CP&L

Carolina Power & Light Company

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BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
ANNUAL REPORT IN ACCORDANCE WITH 10CFR50.59

Gentlemen:

In accordance with 10CFR50.59, the following annual report is submitted for 1987. This report contains brief functional summaries of procedures and plant modifications which change the description given in the FSAR. It also contains those tests or experiments conducted in 1987 which are not described in the FSAR.

Very truly yours,

J. Alan Bishop for

C. R. Dietz, General Manager
Brunswick Steam Electric Plant

TH/bvc
MSC/87-014

Enclosure

cc: BSEP Resident Inspector Office

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CHANGES TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: OENP-16, Procedure for Administrative Control of Inservice Inspection Activities, ENP-16, Revision 22

DESCRIPTION: Soap bubble testing of tubing connections on CAC-AT-4409 and 4410 each refueling at peak accident pressure.

SAFETY SUMMARY: This method has been approved by the NRC per NLS87-264. These lines are considered extensions of containment and this test ensures leak tight integrity of these lines both each refueling and after repair.

CHANGES TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: OSD-25, Main Steam System, SD-25, Revision 9

DESCRIPTION: Revise SRV capacities to comply with SD-01 and SD-20.

SAFETY SUMMARY: This change will not create any new accident scenarios. This change does not adversely affect the basis to any technical specification. This change is being submitted to revise the SRV capacities listed in SD-25. This revision will change the capacity values to correspond with those listed in SD-01 and SD-20. This change will not increase the probability of occurrence or the consequences of any Chapter 15 analyzed accident.

MSC/88-099A

CHANGES TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: OPT-20.8.1, CAC-AT-4409 Rate Test, PT-20.8.1, Revision 0

DESCRIPTION: This is a soap bubble test of tubing fittings in containment atmosphere monitor CAC-AT-4409 and associated equipment. It is performed per commitment to the NRC as addressed in NLS-87-264.

SAFETY SUMMARY: This system is open post-LOCA and treated as an extension of the containment boundary. This test verifies mechanical connections on the system are leak tight. This is consistent with containment integrity evaluated in the FSAP.

CHANGES TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: OPT-20.8.2, CAC-AT-4410 Leak Test, PT-20.8.2, Revision 0

DESCRIPTION: This is a soap bubble test of tubing fittings in containment atmosphere monitor CAC-AT-4410 and associated equipment. It is performed per commitment to the NRC as addressed in NLS-87-264.

SAFETY SUMMARY: This system is open post-LOCA and treated as an extension of the containment boundary. This test verifies mechanical connecti[illegible] on the system are leak tight. This is consistent with containment integrity [illegible] evaluated in the FSAR.

CHANGES TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: 2APP-UA-03, Annunciator Procedures for Panel UA-03, 2APP UA-03, Revision 9

DESCRIPTION: To correct the data for TBCCW head tank hi-lo alarm to reflect the as-built condition and to revise the SJAE off-gas radiation monitors hi and hi-hi setpoints.

SAFETY SUMMARY: The changes to 2APP UA-03, revision 10, consist of the following:

1. The data for the TBCCW head tank hi-lo level alarm is revised to reflect the as-built condition. This includes maintaining the head tank level using bypass valve MUD-V40 instead of level control valve MUD-LV-547.

   The hi/lo alarm readings are changed from inches to percent to reflect the actual indication by local indicator TCC-LI-3217.

2. The SJAE off-gas radiation monitors hi and hi-hi setpoints have been revised. These setpoints are calculated by E&RC and changes as necessary.

The above changes are in accordance with the systems design as described in the FSAR. These changes reflect as-built condition of the systems and will not impact the FSAR accident analysis results or margin of safety as described in the basis to any technical specification.

MSC/88-099A

CHANGES TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: 0 Emergency Response Plan, Book I, Revision 22

DESCRIPTION: Revise population totals and evacuation times within the 10 miles EPE based on a 1987 study by Itimim Associates. Incorporating new agreement with Southport Fire Department.

SAFETY SUMMARY: This is a change to the BSEP Emergency Plan and does not increase the probability of any accident as evaluated in FSAR, Chapter 15.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-85-114, System Engineering RWCU System Testing and Optimization, SP-85-114, Revision 0

DESCRIPTION: This procedure allows troubleshooting on the isolated section of the RWCU System which is not related whatsoever to nuclear safety. All findings and operations are reviewed prior to and after the effort with a licensed individual, the Shift Foreman.

SAFETY SUMMARY: This procedure does not affect any safety-related equipment either directly or indirectly.

MSC/88-099A

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-002, SP-87-002, Revision 0, Inspection of Anchor Darling Antirotation Devices for Units 1 and 2

DESCRIPTION: Technical upgrade.

SAFETY SUMMARY: Revision was technical upgrade and changes for clarity which will have no affect on any tester evaluation previously made.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-013, Diesel Generator Building Fire Damper Test

DESCRIPTION: This procedure is being done to test the operability of existing fire dampers. The test does not affect safety-related systems in a negative manner.

SAFETY SUMMARY: This test is being performed to verify operability. It does not increase the probability of occurrence or consequences of accidents previously evaluated or of new accidents or create additional exposures to equipment.

This test does not reduce the margin of safety as defined in the basis of any Technical Specification Section 7.1.

MSC/88-099A

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-020, Raychem Splice Inspection, SP-87-020, Revision 0

DESCRIPTION: To provide ANSP for the inspection of installed Raychem splices in BSEP in response to IEN 86-53.

SAFETY SUMMARY: Special procedure is to inspect Raychem splices and would result in increased confidence of satisfactory condition of the splice. Special procedure is to conduct an inspection and does not affect consequences of an accident. The purpose of the inspection described in the special procedure is to verify the qualification of the splices. This will not increase the probability of malfunction. The inspection by this special procedure is not changing equipment but is documenting the condition of the splices to ensure their qualification. This will not increase the consequence of malