the facility is not degraded during the renovation. These compensatory measures remain in effect until the integrity of the security system can be restored to the level described in the Security Plan.

The final configuration of the security system maintains the level of protection committed to in the Security Plan. Work is performed under approved IWPs written in accordance with approved plant procedures that ensure work is independent of a plant operating systems or safety-related equipment and in accordance with the transition letter (NPL 99-0299 dated May 21, 1999). The change does not pose a USQ nor does it require a change in the TS. (SE 99-099)

MR 98-090, MR 98-124, MR 97-091, Demolition of Old Energy Information Center (EIC) and Construction of New EIC and Training Building

Changes involve demolition of the old Energy Information Center (EIC) and the building of a new EIC, which are outside the protected area. Utilities to the old EIC are abandoned and new EIC are not connected to the plant (except for the communication system). The new training building is connected to the plant by utilities under MR 97-091 (Construction of the NES Building). The licensing basis commitment affected by the changes involves moving the records storage area from the old EIC to the nuclear engineering services (NES) building. The records in the old EIC were protected by a halon fire suppression system; the records are now protected by a wet pipe sprinkler system in the NES building.

Summary of Safety Evaluation: Protection of auxiliary transformer, X-66, is provided by fuses installed in compartment 4 of switch-fuse unit H-08. Transformer X-66 feeds a new 480 V panel (B-65) that supplies power to the loads in the training building. The new transformer is located on a concrete pad near the southwest corner of the training building. The installation of transformer X-66 allows the 13.8 kV system to operate within its design ratings and capacities.

The loss of AC to the station auxiliaries, station blackout and Appendix R scenarios are bounding events/accidents that cover the installation of auxiliary transformer X-66. Installation of auxiliary transformer X-66 does not increase the probability or consequences associated with the CLB accidents/events. The proposed activity does not create or affect an accident or malfunction discussed in the licensing basis. Additionally, no structure, system or component important to safety is affected by these changes. The fire protection system for the records storage area outside of the plant has no impact on safe operating or shutting down the plant, nor is the records storage area located in the vicinity of equipment important to safety. The changes do not pose a USQ nor do they require a change in the TS. (EVAL 2001-002)

MR 99-077*A, Installation of a 13.8 kV Power Supply For the Sewage Treatment Plant Upgrade

The modification installs a new 13.8 kV transformer primary fused disconnect unit (H-09) and a 500 kVA 13.8 kV/480 V auxiliary transformer (X-72) to facilitate the loads required for MR 99-077, sewage treatment plant upgrade. The feeder cables for the transformer primary fused disconnect unit H-09, are spliced into the existing line from 13.8 kV bus H-02 breaker H52-23 to 13.8 kV switch-fuse unit H-08. H-09 is mounted on a pad outside the perimeter fence south of the potable water pumphouse.

Summary of Safety Evaluation: The new transformer primary fused disconnect unit, H-09, consists of a disconnect switch and fuses to allow isolation of new auxiliary transformer, X-72. Power to switch-fuse H-08 is connected to new transformer primary fused disconnect unit H-09 by lugging the cables routed from H52-23 and the cables routed to switch-fuse Unit H-08 onto the primary side of the H-09 disconnect. Auxiliary transformer, X-72 is isolated and protected by using the set of fuses on transformer disconnect H-09. H-09 is powered from breaker H52-23, power to H-08, on the 13.8 kV bus H-02. X-72 is rated for 500 kVA and steps down the 13.8 kV to a utilization voltage of 480 V. X-72 feeds a new 480 V panel in the sewage treatment plant that accommodates the loads for the new addition and the existing building. This new transformer is located on a concrete pad near the northeast corner of the sewage treatment plant.

The changes allow the 13.8 kV system to operate within its design ratings and capacities. The new cabling, and switch-fuse are tested and found acceptable prior to energization of the new transformer X-72. Isolation requirements are in accordance with NP 1.9.15, Danger Tagging Procedure.

The loss of AC to the station auxiliaries, station blackout, and Appendix R scenarios are bounding events/accidents that cover the changes and associated malfunctions proposed under this safety evaluation. The proposed change does not increase the probability or the consequences associated with these CLB accidents/events. Nor does the proposed change increase the probability of a malfunction of equipment important to safety. There are no radiological consequences associated with the proposed change. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0008)

MR 99-042, Unit 1 PORV Solenoid Cables

The modification installs a new cable for each Unit 1 pressurizer power operated relief valve (PORV) (1RC-430 and 1RC-431C) in dedicated conduits from the containment penetration to the control room. The purpose of this modification is to bring these valves into compliance with the requirements of Appendix R.

Summary of Safety Evaluation: The work on each PORV is broken up into two portions. Conduit installation and cable pulling, that does not affect the operability of either PORV, may be performed when Unit 1 is in normal power operation and Unit 2 is in any mode of operation. Final wiring connections and post maintenance testing is performed with Unit 1 in refueling shutdown with the reactor vessel head detensioned and bolts removed. Under this condition, the PORVs are not required for low temperature overpressure (LTOP), and removing them from service does not require entry into a LCO. Both valves (1RC-430 and 1RC-431C) are shut with power removed to perform the necessary wiring changes. The valves may be worked one valve at a time or in a parallel during this plant condition. Post-installation testing involves stroking the PORVs open and then shut to ensure proper operation. AOP-10A, Safe Shutdown-Local Control, is revised to delete step 3. Step 3 was added as a response to condition report CR 99-1832 to check for fire on the El. 26'of the PAB. The fire is detected, power to 1RC-430 and 2RC-430 is de-energized along with MCCs 1B42 and 2B42 to prevent the potential spurious opening of 1RC-430 and 2RC-430 and their associated stop valves (1RC-516 and 2RC-516). This modification prevents a hot smart short from a postulated fire in the PAB from spuriously opening 1RC-430 and 2RC-430.

The modification has been evaluated and the justification logic is as follows: There are no changes being made that would cause failure of the power-operated relief valves or the reactor coolant system and their ability to perform the design functions. The design prevents the PORVs from spuriously opening, thus allowing RCS integrity to be better maintained; during the installation process, applicable procedures are utilized to ensure proper control is maintained throughout the installation. These procedures should ensure that the final installation is in accordance with the design. In addition, testing is performed in accordance with approved procedures and with plant Operations input to verify that components perform their designed functions before they are placed back into service; equipment associated with this modification important to safety is currently QA for safety-related application. The same or equivalent QA safety-related equipment is already in use throughout the plant and is used to implement MR 99-042 to maintain reliability of equipment and not change the probability of a malfunction of equipment. This change does not pose a USQ nor does it require a change to the TS. (SE 2000-0081-01)

MR 99-046, 3-Hour Fire Rated Wall in the Unit 1 MCC Room (Fire Zone 156).

The modification installs a Seismic Class 1, 3-hour fire-rated wall west of column line J in fire zone 156(Unit 1 MCC room) of El. 8' PAB. The wall provides a 3-hour fire barrier between redundant trains of the Unit 1 charging pump cables that are routed through the room. Currently the relied upon separation method of cable tray fire stops and detention/suppression has been found to be inadequate. Condition report 99-2063 and LER 266/99-007-00 document this. The new wall provides full compliance with Appendix R for separation of redundant trains of equipment in this fire area.

Summary of Safety Evaluation: Installation of the fire rated wall does not interfere with the operation or maintenance of plant equipment. The new wall provides full compliance with Appendix R for separation of redundant trains of equipment in this fire area. With the new wall in place there is no longer a need for the exemption to Appendix R, Section III.G.2.b for the Unit 1 MCC room. The change does not pose a USQ nor does it require a change in the TS. (SE 2000-0107)

MR 99-069*A & MR 99-070*A, Reactor Fuel Upgrade

The modifications start 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) are loaded in the reactor cores as feed assemblies. The improved features permit higher burnups, future operation at an uprated power level of 1650 MWt, and a reduction in the number of feed assemblies. In addition, the 422V+ fuel has increased safety margins, improved structural integrity, improved fuel assembly components, and supports extended 18-month fuel cycles. The modifications are documenting this upgrade to 422V+ fuel and tracking necessary document updates as part of the fuel upgrade project.

Summary of Safety Evaluation: 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) are loaded in the reactor cores as feed assemblies. TSCR 210 was submitted and approved by the NRC per License Amendments 193 (Unit 1) and 198, (Unit 2). MR 99-069*A (Unit 2) and MR 99-070*A (Unit 1) are documenting this upgrade. This evaluation addresses required FSAR changes that were not specifically approved by the NRC, or require further evaluation by this safety evaluation. These items are: Increase in the low-low steam generator trip analytical setpoint in the LONF accident analysis; Increase RCS operating pressure to a nominal 2235 psig; Revisions to procedure that operate the RCS at a nominal 2235 psig, test the RCS at 2350 to 2390 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; Addition of each unit's steam generator design differential pressure data to the FSAR. This evaluation covers the fuel upgrade for both units, and the FSAR revisions apply prior to loading in the Unit 2 is in the interim before Unit 1 is upgraded, and after both units is upgraded.

These changes do not adversely affect equipment important to safety, or does it potentially initiate or create the possibility of an accident or event or increase the radiological consequences of an accident or event, nor does it constitute a reduction in a margin of safety for the following reasons. Components exposed to RCS pressure have retained the required design rating of 2485 psig. A review of components exposed to RCS pressure concludes they 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. 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 plant systems or equipment, and does not constitute a reduction in margin of safety. The required procedure revisions do not affect equipment important to safety or cause an accident or event. The additional steam generator design differential pressure data does not affect plant systems or components. The change does not pose a USQ. The NRC has approved the TS changes. (SE 2000-0070-01)

MR 00-002, 2P-2C Control Circuit

The modification installs a redundant 3 amp fuse in the 120 volt control circuit of the direct current controller for Unit 2 charging pump 2P-2C. This fuse is installed in parallel with the existing 3 Amp fuse and used when the remote/local transfer switch is placed in the LOCAL position. Make-before-break contacts from the remote/local transfer switch are placed in series with each fuse.

The purpose of this modification is to bring the control circuit of 2P-2C into compliance with the Appendix R requirement that hot shutdown equipment remain free from damage following a postulated fire event. Charging pump 2P-2C is an Appendix R hot shutdown component that is relied upon for Unit 2 after a fire in fire zones 142 and 187. However, the current control circuit is susceptible to a fault on the remote cables, that would blow the 3 amp fuse and de-energize the direct current power to the pump motor. The local transfer switch isolates the faulted cables, but the fuse is replaced before the pump is restarted. Per Appendix R requirements, replacing the fuse is not allowed for hot shutdown equipment. A redundant fuse, for use when the transfer switch is in the local position, will allow Operations to switch to local and restart the pump without replacing the fuse and satisfies the Appendix R requirement. This deficiency was documented in condition report CR 00-0022 and its subsequent operability determination. This modification corrects this deficiency.

Summary of Safety Evaluation: The work may be performed when Unit 1 and/or Unit 2 is in any mode of operation. The work is performed with Unit 2 in normal power operation. Installation work requires taking charging pump 2P-2C out of service while the control circuit is modified. Depending on plant conditions, the applicable TS LCO is entered. Appropriate fire watch per OM 3.27 is posted when 2P-2C is taken out of service. Post-installation testing involves running 2P-2C in remote, transferring to local control, stopping and starting the pump in local control, and finally transferring to remote control ensuring satisfactory results in all conditions. Since 2P-2C cannot be declared in operation during this testing, 2P-2A and 2P-2B must be operable. Running the third charging pump for post-installation testing does not pose impacts on charging system operations since 2P-2C is run at minimum speed. Provisions in the installation work plan have been made to adjust the speed of 2P-2A and 2P-2B as necessary to maintain charging flow and seal injection.

There are no new changes that would cause a failure of 2P-2C or the chemical and volume control system and their ability to perform their design functions. The function of the control circuit is not changed. The redundant fuse will ensure that 2P-2C is available for use in certain Appendix R scenarios to support maintaining the Appendix R performance goals. During the installation process, applicable procedures are utilized to ensure proper control is maintained throughout the installation. In addition, testing is performed in accordance with approved procedures and with plant Operations input to verify that components perform their designed functions before they are placed back into service. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0013)

MR 00-010, Nitrogen Supply to Units 1 and 2 X01 Transformers

The modification supplies nitrogen to the main transformer units (1-X01B, 1-X01C, 2-X01A, and 2-X01B) from the nitrogen supply tank (T-101) in lieu of the portable 12-pack of nitrogen tanks currently used. Tubing is run from the nitrogen tank, where the low-pressure header begins, to both units' transformers through the turbine building. The ½" OD tubing exceeds its design requirements of 35-75 psi and a flow of 2-3 cfh. The demand does not adversely affect the headers operation and ability to supply a cover gas to plant equipment. The plumbing of the nitrogen purge to the 1-X01B, 1-X01C, 2-X01A, and 2-X01B transformers reduces industrial safety issues, specifically operator risks from handling the 12-pack carts of nitrogen tanks. The transition of adding the transformer supply branches to the header is done with a minimal downtime for the header. The low-pressure header of the nitrogen gas system supplies components that can be without nitrogen for short duration of the tie-in. A complete isolation of component branches ensures water, air, and debris cannot enter the components and purge of the open parts of the header before returning it to service. After the branches are added to the header, the connection to the 12-packs at the transformers is transferred to the tubing ends of the new header branches. Check valves are installed to prevent back flow from the transformer gas space into the nitrogen header. This is consistent with the nitrogen system's design and prevents contaminants entering other nitrogen system components.

Summary of Safety Evaluation: The nitrogen gas system supplies a low-pressure header for non-safety-related equipment with the exception of the SI spray additive tanks. The additive tanks are supplied nitrogen as a cover gas in a non-vital function of tank operation. The remaining non-safety systems can operate safely without a nitrogen supply for a short period to make the tie-in.

The switch from 12-pack delivered nitrogen to header delivered for transformers purge does not create change in operation of the main transformers since the quality of nitrogen is the same or better than the 12-packs.

The proposed alteration does not increase the probability of occurrence of an accident or event previously evaluated in the CLB. The safety-related equipment supplied, which the nitrogen cover gas is a non-vital part of, allows for adequate down time for the work to be done to the header. Each branch of the header is isolated to prevent intrusion of air, water, and debris. Isolation reduces the amount of time and effort required for purging to return to a normal header. The existing regulators are bypassed by a new equivalent regulation system.

The use of nitrogen as a cover gas over the cooling oil is a long-term preventive measure. The scope of the down time for the cover gas involved with this modification is minimal and does not negatively affect the transformers performance. Check valves are installed to prevent back flow from the transformer gas space into to nitrogen header. This is consistent with the nitrogen system's design and prevents contaminants from entering other nitrogen system components.

The activity does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB. The activity does not affect the radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB since the nitrogen gas system is used in a non-vital operation of its loads. The activity, including interim configurations, does not affect TS since the nitrogen gas system is not involved directly with TS. The addition of the transformer nitrogen branches does not add an unmonitored radiological flow path that is inconsistent with system's current design, which includes relief and check valves. Other connections to the nitrogen header are equipped with check valves to prevent radiological back flow.

The addition of the transformer's purge to the nitrogen gas system does not adversely affect the under utilized nitrogen tank or the transformers. The interim configuration with the header down for a short period does not effect its supplied isolated systems for the expected header down time. The existing regulators are bypassed by a new equivalent regulation system. Since the nitrogen gas system is not directly involved with the TS, the change does not pose a USQ nor does it require a change in the TS. (SE 2000-0087)

MR 00-016, Appendix R Upgrade of El. 8' PAB Sprinkler System

This modification upgrades the El. 8' PAB sprinkler system to conform to NFPA 13-1999 requirements. It consists of the addition of a door spray nozzle, a sprinkler head and new piping, as well as the relocation of several other sprinkler heads and nozzles to provide improved system performance. The El. 8' PAB sprinkler system is taken out of service to perform this modification, and a fire watch is performed hourly.

Summary of Safety Evaluation: The El. 8' PAB sprinkler system is credited in the Appendix R scenario to provide fire protection to each of the three charging pump cubicles, the redundant trains of Unit 1 charging pump cables, valve control cables, primary system instrument cables, 480 V motor control center 1B-32 and remote instrumentation panel 1N-11. Failure of the El. 8' PAB sprinkler system is not an initiator to design or licensing basis accident or event. Taking the El. 8' PAB sprinkler system out of service is not an initiator to design or licensing basis accident or event. Taking the El. 8' PAB sprinkler system out of service cannot cause an accident or event. Therefore, the modification does not increase the probability of occurrence of an accident or event previously evaluated in the CLB, or of a different type than previously evaluated.

This modification does not result in a reduction of fire suppression capability, and increases reliability by upgrading the design density to allow the system to work in accordance with PBNP water demand. New sprinkler heads and nozzles installed are essentially identical to the existing heads and nozzles with the same actuation temperatures. Components are selected with pressure ratings that meet or exceed the design ratings of the fire protection system. The modified system retains its seismic qualification. Taking the El. 8' PAB sprinkler system out of service is allowed per the FPER, as long as hourly fire watches are performed. The hourly fire watch aids in the fire detection and suppression of El. 8' PAB area if a fire occurred. Therefore, the modification does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated in the CLB. The modification does not reduce the margin of safety in the Basis for TS. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0035)

MR 00-019 & MR 00-020, Units 1 and 2 Condenser Steam Dump Valve Positioner Controls Upgrade

The modification replaces the condenser steam dump valve positioners with an improved higher capacity double acting positioner. This allows removal of the volume booster and the existing current to pneumatic (I/P) converter for each valve. Improved feedback linkage is installed. The restrictive 3-way solenoid is replaced with a 2-way solenoid. The solenoid added to the closing air line provides for normal blow open operation. A supply air pressure regulator and new air filter are added to ensure clean supply air within the manufacture's recommendations for the equipment in the pneumatic control scheme and valve actuator.

Summary of Safety Evaluation: These upgrades to the condenser steam dump control scheme do not impact or decrease the ability of the steam dump valves to operate. The steam dump valves are not credited in an accident or radiological scenario per the FSAR or TS.

Condenser steam dump valve failure has been previously evaluated as an initiator for the excessive load increase accident. FSAR Chapter 14 events assume the condenser steam dump system not to operate during or after the transient since their operation would minimize the pressure transient or would not result in the most severe accident consequences. The modification of the condenser steam dump system does not affect the other systems as described in the Basis of the TS. An excessive load increase accident does not have radiological consequences. A failure of the condenser steam dump valves does not have a radiological consequence. The change does not pose a USQ nor does it require a change in the TS. (SE 2000-0069-02)

MR 00-032, Feedwater Sampling

The modification installs an independent corrosion product monitor (CPM) Panel (1C-198) with integral sample cooler (1HX-268) on Unit 1 to allow monitoring the amount of corrosion products being fed into the Unit 1 steam generators from the main feedwater system. The panel is being installed to meet the requirements EPRI PWR secondary water chemistry guidelines, Revision 5.

As part of this modification, the #5 feedwater heater oxygen sensor (1OE-4238C) is moved from its present location on the Unit 1 sample panel (1C-26) to the new CPM panel that is being installed closer to the #5 feedwater heater sample points than the Unit 1 sample panel. Moving the oxygen sensor closer to the sample point improves the accuracy of the indications derived from the oxygen sensor as compared to oxygen concentrations determined from grab samples. Movement of the oxygen sensor does not change chemistry action levels or alarm conditions as the output from the oxygen sensor is routed back to the original oxygen analyzer 1OA-4238C on Unit 1 sample panel.

Summary of Safety Evaluation: As part of the CPM panel installation, a sample chiller unit 1HX-377 is installed to supply cooling water to the sample cooler on the CPM panel. Power to the self-contained refrigerant type sample chiller unit 1HX-377 is from non-vital MCC 1B-41. The power-cable routing from non-vital MCC 1B-41 does not affect safety-related or safe shutdown equipment. Loading of the sample chiller on MCC 1B-41 has been determined to be bounded by the current loading calculations. The main feed regulating valve 1CS-00476 is in the immediate area of where the ¼" sample line to the CPM panel connects to the main feedwater piping, however the new sample line ties into the same point as the existing one and they are the same size, therefore the impact from a leak from the new sample line is no greater than that for the existing sample connections. Other than this connection point, no other safety-related or safe shutdown equipment is within this modification. The entire modification is being installed outside QA boundaries.

Installation and operation of the CPM panel allows a direct, more timely and accurate tracking of the corrosion products being introduced into the steam generators by the main feedwater system. Also, relocating the feedwater oxygen sensor closer to the actual sample point results in a more accurate indication from the sensor. Providing more timely and accurate monitoring of the parameters conducive to corrosion results in quicker implementation of necessary changes in secondary water chemistry to minimize corrosion product buildup in the steam generators. By minimizing the corrosion product buildup in the steam generators, the probability of deterioration of the steam generator tubes should result in a decrease in the probability of a steam generator tube rupture. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0045)

MR 00-033, Final Feedwater Sampling

The modification installs an independent corrosion product monitor (CPM) panel (2C-198) with integral sample cooler (2HX-268) on Unit 2 to allow monitoring the amount of corrosion products being fed into the Unit 2 steam generators from the main feedwater system. The panel is being installed to meet the requirements EPRI PWR secondary water chemistry guidelines, Revision 5.

As part of this modification, the #5 feedwater heater oxygen sensor (ZOE-4238C) is moved from its present location on the Unit 2 sample panel (2C-26) to the new CPM panel that is being installed closer to the #5 feedwater heater sample points than the Unit 2 sample panel. Moving the oxygen sensor closer to the sample point improves the accuracy of the indications derived from the oxygen sensor as compared to oxygen concentrations determined from grab samples. Movement of the oxygen sensor does not change in chemistry action levels or alarm conditions as the output from the oxygen sensor is routed back to the original oxygen analyzer 2OA-4238C on Unit 2 sample panel.

Summary of Safety Evaluation: As part of the CPM Panel installation, a sample chiller Unit 2HX-377 is installed to supply cooling water to the sample cooler on the CPM Panel. Power to the self-contained refrigerant type sample chiller Unit 2HX-377 is from non-vital MCC 2B-41. The power-cable routing from non-vital MCC 2B-41 does not affect safety-related or safe shutdown equipment. Loading of the sample chiller on MCC 2B-41 has been bounded by the current loading calculations. The main feed regulating valve 2CS-00476 is in the immediate area of where the ¼" sample line to the CPM Panel connects to the main feedwater piping, however the new sample line ties into the same point as the existing one and they are the same size, therefore the impact from a leak from the new sample line is no greater than that for the existing sample connections. Other than this connection point, no other safety-related or safe shutdown equipment is within this modification. The modification is outside QA boundaries.

Installation and operation of the CPM panel allows a direct, more timely and accurate tracking of the corrosion products being introduced into the steam generators by the main feedwater system. Relocating the feedwater oxygen sensor closer to the actual sample point will result in a more accurate indication from the sensor. Providing more timely and accurate monitoring of the parameters conducive to corrosion results in quicker implementation of necessary changes in secondary water chemistry to minimize corrosion product buildup in the steam generators. By minimizing the corrosion product buildup in the steam generators, the probability of deterioration of the steam generator tubes results in a decrease in the probability of a steam generator tube rupture. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0046)

MR 00-048, Service Water Blowdown Valves

Blowdown ball valves are added to the Y-strainers on Unit 2 and the existing ones upgraded to ball valves on the Unit 1 Y-strainers. These are on the service water system for condensate pump motor oil cooler (HX-84A/B) supply lines. The components replaced are non-QA and non-safety-related. The addition of the blowdown valves do not affect other service water components and does not change the flow rates to the safety-related service water loads. The installation of the blowdown valves is during the Y-strainers disassembly and cleaning for PC-43 Part 3.

Summary of Safety Evaluation: Service water components are not specifically identified as an accident or event initiator in the CLB. The upgrade/addition of the blowdown valves to the Y-strainers do not impact the ability of the service water system to perform its accident mitigating functions as described in the CLB. The alteration does not degrade the ability of the service water system to provide cooling water in support of continued condensate pump operation. The change does not increase the probability of the loss of normal feedwater event described in FSAR Section 14.1.10. Flow to service water system components is unchanged. The change does not add new types of equipment or introduce unusual plant operating configurations. The additional/upgraded valves are attached to a non-safety-related section of the service water system and do not create the possibility of an accident or event. This portion of the service water system is not located in a safe shutdown area. The change does not degrade the service water pressure boundary, the ability to provide cooling to the condensate pump motors, and does not affect equipment important to safety. Installation of the valves is done during routine disassembly and cleaning of the SW strainers.

The change reduces the probability of overheating/loss of condensate pumps by allowing the Y-strainers to be cleaned without isolation of service water flow to the oil coolers. The change does not pose a USQ nor does it require a change in the TS. (SE 2000-0086)

MR 00-061, Removal of Pressure Indicating Switch, PISL-LW-037

The modification documents a previously undocumented modification identified by CR 99-0860. PISL-LW-037, Suction Pressure Switch for the Blowdown Evaporator Bottoms Pump (P-134), was removed and replaced with a pressure gauge. The FSAR shows this switch on Figure 11.1-2. The evaporator bottoms pump is manually controlled and is infrequently used.

Summary of Safety Evaluation: There is no nuclear safety significance associated with this switch. The switch is not associated with safety-related or equipment important to safety. The bottoms loop is not discussed in TS. The bottoms pump, P-134, is manually controlled as directed by procedure and infrequently used to transfer blowdown evaporator waste to a high integrity container for disposal off-site. The change does not pose a USQ nor does it require a change in the TS. (SE 2000-0126)

MR 00-063, New 900 MHz Radio System

The modification replaces the existing 5 channel conventional Motorola VHF radio system. This system is replaced with a new five channel, trunking 900 MHz Motorola radio system. This includes the removal/replacement of antennas, transmitters, receivers, main dispatch consoles and hand-held radio units, with the exception of System E (OPS 3). System E provides the necessary ten-mile offsite radio coverage for radiation protection personnel during an actual emergency or drill. This system remains in service until the high gain antenna, associated with the new 900 MHz radio system is installed. The new 900 MHz radio system is available to in-plant and near-plant areas. The scope of this work is non-QA and non-safety-related.

Summary of Safety Evaluation: The new radio equipment is staged and connected as much as possible prior to removing of the existing system from service. The new radio system is placed in service in parallel with the existing radio system, one antenna at a time. As existing antennas are replaced with new antennas, a system test is performed to verify functionality and proper coverage. After all of the antennas have been replaced (except the antennas in Unit 1 and Unit 2 containments), the old system is removed from service. Additional testing is performed prior to placing the new system in service and after completion of the installation. This additional testing checks energy levels in the plant, particularly in the control room. During the interim configuration of cutting over to the new system, primary radio system users use both radio systems. In addition, during at-power entries into containment, the Gai-tronics system must be used for communication between personnel inside and outside of containment. Point-to-point communication between personnel within containment is possible.

New antennas are being installed in the same locations as the existing antennas. The existing antenna locations have been evaluated for their potential interaction with RFI sensitive equipment. Field walkdowns verify that no currently identified RFI sensitive equipment is in the immediate area of the existing antenna locations. The output power of the new system is less than the existing system; therefore, there is reasonable assurance the new 900 MHz system does not impact non-safety or safety-related RFI sensitive equipment.

The portable radios operate at less power and therefore should not increase plant energy levels above equipment susceptibility levels. The operation of the new portable radios is the same as the existing portable radios they transmit only when the push-to-talk (PTT) switch is depressed. The new portable radios contain a "transmit inhibit" switch blocks transmission when enabled. The new radio system is primarily powered from non-safety-related power supplies, while the remainder is powered from emergency diesel generator (EDG) backed power supplies. Supplying a portion of the new system from EDG backed power supplies allows PBNP to continue to meet commitments in the PBNP Security Plan. The additional load from this equipment does not change the calculated EDG injection or recirculation loads for emergency lighting transformers or the security battery chargers shown in Tables 8.8-1 and 8.8-2 of the PBNP FSAR. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0041-01)

MR 00-102, Service Water Upgrades to Emergency Diesel Generator G01

The modification adds a flow orifice and a duplex strainer in the jacket water cooler inlet piping; eliminates the SW flow control valve and raw water low pressure switch; and increases the size of piping from 4" to 6".

Summary of Safety Evaluation: Interim operating configurations are not applicable since the EDG is out of service for the intrusive portion of the modification. Interim seismic calculations are performed since the G01 and G02 seismic analysis are inclusive of each other and G02 must remain operable. No new or unique equipment or components are being added. Standard piping materials compatible with existing piping are utilized. No other operating parameters of the diesel generators are affected; consequently the reliability and availability of the EDG is increased. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0028)

MR 00-103, Service Water Upgrades to Emergency Diesel Generator G02

The modification adds a flow orifice and a duplex strainer in the jacket water coolers inlet piping; eliminates the SW flow control valve and raw water low pressure switch; and increases the size of piping from 4" to 6".

Summary of Safety Evaluation: Interim operating configurations are not applicable since the EDG is out of service for the intrusive portion of the modification installation only. Interim seismic calculations are performed since the G01 and G02 seismic analysis are inclusive of each other and G01 must remain operable. No new or unique equipment or components are being added. Standard piping materials compatible with existing piping are utilized. No other operating parameters of the EDG are affected; consequently the reliability and availability of the EDG is increased. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0027)

MR 01-001, Intake Crib Reconfiguration

This modification removes the structure the Cormorants have been roosting on to fish. Killing Migratory birds is a violation of the Migratory Bird Treaty Act Violation Title 16 - Conservation ss. 703, 707. The intake crib is located 1,750' offshore in about 22' of water. Rock and supporting steel structure are removed down to approximately 11' above the lakebed. The divider wall is removed. One of the 6' feeder pipes sits higher than the other three. It is removed or cut off at approximately 11' above the lakebed then back filled. The remaining 60' opening above the intake cones is covered with a trash rack. The trash rack has a steel super structure and a high-density polyethylene (HDPE) trash rack having approximately 7"x18" openings. The RTD's that are used to monitor ice melt operations are moved and upgraded as necessary to monitor the new ice melt flow characteristics. This work can be performed during plant operating condition. One or more circulating water pumps may be stopped as required to support work closer to the intake funnels. Construction is accomplished in two phases. The first removes most rock from above the water level and the south side rock down to approximately 11' above the lake bottom, and installs a trash rack over the south half of the 60' opening. The second phase removes the divider wall and the north side rock down to approximately 11' above the bottom, and installs the north side trash rack.

Summary of Safety Evaluation: No accidents previously evaluated in the CLB describe loss of circulating water as an initiator to an accident. Loss of water supply to the forebay through the 14' pipes event has been evaluated in the CLB. Installation procedures ensure a reliable supply of Lake Michigan water, regardless of weather or lake conditions. Therefore, this modification does not increase the probability of occurrence of an accident or event previously evaluated in the CLB.

The intake crib is non-safety-related, non-QA and non-seismic but is considered important to safety. This modification improves the reliability of the intake crib. The modification includes detailed analysis of the new crib configuration to predict ice-melt operations and verify equivalent or better operation during winter months. The coarse trash rack is constructed of HDPE, which has superior resistance to frazil ice accumulation. Head loss through the existing crib is relatively insignificant. The modification removes the upper portion of the crib and installs a coarse trash rack. The head loss through the new configuration is equally insignificant. The chlorination/dechlorination system operation is unaffected by this modification.

NRC/AEC Safety Evaluation Report NPC-35592 states the two 14' diameter offshore intakes are not designed to Class 1 standards and could be damaged. In this event, lake water for the service water pumps can be provided directly to the screenwell from the circulating water discharge flume via an ice melt valve. AOP-13A, Circulating Water System Malfunction, provides this guidance in the event of blockage of the intake crib and dropping forebay level. A letter to the NRC dated January 12, 1990, committed to inspect the outside of the intake crib as part of the response to GL 89-13, this commitment remains in effect. The commitment may be revised as necessary as part of the on going resolution of GL 89-13. The new intake crib has the trash rack installed year-round, this is permissible due to the superior resistance to frazil ice accumulation. Therefore, this modification does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB. No accident, event, or malfunction of equipment takes credit for the configuration of the intake crib. Service water is required to mitigate the consequences of design basis accidents, but water can be supplied directly to the forebay via the circulating water discharge flume. Therefore, this modification does not increase the radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB.

The configuration of the intake crib is not discussed in the Basis for TS. The normal service water supply is through the intake crib. This modification does not affect normal forebay levels or temperatures. Therefore, this modification does not reduce the margin of safety defined in the Basis for TS. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0017)

MR 01-012, Removal of Doors 58, 335 and 337.

The modification removes existing doors and replaces them with walls. The doors being removed are Door 58, located at the south end of the operating level of the turbine building; door 335, located on the south end of the east wall of the pump house; and door 337, located on the east end of the north wall of the pump house. Doors 335 and 337 are non-vital security doors that are normally locked, door 58 is a non vital security door that is normally open but can be locked as required. Door 58 is an industrial fire door.

Summary of Safety Evaluation: Removal of the doors does not increase the probability of occurrence of an accident or event previously evaluated in the CLB. The doors are not systems, structures or components associated with the cause of an analyzed accident or event or relied upon to prevent an accident. The presence or absence of the doors does not affect the probability of the analyzed accident occurring. Removal of the doors and installation of the walls does not increase the probability of occurrence of a malfunction of equipment important to safety. The doors are not credited directly or indirectly in the accident analysis identified in Section 14 of the FSAR. Barriers of equal or greater ability to perform their function are replacing the industrial fire and non-vital security doors. The doors are not required to prevent, mitigate, or respond to other design basis events. The doors are not required to function during a high-energy line break event. The doors do not cause a transient or event that results in a challenge to safeguards systems, function or equipment.

Removal of the doors and the installation of the walls do not increase the radiological consequences of an accident, event or malfunction of equipment important to safety previously evaluated in the CLB. The doors are not relied upon to prevent the consequences of an accident analyzed in the CLB. The door removal and wall installation does not increase the release rate or release duration of radioactive materials. The presence or absence of the doors and replacement walls has no bearing on the possibility of an accident or event occurring. The removal of the doors and the installation of the walls do not affect equipment important to safety or cause damage that would result in an accident different from those previously evaluated in the CLB. Removal of the doors and the installation of the walls have no effect on margin of safety defined in the Basis for TS. The doors or walls are not addressed in the TS. The walls being installed provide fire and security barriers equal to or better than the existing doors. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0006)

MR 01-052, Steam Generator Blowdown Isolation Circuitry

The modification converts part of the Unit 1 and Unit 2 steam generator blowdown (SGBD) isolation circuitry from a function of the motor driven auxiliary feedwater pump (P-38A and B) breaker closures (1B52-12C and 2B52-31C) to a function of the motor-driven auxiliary feedwater pump (P38A and B) auto starts. In addition to the current SGBD isolation, interlocks from; the steam to turbine-driven auxiliary feed pump motor operated valves, a containment isolation signal and a high radiation signal, the final configuration allows Unit 1 and Unit 2 SGBD isolation to occur automatically as a function of Low-Low level in a steam generator, safety injection or an AMSAC actuation. These three isolation functions are captured automatically by using the auxiliary feedwater motor driven pump breaker normally closed "b" contacts in the SGBD isolation control circuitry.

When the pump starting circuitry receives a signal from any of the three isolation functions, the motor-driven AFW pumps are started, thus opening the associated breaker "b" contact in the isolation control circuitry and providing SGBD isolation. Current plant operating procedures allow for bypassing this automatic isolation function associated with the auxiliary feedwater motor-driven pump breaker contacts while performing testing, breaker maintenance and unit start-up and shut down. Once bypassed, and in accordance with single failure criteria the auxiliary feedwater system is placed in a condition of limited capacity until SGBD is manually isolated. Manual action is permitted during testing through the use of dedicated operators.

However, long term operation does not meet the intent of testing and manual action is not relied upon for this automatic function, that supports a safeguards function. This modification removes the associated AFW pump breaker "b" contacts from the SGBD isolation circuits and provides isolation from a new set of relay contacts (via a new set of relays) that receive inputs from Low-Low level in a steam generator, safety injection or an AMSAC actuation. The new relays are installed in parallel with the existing motor-driven auxiliary feedwater pump starting relays. The new configuration allows some testing activities, breaker maintenance and unit start-up and shut down to occur without bypassing the automatic blowdown isolation signal.

Summary of Safety Evaluation: The modification changes the SGBD function from automatic to manual upon a manual AFW system start. This is acceptable since manual AFW initiation is not credited in an accident analysis. An automatic AFW system demand still results in automatic isolation of the SGBD. In addition to revising the isolation circuitry associated with the motor driven auxiliary feedwater pumps, the existing key bypass switches are replaced with new on/off bypass control switches. These bypass switches allow bypassing of the steam to turbine-driven auxiliary feed pump motor-operated valve interlocks in the SGBD isolation control circuitry. The bypass feature allows for testing of the motor-operated steam admission valves (and for use of the turbine-driven auxiliary feedwater pump during hot standby, start-up and shutdown) without losing the SGBD capabilities. Although this defeats the purpose of the automatic isolation signal during testing of the turbine-driven auxiliary feedwater pump motor-operated steam admission valves, an LCO is entered during this time. Therefore, no single failure is credited, allowing for sufficient supply capacity of the auxiliary feedwater system (through the motor-driven AFW pumps) if demanded, since the motor-driven pump start provides SGBD isolation.

SGBD isolation indirectly supports the mitigation of accidents and events analyzed in Chapter 14 of the CLB, however, SGBD is not an initiator of an accident or event. Therefore, the final configuration of this modification allows the auxiliary feedwater system to perform its design function during accidents or events as described in Chapter 14 of the CLB, does not increase the probability of a malfunction of equipment important to safety, does not create the possibility of a malfunction of equipment important to safety of a different type, or decrease the margin of safety defined in the basis for TS.

This modification only affects the AFW auto start function and SGBD isolation. Therefore, the only accidents or events that could be affected are a loss of AC power or a loss of normal feedwater and the AFW systems capability to mitigate these events. However, a loss of the AFW auto start function or a loss of SGBD isolation is not an accident, event or initiator to an accident or event as evaluated in the CLB. Further, as discussed above, the overall reliability of the AFW system supply capability is not decreased.

Therefore, this modification does not increase the probability of occurrence of an accident or event, create an accident or event of a different type, or increase the radiological consequences of an accident or event previously evaluated in the CLB.

Prerequisites have been established to allow this work to be performed during any mode of operation on either unit. During interim configurations, it is necessary to remove auxiliary feedwater pumps from service. This is procedurally controlled in accordance with voluntary LCO entries per TS 15.3.4.c. It is noted that only one pump is removed from service at a time and restored to service prior to taking the other pump out of service.

In addition, SGBD on each unit is removed from service and isolated while installation and testing is performed. While there is no limitation per the licensing basis for securing SGBD, this evolution is limited to the least time practical and is monitored for actions consistent with procedural requirements. Additionally, during the installation process, applicable procedures are followed to ensure that other equipment near the work remains unaffected. Therefore, during installation and testing no components are disabled beyond what is currently analyzed for in the accident analysis, and no equipment is operated beyond the acceptance criteria described in the TS. The change does not pose a USQ nor does it require a change in the TS.
(SE 2001-0034-001)

TEMPORARY MODIFICATIONS

The following temporary modifications (TMs) were implemented in 2001:

TM 01-009 and TM 01-010, Temporary Power for Motor Control Center 1B-31 and 1B-32.

TM 01-009 and 01-010 supply temporary power to MCCs 1B31 and 1B32 from electrical switchgear 1B04 during the Unit 1 defueled window. This may be considered a partial crosstie of 1B-03 and 1B-04, and took place during the U1R26 refueling outage. To enable the change, cables are routed from switchgear 1B04 and 1B-42 to the two MCCs. The equipment fed from 1B31 and 1B32 are considered inoperable from a TS standpoint while the temporary modifications are installed.

Summary of Safety Evaluation: The TM serves loads normally supplied by 1B03. The situation of servicing 1B03 loads from 1B04 has been previously analyzed by CR 99-1856. TMs 01-009 and 01-010 assure that circuit loading is bounded by the conditions outlined in this analysis. Although safety-related loads receive reliable power from a safety-related power source, they are considered inoperable. The change has no adverse effects on EDG operation or the initiation of safety injection.

The activity ensures necessary maintenance activities can be performed on bus 1B03 while simultaneously supplying required loads. The supply of this temporary power is bounded by analysis for the supply of 1B03 loads from 1B04. This change does not pose a USQ nor does it require a change to the TS. (SE 2001-0010)

TM 01-016, Temporary Power Supply to NES and Training Buildings.

MR 99-007*A, installs a new 13.8 kV feed to the sewage treatment plant. This new feed is tapped into the cables that currently feed switch-fuse unit H-08 it supplies the NES (X-65) and training building (X-66) transformers. To install the tap into the existing line, the line must be de-energized for the duration of the installation. The temporary diesel generator is used to supply temporary power to the NES and training buildings for the duration of the installation.

The training and NES building loads are supplied from a diesel generator while the normal supply to these buildings is out of service. A 125KVA generator is used to supply the loads normally supplied by X-65 (500KVA) and X-72 (750KVA). Multiple 4/0 cables are used to connect the diesel generator to the cables that supply B-60 (NES) and B-65 (training). Circuit protection at the motor control centers remains unchanged with the use of the alternate power supply.

Summary of Safety Evaluation: There is no equipment important to safety, powered from the NES (X-65) and training (X-66) building transformers. The temporary diesel generator is isolated from the 13.8 kV system during installation and operation by opening disconnects on switch-fuse unit H-08. The temporary modification to power the loads normally supplied from the NES (X-65) and training (X-66) buildings does not increase the probability of occurrence or consequences of new or previously evaluated accidents, events, or malfunctions of equipment. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0018)

TM 01-020, Emergency Water Supply to Vacuum Priming Pump P-041.

TM 01-020 will provide water to the common P-041 vacuum priming pump, because the normal water source (SW) for this pump is not available due to U1R26 outage activities. The water supply consists of a hose attached to DI-00091 and runs overhead to the cleanout of Y-strainer YS-02983 on the inlet side of P-041. Water is the medium used by this pump to affect the required function of the pump (air removal from the CW system). A check valve, at the water source and an isolation valve, at the pump ensure there are no system cross-connection concerns. Review of the overhead route for the hose indicates no challenges to the structural integrity of pipe hangers or other supports, nor does it create personal safety concerns.

Summary of Safety Evaluation: The vacuum priming pump(s) are not initiators of design basis accident or scenario. They provide a means to maintain circulating water level in the condenser water boxes and the condensate coolers. There are no CLB challenges created by the operation of these pumps. Provisions have been made to address concerns regarding the cross connections of systems via the use of check and isolation valves. The intent of this TM is for use on an "as needed" basis to augment the operating unit's vacuum priming pump.

Selection of the alternate water source for the vacuum-priming pump was based on logistics and system design requirements. The vacuum priming pump and the air removal system cannot initiate a design basis accident nor can they create a scenario different from those described in the CLB. The TM provides the redundancy function, as in the current design, which is not available due to the U1R26 unavailability. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0021)

TM 01-021, Mechanical Blocking Clamp for 1CV-112A

1CV-112A T-4 Volume Control Tank (VCT) level control divert valve is not always returning completely to the VCT position after a divert to the holdup tanks. The temporary modification installs a clamp to hold the valve in the VCT position to allow work to be performed on the valve operator. This is the normal position of the valve.

The temporary modification disables the ability to divert letdown to the holdup tanks to provide space in the RCS and VCT to perform dilutions and borations. An inability to perform a dilution will cause reactor power to decrease over time. Per FSAR Chapter 9, the pressurizer has sufficient volume to borate the unit to hot shutdown if letdown is not available. The blocking device is capable of maintaining the valve position so letdown is directed to the VCT. This prevents the valve from moving to the divert position so that an inadvertent loss of RCS inventory does not occur.

Summary of Safety Evaluation: The letdown divert function is not an initiator of an accident in the CLB, nor is it relied upon to prevent or mitigate an accident described in the CLB. The letdown divert function is not mentioned in the Basis for TS so the proposed change does not create a conflict with the TS. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0023)

TM 01-025, Temporary Seal Water Supply to P-134.

A hose is run from a local de-ionized water supply, DI-00169, and connected to the P-134 blowdown evaporator bottoms pump seal inlet. The P-134 seal outlet is routed to a floor drain.

Summary of Safety Evaluation: The blowdown evaporator bottoms pump, is not a safety-related pump and is not an initiator of previously evaluated accidents or events. The new seal supply water does not increase the probability of a malfunction to P-134, nor does it affect the operation of other safety-related equipment. The use of a different seal water supply does not increase the inventory of radionuclides available for release, so this activity does not increase the consequences of an accident or event. Because this change only affects the operation of P-134, it does not create the possibility of a malfunction of safety-related equipment not previously evaluated. The required DI water flow rate to the P-134 seal is within the capability of the DI and water treatment systems, and does not affect other systems supplied by DI water. A check valve is installed in the DI water supply line. The seal water return line is routed to the floor drain at atmospheric pressure, and the DI water header pressure is considerably higher than the pressure at the P-134 seal package. These actions prevent backflow of blowdown evaporator liquid into the DI water header.

The selection of the alternate water source for the blowdown evaporator bottoms pump was based on logistics and pump design requirements. The blowdown evaporator bottoms pump does not initiate a design basis accident nor can it create a scenario different from those described in the CLB. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0029)

TM 01-026, Temporary Seal Water Supply to P-133.

A hose is run from a local DI water supply, DI-00169, and connected to the blowdown evaporator-circulating pump P-133 seal inlet. The P-133 seal outlet is routed to a floor drain.

Summary of Safety Evaluation: The blowdown evaporator circulating pump, is not a Safety- related pump and is not an initiator of previously evaluated accidents or events. The new seal supply water does not increase the probability of a malfunction to P-133, nor does it affect the operation of other safety-related equipment. The use of a different seal water supply does not increase the inventory of radionuclides available for release, so this activity does not increase the consequences of an accident or event. Because this proposed change only affects the operation of P-133, it does not increase the possibility of a malfunction of safety-related equipment not previously evaluated. The required DI water flow rate to the P-133 seal is well within the capability of the DI and water treatment systems and does not affect other systems supplied by DI water. Check valves are installed in the DI water supply line. The seal water return line is routed to the floor drain, then to the waist holdup tank at atmospheric pressure. The DI water header pressure is considerably higher than the pressure at the P-133 seal package. These actions prevent backflow of blowdown evaporator liquid into the DI water header.

The temporary modification also installs a tee after DI-169 to supply DI water to either/or both P-133 and P-134. The selection of the alternate water source for the P-133, blowdown evaporator circulating pump, was based upon logistics and pump design requirements. The blowdown evaporator, circulating pump does not initiate a design basis accident nor can it create a scenario different from those described in the CLB. The change does not pose a USQ nor does it require a change in the TS. (SE 2001-0030)

TM 01-030, Trending Pressure Indicator Installation on condensate cooler inlet, Unit 1 HX-024

The temporary modification installs a pressure gauge on the inlet vent line of the tube side of the 1HX-024, condensate cooler. The pressure gauge provides the trend of the inlet pressure as fouling of the tube sheet increases. A pressure gauge is installed downstream of manual valve 1CW-0021, the condensate cooler inlet vent. Installing the gauge downstream of 1CW-0021 eases installation of the gauge, since the gauge can be installed with the valve shut, the operation of the cooler is not affected.

Summary of Safety Evaluation: The circulating water system is not the initiator of design basis accidents or scenario. This system is designed to provide a supply of cooling water to condensate coolers, and a heat sink for converting steam to condensate. As such, no CLB changes are required by the installation of the pressure gauge to the vent pipe on the inlet side of the 1HX-024, condensate cooler. Installation of the gauge can be performed at power via manipulation of valve 1CW-00021. This valve can remain shut during removal of the pipe cap and installation of the pressure gauge, and then be opened to introduce circulating water pressure to the gauge again.

Using the pressure gauge to trend inlet pressures optimizes maintenance of the condensate cooler. The CW system cannot initiate design basis accident or create a scenario different from those described in the CLB. The modification does not pose a USQ nor does it require a change in the TS. (SE 2001-0039)

MISCELLANEOUS EVALUATIONS

CR 96-394, Service Water Pump Discharge Pressure Indicator Isolation Valves

The activity adds service water pump discharge pressure indicator isolation valves on each service water pump. The maximum service water pressure at the valve elevation is 123 psig. The ½" brass isolation valve has a pressure rating of 3000 psig at 100°F. This large pressure-rating margin shows that the pressure retaining capability of the pressure indicator installation is not impaired. This temperature rating is above the 80°F temperature considering service water flow model calculation. This is a brass valve installed in a copper tubing line. (These are appropriate and commonly used materials for Lake Michigan water.) This is a static pressure measurement pressure indicator therefore there is no flow through the isolation valve. Hence, erosion is not a concern. The two additional tubing joints are not likely to fail, because they have a pressure rating equal to the tubing. The tubing pressure rating is at least 750 psig for the thinnest wall tubing commercially available to be used. The valve and its tubing joints are supported in a steel frame, which protects the valve and its tubing joints from damage. The extensive use of compression fittings for tubing joints in safety-related applications demonstrate it to be reliable joint.

Summary of Safety Evaluation: The seismic evaluation completed by Condition Report CR 96-394 Action #5 concludes the pressure indicator and its isolation valve installation is qualified for seismic Class 1 service. The valve is supported in steel channel, which in turn is supported off the pump support flange such that the ground acceleration is not amplified at the valve. The ½" brass needle valve is seismically rugged in accordance with the GIP for Seismic Verification of Nuclear Plant Equipment. In addition, the valve is exposed to seismic loads that are lower than the maximum loads used in the experience database formulated by the SQUG in response to NRC Unresolved Issue A-46. Therefore, the valve and its installation do not increase the probability of failure by a seismic event.

Adding service water pump discharge pressure indicator isolation valves does not challenge fission product barriers and does not increase the source term. The number of pumps required to be in service are not affected by the proposed activity. The pressure indicator isolation valves are not connected to the ring header. Therefore, its continuity and integrity is not affected. The activity does not change a pump to header alignment combination already permitted or evaluated. Capability to isolate non-essential service water loads during an accident is not changed. This change did not pose a USQ nor does it require a change in the TS.
(SE 2001-0005)

FPER Fire Protection Evaluation Report, Appendix R Rebaselining

The FPER was revised to support the Appendix R Rebaselining effort. In general, the FPER is reformatted to better follow the guidelines set forth in BTP 9.5.1.

Summary of Safety Evaluation: This safety evaluation evaluates the FPER change and the creation of new Safe Shutdown Analysis Report (SSAR) and Fire Hazard Analysis Report (FHAR) documents. The SSAR identifies the strategy for achieving safe shutdown in areas, and gives a detailed listing of what is credited for satisfying Appendix R requirements. The FHAR contains detailed fire hazards analysis information. The changes to the safe shutdown components, identifies changes to reflect the new computer analysis program, Safe Shutdown Analysis Management System (SSAMS). A table of fire zones is added to correlate each fire zone to a fire area. The Appendix R safe shutdown equipment list for each fire zone is revised to delete the detail of cable and raceway for each component. This information does not change the fire hazard analysis.

The changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. This change did not pose a USQ nor does it require a change in the TS.
(SE 2001-0031)

FSAR 4.4, RCP Flywheel Examination Frequency

The frequency of RCP flywheel examination is revised from once every 5 years to once every 10 years. In addition, the specified examination techniques of "a surface examination of the reactor coolant pump flywheel bore and keyway, a 100% volumetric examination of the flywheel, and a visual examination of the surface of the flywheel" (as specified in the FSAR Section 4.4 and the ISI Long Term Plan) is revised to, "either an ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination of exposed surfaces defined by the volume of the disassembled flywheels once every 10 years." The inspection requirements are removed from the FSAR and placed in the Technical Requirements Manual (TRM).

Summary of Safety Evaluation: Inspections of the RCP flywheel are being performed in response to RG 1.14. NDE is periodically performed as a preventive maintenance measure to detect inservice flaws that may develop that may result in failure of the RCP flywheel. Using industry experience and advanced analysis techniques, it has been determined that flywheel failure from design operation is not credible. The NRC has reviewed this information but has decided that some form of NDE be periodically performed. The NDE performed has no correlation to crack initiation or propagation (i.e. does not prevent failure). It does not alter the flywheel physically. The changes in inspection frequency/methodology had no impact on accident analysis previously performed, RCP design or operation.

A reduction in the frequency and type of examinations has no impact on the integrity of the RCP flywheel. These actions implement best practices and ALARA techniques without a reduction in nuclear safety. This change did not pose a USQ nor did it require a change in the TS.
(SE 2001-0051)

FSAR 6.2.2, Safety Injection System Design and Operation

The statement in Chapter 6.2.2 of the FSAR that reads, "The operators are tested to open the valve against pressures in excess of that occurring in the containment during a loss-of-coolant-accident," refers to the RHR Pump Sump B Suction Valves (SI-850A&B). The change being evaluated revises the wording of the above, discussed statement in Chapter 6.2.2 to read, "The operators are tested to verify that they can open the valves against pressures in excess of that occurring in the containment during a loss-of-coolant accident." This clarifies that while these valve operators are, "tested to open the valve against pressures in excess of that occurring in the containment during a loss-of-coolant accident," they are not opened against a containment actually pressurized to pressures in excess of that occurring during a loss-of-coolant accident.

Summary of Safety Evaluation: The sump "B" valves are required to open during the plant's recovery from a LOCA after the borated water contained in the refueling water storage tank has been injected into the faulted reactor coolant system. Quarterly testing, via the IST Program, assures they can perform the safety-related function of opening under accident conditions and meets the intent of the FSAR statement.

This change to the wording in Chapter 6.2.2 of the FSAR does not physically alter the plant. The testing is in full compliance with ASME Section XI requirements and assures the operational readiness of the valves. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0003)

FSAR 9.1, Component Cooling Water

The change re-classifies the component cooling water system (CC) from closed loop outside containment (CLOC) to closed loop inside containment (CLIC). This change requires no physical change to plant systems, structures or components. A change in the designation of containment isolation valves satisfies the requirements of Class 4 containment penetrations. This includes moving the CIV designation from CC-755A to CC-754A, from CC-755B to CC-754B, and from CC-767 to CC-766. The first two changes move the designations from check valves to motor-operated valves, while the last one moves the designation from a check valve to a manual globe valve.

Summary of Safety Evaluation: The CC System is re-classified from its current designation of CLOC to CLIC. This change is being done via analysis. No physical changes are needed to the plant. Calculations and analysis were performed in support of this change. The system is capable of being designated as a CLIC using the hardware that is currently in place. An analysis was performed of lines inside the reactor cooling pump cubicles with greater than 6" diameter, i.e. the accumulator piping, pressurizer surge line, and the residual heat removal lines.

The leak-before-break assessment uses General Design Criteria 4, Environmental and Dynamic Effects Design Basis. It states that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analysis reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." Essentially, the analysis is required to show that there is a high probability of piping systems leaking sufficiently to provide indication to operators such that recovery actions could be taken prior to complete rupturing of the pipe hence the pipes would leak before they would break. Leak-before-break, methodology was accepted on the lines that were reviewed for this change.
This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0007)

FSAR 14.3.1, Small Break Loss of Coolant Accident Analysis

A +13°F Peak Cladding Temperature (PCT) model assessment penalty was reported for the SBLOCA for both units. The penalty is related to modeling errors discovered in the NOTRUMP methodology. The PCT analysis and assessment results are maintained current in Section 14.3.1. Page 14.3.1-4 is changed to reflect the new total ECCS evaluation model change penalty of 400°F (from 387°F), and the resulting PCT of 1209°F (from 1196°F).

Summary of Safety Evaluation: The change revises the SBLOCA PCT results in Section 14.3.1 that resulted from NOTRUMP modeling errors in the SBLOCA analysis. No assumptions, inputs, or initial conditions to the analysis have been changed. The resulting PCT of 1209°F is well below the 10 CFR 50.46 limit, of 2200°F. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0009)

FSAR 6.3.2, Containment Air Recirculation Cooling System (VNCC)

FSAR Section 6.3.2 is revised to delete reference to the airflow rate where the containment fan coolers (CFCs) remove heat from the containment during design basis accident conditions.

Summary of Safety Evaluation: By having the CFC minimum design air flow rate specified in the FSAR, it could be misinterpreted as a critical performance requirement of the containment accident recirculation fans for assuring that the CFCs satisfy their design basis heat removal rate. This is not the case, however, as Engineering Evaluation 2001-0011 has demonstrated that CFC thermal performance is relatively insensitive to variations in air flow across the coolers, and that their thermal performance requirement can be satisfied at air flow rates well under the value stated in the FSAR. The design, configuration, operation, performance, testing and reliability of the containment accident recirculation fans are not affected by this change. The specified temperature and pressure limits of the containment are not exceeded because the ability of the CFCs to remove heat from the containment at the post-accident design basis rate is not affected. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0024)

FSAR 9.10, Fire Protection System

The change moves the fire protection information from the FSAR to the FPER with some editorial changes, clarification of wording, and deletion of information. The change does not involve a physical change to the plant. Both the FSAR and FPER are licensing basis documents.

Summary of Safety Evaluation: The design of the plant systems is not affected by the change. The change does not involve new system interactions or connections, and system integrity is maintained. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0037)

FSAR Appendix A.5.1, Seismic Design Classifications

FSAR Appendix A.5.1, Seismic Design Classifications describing the interface between a seismic Class I system and a lower seismic Class system is clarified. This is accomplished by adding a phrase at the beginning of the paragraph, "When required to maintain a system's safety-related functions." The revised paragraph reads, "When required to maintain a system's safety-related functions, the interface between a Class I system and lower Class system is at a normally closed valve, a valve which is capable of remote operation from the control room, or a valve which is capable of self actuation."

Summary of Safety Evaluation: The initiating event of concern for seismic design class interfaces is a safe shutdown earthquake which is a natural phenomenon whose probability is not affected by the design of plant components. Seismic events are already evaluated in the CLB. This FSAR clarification does not change the current design requirements for a seismic class interface component, which is to be designed to the same criteria as the seismic Class I system. Suitable isolation between seismic designs Class 1 systems, lower class systems continue to be provided so release rates or radiological consequences from accidents, and events already evaluated remain unchanged. The safety-related functions of a system or components are not being changed. The commitments, rules, and practices for determining acceptability of isolation between safety-related functions and non-safety-related functions are not changed. The licensing basis FFDSAR did not include the current FSAR description of seismic design interface. Instead, the seismic design interface was described by:

- The response to AEC Question #4.13 which is "The communication in an understandable manner by list or verbiage of the Class I system interfaces would be most difficult. However, Appendix A, Section 2 lists those systems and equipment by Class. By referring to the system flow diagrams in Section 9 and 10, the interfaces may be easily seen in a meaningful graphical manner. The interface between a Class I system and lower Class system would be at a normally closed valve or a valve which is capable of remote operation from the control room."

- These flow diagrams depicted the plant design at the time of licensing, which, we know from a review of the plant physical design, utilized a select few normally open valves as the interface between seismic Class 1 and lower Class systems. This response invokes the flow diagrams to show the specific seismic interfaces, and then generalized about the nature of these interfaces.

- FFDSAR Appendix A, Section 1.0 Definition of Seismic Design Classifications page A-3, states "Primary steady state stresses are limited so that the function of the component, system, or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant." Page A-4 states, "The component is analyzed using seismic loads as obtained from building response calculations to show that stresses and deflections are within allowable limit and does not result in loss of function." Therefore, the seismic design philosophy design permits use of a normally open valve in certain selected and evaluated situations.

Therefore, there is no impairment of the safety functions in a seismic Class 1 system. The malfunction does not impair the capability to safely shut down. The malfunction does not cause radiological release to exceed design basis criteria. The change does not alter margin of safety in TS. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0002)

FSAR 11.0, Waste Disposal System

The changes include the addition of several waste disposal system components in the component data table to be consistent with plant configuration and the design basis documentation. The changes do not involve a physical change to the plant.

Summary of Safety Evaluation: The design of the plant systems or procedures is not affected by the changes. The changes do not involve new system interactions or connections, and system integrity is maintained.

Revising FSAR Section 11.0 as described does not result in a significant increase in occupational radiation exposure, or a significant unreviewed environmental impact, and does not conflict with a license condition as contained in the ISFSI Certificate of Compliance are involved. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0011)

FSAR 9.3, 9.3.3, 11.1.2, 11.2, 11.6, 14.1.4, 14.2.2, 14.2.4, 14.2.5, Mixed Bed Deminralized Allowable Flow Increase and SGTR/MSLB Dose Consequence Revision

The changes increase the allowable letdown flow through mixed bed demineralizers from 72 gpm to 90 gpm. This accommodates the increase in letdown flow through the 40 and 80 gpm orifices because of the return to a primary system operating pressure of 2250 psig. A revision to the dose consequences for the Steam Generator Tube Rupture (SGTR) and the Main Steam Line Break (MSLB) current FSAR analysis corrects for non-conservative input parameters (i.e., primary leakage, letdown flow, letdown demineralizer iodine removal efficiency, and RCS mass) used to calculate the equilibrium iodine appearance rate.

Summary of 50.59 Evaluation (performed per the new 50.59 rule): Increasing the design flow across the mixed bed demineralizers is not an initiator of new malfunctions and no new failure modes are introduced. The letdown piping to the mixed bed demineralizers is designed for a flow rate of 120 gpm. Increasing flow to the mixed bed demineralizers up to 90 gpm is within the design capacity of the system and does not increase the risk of a pipe rupture. Calculations demonstrate the mixed bed demineralizer can support the increase of letdown flow. In addition, revisions to the SGTR and MSLB dose correct the effects of the non-conservative input parameters, and do not introduce new malfunctions or failure modes. Values chosen are with in the design of plant systems. The results of the correction to the iodine appearance rate have a less than minimal increase on the dose consequences of the SGTR and MSLB accident analysis. This change did not require prior NRC approval nor did it require a change in the TS. (EVAL 2001-0004)

ISI IWL Containment Inspection Program

Implementation of the ISI IWL, containment Inspection Program as required by 10 CFR 50.55a, provides a more conservative containment inspection than required by TS 5.5.17. As a result of implementing ASME Subsection IWL requirements, one additional vertical tendon and one additional dome tendon are required to be added to the current inspection program. FSAR Section 5.7.1.5 is revised to reflect this additional requirement. Other more restrictive changes required by the IWL program over the existing containment inspection include NDE qualified examiners, specific detailed concrete recording criteria as provided by industry standards, specific reporting criteria for certain indications, and program oversight by the IWL Responsible Engineer.

Summary of Safety Evaluation: Since a more conservative containment inspection program is implemented with the IWL containment inspection program, the degraded conditions that would affect the safety-related functions of containment are documented, evaluated and corrected.

There is no change in the margin of safety to existing TS as a result of implementing the IWL containment inspection program. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0013)

Unit 1 Cycle 27 Reload

Unit 1 operated in Cycle 26 with 121, 14x14, Upgraded Optimized Fuel Assemblies (OFAs). For Cycle 27, 14x14, 422 VANTAGE+ (422+) fuel assemblies are introduced into the Unit 1 core. 37 OFAs are replaced with 9 fresh Region 29A (4.60 w/o U-235), and 28 fresh Region 29B (4.95 w/o U-235) 422+ fuel assemblies. The 422V+ fuel product was licensed under the Fuel Criteria Evaluation Process (WCAP-12488-A) and is supported by new analysis that has been accepted by the NRC. License Amendments 193 (Unit 1) and 198 (Unit 2) approved the use of the 422V+ fuel product and cover required TS changes. A separate evaluation (SE-2000-0070) was approved. It 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 this new fuel product and the fuel upgrade program, and not to the core reload, are not addressed by this evaluation.

Summary of Safety Evaluation: The safety evaluation covers the mechanical design, nuclear design, thermal-hydraulic design, power capability, and FSAR accidents that apply to the U1C27 reactor core and covers all modes of operation for U1C27. The Cycle 27 design results in no unreviewed safety questions or TS changes provided that the following conditions are met: end-of-cycle 26 burnup is bounded by 15,600 to 17,100 MWD/MTU; U1C27 burnup does not exceed the End-Of-Full Power Capability (which is defined as control rods fully withdrawn and less than or equal to 10 ppm or boric acid at the Cycle 27 rated power condition of 1518.5MWt) plus up to 1,500 MWD/MTU of power coast down operation; adherence to the plant operating limitations given in the TS; safety aspects on the reactor internals of utilizing peripheral power suppression assemblies have been assured; effect of the U1C27 design on the boron dilution event in cold shutdown is assessed to be acceptable.

The U1C27 reload core design meets applicable design criteria or has been shown to maintain the same levels of safety as considered in the reference design basis evaluations and ensures that pertinent licensing basis acceptance criteria are met. Though fuel and core design are not directly related to the probability of 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 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. An evaluation confirms that Cycle 27 core does not return to critical following a SLB. Therefore, the evaluation of the containment integrity during a SLB is valid. No new performance requirements are being imposed on a system or component such that design criteria is exceeded, nor will the changes cause the core to operate in excess of pertinent design basis operating limits. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0025)

Q-List, Downgrade Component Cooling Water Heat Exchanger Service Water Relief Valves from Safety-Related to Augmented Quality

The change downgrades SW-02800, SW-02801, 1SW-02802, and 2SW-02923 component cooling water heat exchanger relief valves from safety-related to augmented quality. They are classified as safety-related for maintaining the pressure boundary associated with cooling/seal water supplied to safety-related equipment. This change documents this function is not safety-related with respect to the relief valves and would scope the relief valves as augmented quality to preserve their ASME Section XI classification and seismic 1 design class.

Summary of Safety Evaluation: The only time these relief valves perform an overpressure protection function is when the associated heat exchanger is isolated and out of service. Under these conditions, the heat exchanger cannot perform a safety-related function. Therefore, the overpressure protection function performed under these conditions is not safety-related and a failure to perform this function could not cause or increase the probability of an accident or event. Under normal and accident conditions, should the relief valves fail open, the service water that would leak through the relief valve would have already performed its cooling function. The overpressure protection function performed by these relief valves is not required at these times as the heat exchangers are provided with a flow path to the lake, which prevents over-pressurization. In addition, the very limited leakage past the valves would not affect the service water system's ability to provide cooling to other safety-related head loads or cause a flooding concern for other safety-related components. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0040)

TRM 3.8.3, Insert Standby Emergency Power Source Inspections

The change allows performance of an operability determination instead of an immediate inoperability declaration if it is determined that an EDG was not inspected in accordance with the manufactures recommendations.

Summary of Safety Evaluation: Failure to follow the manufacture's recommendations does not necessarily render the EDG inoperable. The required operability determination evaluates the situation and determines the operability status of the equipment. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0056)

TRM 3.5.1, Chemical and Volume Control System

This change deletes TRM Charging Pump Surveillance requirement TSR 3.5.1.3 and associated Bases discussion. The current surveillance requirement implies the pumps perform a credited safety-related function that would subject the pump to surveillance testing under 10 CFR 50.55a These pumps perform an augmented quality function during an Appendix R event.

Summary of Safety Evaluation: The boric acid subsystem is not necessary to mitigate a DBA or transient and is outside of the scope of 10 CFR 50.55a. The charging pumps are not credited with the performance of an active safety-related function that would require the pumps be included in the IST Program. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0057)

WP 2001-011, Unit 1 Turbine Building SW Flow and Pressure Measurements

This work plan measures service water flow to the Unit 1 turbine hall for two different configurations. Service water flow to the Unit 1 turbine hall is a flow path that divers flow away from required users during a design basis event. One configuration is the same as the configuration analyzed in the design basis calculations, e g. all variable flow valves fully open. The other configuration has the same open flow paths as the first, but also includes five open bypasses. The bypasses are around piping subsections that experience small pressure drops and therefore, the additional pressure drop reduction has a negligible effect on the flow resistance of the entire subsystem. However, to ensure that potential additional SW flow to the turbine hall does not exceed the current analysis, the work plan prohibits placing in service a particular combination of loads totaling 650 gpm, which were assumed to be in service in the design basis analysis. Both configurations are bounded by the current design basis analysis.

Summary of Safety Evaluation: The Unit 1 turbine hall SW flows is a diversionary flow from safety-related components needed in an accident. Therefore, its safety significance lies in its effect on the reminder of the SW system. This effect is unchanged from the effect in the CLB because the configurations that are used in this work plan are bounded by the current SW hydraulic analysis. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0012)

WO 9928639, FIC-609 P-1A RCP CC Return Header Flow Indicator Controller

The P-1A RCP CC return header flow indicator controller FIC-609 is removed and replaced. The control room alarm for low flow from component cooling for the "A" RCP loop is disabled and the automatic function, that closes AOV CC-761A, becomes inoperable. CC-761A has an active non-safety-related function that closes to preclude leakage of high pressure radioactive RCS fluid beyond the boundary of the high pressure CC piping following the reactor coolant pump thermal barrier rupture design basis event.

Summary of Safety Evaluation: The containment isolation function for the CC system is safety-related to mitigate an event that could result in excessive discharge of primary system fluid outside containment. Without isolation there is the potential for inadequate sump volume for transfer to the post-LOCA recirculation phase that could result in inadequate core cooling and offsite dose consequences comparable to the guidelines of 10 CFR 100. Safety-related motor-operated valve CC-759A isolates RCS break flow to systems outside containment. Relief valve CC-763A provides overpressure protection to the low-pressure RCP cooling water piping up to the first isolation valve (CC-759A). Additionally, the CC system is equipped with radiation monitoring equipment that can provide the operators an indication of RCS leakage into the CC system and the need to isolate thermal barrier cooling flow via CC-759A.

Maintenance on FIC-609 P-1A RCP CC return header flow indicator controller disables an automatic function, as is described in the CLB. This activity does not place the plant outside of the CLB nor will it introduce additional failure modes not previously evaluated. This change did not pose a USQ nor did it require a change in the TS. (SE 2001-0038)

WO 9944420, Maintenance of FIC-613 or FIC-609 CC Return Header Flow Indicator Controller

The P-1B RCP CC returns header flow indicator controller FIC-613 switches are adjusted, and if necessary, removed and replaced. During the course of the maintenance, the control room alarm for low flow from component cooling for the "B" RCP loop is disabled and the automatic function, that shuts AOV CC-761B, is inoperable. Manual control via the operating switch on the control board is maintained. CC-761B has an active non-safety-related function that closes to preclude leakage of high pressure radioactive RCS fluid beyond the boundary of the high pressure CC piping following the reactor coolant pump thermal barrier rupture design basis event.

Summary of Safety Evaluation: The containment isolation function for the CC system is safety-related for mitigation of an event that could result in excessive discharge of primary system fluid outside containment. Without isolation there is the potential for inadequate sump volume for transfer to the post-LOCA recirculation phase that could result in inadequate core cooling offsite dose consequences comparable to the guidelines of 10 CFR 100. Safety-related motor-operated valve CC-759B has a safety-related function to isolate RCS break flow to systems outside containment. Relief valve CC-763B has the safety-related function to provide overpressure protection to the low pressure RCP cooling water piping up to the first isolation valve (CC-759B). CC-761B is neither necessary nor credited to perform this isolation function. Additionally, the CC system is equipped with radiation monitoring equipment that can provide the operators with an indication of RCS leakage into the CC system and indicating the need to isolate thermal barrier cooling flow via CC-759B.

Maintenance on FIC-613 P-1B RCP CC return header flow indicator controller disables an automatic function that is described in the CLB. This activity does not place the plant outside of the CLB nor introduce additional failure modes not previously evaluated. This change does not pose a USQ nor did it require a change in the TS. (SE 2001-0044)

COMMITMENT CHANGE EVALUATIONS

GL 89-13, Containment Fan Cooler Testing, The containment fan coolers are monitored during normal operation for maintaining heat transfer capability. Performance testing and/or frequent regular maintenance are performed on an appropriate frequency based on the results of future testing and/or frequent regular maintenance in accordance with the guidance in Generic Letter 89-13 and Supplement 1; the frequency is not less than once every five years.

Justification for Change: A review of the results of the past performances of OI 130/131 indicates a positive trend. The results demonstrate that the various fouling prevention activities, including chlorination, EVAC treatment, Rydlyme treatment, etc., are effectively controlling fouling in the containment fan coolers.

Generic Letter 89-13 and Supplement 1 provide specific guidance directing the licensee to determine the appropriate frequency of testing and/or frequent regular maintenance based on the results of the testing and or frequent regular maintenance, after a baseline (i.e. three tests) is established. Commitments related to Generic Letter 89-13 should not specify frequencies for testing or frequent regular maintenance. (CCE 2001-001)

GL 89-13, Fire Protection Backup to Auxiliary Feedwater Pump Bearing Coolers, The fire protection system emergency backup to the cooling water supply to the auxiliary feedwater pump bearing coolers are tested and/or maintained as needed to maintain system operability. Performance testing and/or frequent regular maintenance are performed on an appropriate frequency based on the results of future performance tests and/or frequent regular maintenance in accordance with the guidance in Generic Letter 89-13 and Supplement 1. The frequency is not less than once per every five years.

Justification for Change: This Commitment Change is not intended to reflect changes in implementation; it changes the language in the commitment to acknowledge the flexibility already provided by Generic Letter 89-13. PC 73 Part 6 is a flush procedure and does not take trendable data. However, the continued operability of the auxiliary feedwater pumps demonstrates that the flushing is effective and the current frequency is appropriate. However, if in the future, a change is determined to be necessary or appropriate, the change is made in accordance with the revised commitment and Generic Letter 89-13. Generic Letter 89-13 and Supplement 1, provide specific guidance directing the licensee to determine the appropriate frequency of testing and/or frequent regular maintenance based on the results of the testing and/or frequent regular maintenance, after a baseline (i.e. three tests) is established. Commitments related to Generic Letter 89-13 should not specify frequencies for testing or frequent regular maintenance. (CCE 2001-002)

GL 89-13, Component Cooling Heat Exchangers Cleaning and Inspection Schedule, The component cooling water heat exchangers are tested and/or maintained as needed to maintain system operability. Performance testing and/or frequent regular maintenance is performed on an appropriate frequency based on the results of future performance tests and/or frequent regular maintenance in accordance with the guidance in Generic Letter 89-13 and Supplement 1, the frequency is not less than once per every five years.

Justification for Change: The commitment change is not intended to reflect change in implementation; it changes the language in the commitment to acknowledge the flexibility already provided by Generic Letter 89-13. The commitment listed describes "a yearly cleaning schedule," however this commitment was changed to performance testing. The component cooling heat exchangers are presently performance tested on a refueling frequency. They are also opened, inspected and cleaned on a two-year frequency. At this time, a change in implementation is not necessary or appropriate. However, if in the future, a change is determined to be necessary or appropriate, the change is made in accordance with the revised commitment and Generic Letter 89-13. Generic Letter 89-13 and Supplement 1 provide specific guidance directing the licensee to determine the appropriate frequency of testing and/or frequent regular maintenance based on the results of the testing and/or frequent regular maintenance, after a baseline (i.e. three tests) is established. Commitments related to Generic Letter 89-13 should not specify frequencies for testing or frequent regular maintenance.
(CCE 2001-003)

GL 89-13, Intake and Discharge Flume Inspections, The discharge flumes and the outside of the intake structure is inspected and maintained as needed to maintain system operability. Inspections and/or frequent regular maintenance is performed on an appropriate frequency based on the results of future inspections and/or frequent regular maintenance in accordance with the guidance in Generic Letter 89-13 and Supplement 1, the frequency is not less than once per every five years.

Justification for Change: This commitment change evaluation changes the language in the commitment to acknowledge the flexibility already provided by Generic Letter 89-13. The commitment describes a semi-annual inspection schedule, however, implementation of this commitment has consisted of an inspection of the discharge flume at one unit in the spring and at the other unit in the fall, coinciding with the refueling outages at each unit. Past inspections of the discharge flumes have shown that, due to the higher flow velocities, zebra mussel intrusion and subsequent growth is minimal and is rarely a course for cleaning. These inspections have been included in the forebay inspections that are performed on a refueling frequency at each unit. The intake structure inspections are performed semi-annually, coincident with the fish screen installation and removal each spring and fall. At this time, a change in implementation is not necessary or appropriate. However, if in the future, a change is determined to be necessary or appropriate, the change is made in accordance with the revised commitment and Generic Letter 89-13. Generic Letter 89-13 and Supplement 1 provide specific guidance directing the licensee to determine the appropriate frequency of testing and/or frequent regular maintenance, after a baseline (i.e. three tests) is established. Commitments related to Generic Letter 89-13 should not specify frequencies for testing or frequent regular maintenance. (CCE 2001-004)

GL 89-13, Forebay Inspections, The forebay is inspected and maintained as needed to maintain system operability. Inspections and/or frequent regular maintenance is performed on an appropriate frequency based on the results of future inspections and/or frequent regular maintenance in accordance with the guidance in Generic Letter 89-13 and Supplement 1, the frequency is not less than once per every five years.

Justification for Change: This commitment change evaluation is not intended to reflect changes in implementation; it changes the language in the commitment to acknowledge the flexibility already provided by Generic Letter 89-13. The commitment describes an inspection schedule performed on a refueling frequency at each unit, with one half of the forebay inspected during a Unit 1 refueling and the other half during a Unit 2 refueling. Inspection and cleaning at this frequency is providing reasonable control of the Zebra mussel population in the forebay. At this time, a change in implementation is not necessary or appropriate. However, if in the future, a change is determined to be necessary or appropriate, the change is made in accordance with the revised commitment and Generic Letter 89-13. Generic Letter 89-13 and Supplement 1 provide specific guidance directing the licensee to determine the appropriate frequency of testing and/or frequent regular maintenance based on the results of the testing and/or frequent regular maintenance, after a baseline (i.e. three tests) is established. Commitments related to Generic Letter 89-13 should not specify frequencies for testing or frequent regular maintenance. (CCE 2001-005)

GL 89-13, Auxiliary Feewater Turbine Inspections, Performing periodic inspection and cleaning of the turbine oil coolers will demonstrate the adequacy of the cooling water supply for the auxiliary feedwater turbines. The frequency is not less than once per every five years.

Justification for Change: In a letter dated February 11, 2000, the auxiliary feedwater pump vendor stated that cooling water to the pump stuffing boxes is required for applications with a product temperature of 175°F or greater and required for bearing housings for applications with a product temperature of 250°F or greater. Since the maximum temperature of auxiliary feedwater supply is 120°F cooling water is not required for the auxiliary feedwater pumps. PBTP 105 was also performed and it verified the vendor statements.

The auxiliary feedwater turbine (1/2P-029-T) bearing oil coolers require cooling water. Per GL 89-13, an acceptable alternative to a periodic performance test program is the performance of frequent regular maintenance. Heat exchanger maintenance is more practical for these coolers since the data required to valuate heat transfer capability of the oil coolers is difficult to obtain. These difficulties include operation of the pump for extended durations to allow oil temperatures to stabilize, lack of bearing oil temperature indication, and simulation of accident conditions (flow, pressure, temperature) of the service water system in a practical manner. (CCE 2001-010)

Revised Control Room Radiological Dose Analysis, Revise and submit radiological dose analysis for the control room, and a license amendment proposal as necessary, by February 1, 2002, to demonstrate continued conformance to the regulatory requirements and the PBNP licensing basis.

Justification for Change: During a call with the NRC staff on February 8, 2001, an extension of the initially proposed May 1, 2001, completion date was discussed. The initial timeline, which was agreed to in a teleconference on April 6, 2000, was based upon the expectation that publication of NEI 99-03, Control Room Habitability Assessment Guidance, would occur by the end of 2000. However, activities within the NRC and the industry to resolve control room habitability issues have been prolonged. This has delayed publication of NEI 99-03 and necessitated extension of the control room habitability initiative.

The commitment date for submitting a radiological dose analysis for the control room and the corresponding license amendment request was changed to February 1, 2002. The revised submittal date allows for receipt of vendor analysis, provides adequate time to finalize the analysis in light of the most current NEI information, and evaluates the need for associated equipment modifications. (CCE 2001-006)

Enhanced Primary-to-Secondary Leakage Monitoring. The plant will maintain a primary-to-secondary leakage-monitoring program that follows industry guidance as defined by EPRI TR-104788-R2, PWR Primary-to-Secondary Leak Guidelines – Revision 2, and future revisions thereto.

Justification for Change: A letter to the NRC dated November 20, 1987, describes the primary-to-secondary leakage-monitoring program. The program was designed to meet the criteria of being able to monitor a tube failure with leakage characteristics similar to that, which occurred at North Anna. The monitoring program addressed concerns dealing with increased rates of leakage, when to evaluate shutting the plant down and alert alarm setpoints for the RMS monitors.

In May 1995, EPRI issued TR-104788, PWR Primary-to-Secondary Leak Guidelines, "based on the latest technical information on tube leaks." The EPRI guideline address management considerations, monitoring methods and equipment, leak rate calculations, operational response, and data evaluation. The Nuclear Energy Institute in NEI 97-06 has adopted this guideline.

This commitment change documents a comprehensive primary-to-secondary leakage monitoring program that is based on the technical requirements contained in the EPRI Guidelines that reflect industry practice, rather than the 7 individual bullets contained in a 1987 letter on the NRC docket.

EPRI TR-104788-R2 fully meets the requirements set forth in the 1987 commitment letter as revised. Adherence to this EPRI Guideline is also required by NEI 97-06. There is no reason to believe that future revisions to the guideline will fall below the thresholds of the commitments. Therefore, the commitment was revised to state that PBNP would implement a comprehensive primary-to-secondary leakage-monitoring program per the requirements of NEI 97-06 and EPRI TR-104788-R2 and the revisions thereto. (CCE 2001-009)

AFW Locked Open Valves and Position Verification, The commitment that all manual valves in the auxiliary feedwater system that could interrupt all auxiliary feedwater flow are locked open and verified monthly was cancelled.

Justification for Change: TS SR 3.7.5.1 require verifications that each manual, power-operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam driven pump, that is not locked, sealed or otherwise secured in position, is in the correct position. The frequency is 31 days. Even though TS does not require verification of locked valves, the Bases justify the intended surveillance and allow cancellation of this commitment.
(CCE 2001-011)

Disposal of Contaminated Sewage Sludge, The contaminated sewage sludge disposal procedures with commitments as documented in the letter dated October 8, 1987, are found to be acceptable, provided that the letter is incorporated into the ODCM by reference and that future commitment changes are documented and reported to the NRC in accordance with established station procedures that incorporate commitment management guidance endorsed by the NRC.

Justification for Change: When the NRC SER was written in 1987, there was no industry-wide guidance for tracking regulatory commitment changes and ensuring these commitment changes are reported, as appropriate, to the NRC. Since that time, NEI developed a succession of guidance documents that have been endorsed by the NRC as meeting NRC requirements. The most recently issued industry guidance is NEI 99-04. The NRC via NRC Office Instruction LIC-100, dated March 2, 2001, Control of Licensing Bases for Operating Reactors endorsed this document. At PBNP, implementation of this guidance is accomplished via use of NP 5.1.7, Regulatory Commitment Management.

The ODCM is a mandated licensing basis document. There should be no difference in the method by which changes to commitments made in our application for contaminated sewage sludge disposal be handled differently from changes that are made to other mandated licensing basis documents. NP 5.1.7 provides guidance for licensing basis and non-licensing basis commitments to ensure NRC is appropriately notified of such changes.

The second part of the commitment change deals retention of the October 8, 1987, application letter and subsequent modifications in the ODCM. It is not desirable that this practice be continued. The original application is of historical interest only and is readily retrievable from the plant's electronic document management system. The individual commitments are maintained in the CLB database, with changes to these commitments being documented in accordance with NP 5.1.7. The method for managing these commitments delineated in the NRC SER does not meet the standards or expectations for rigor in documentation of today.
(CCE 2001-013)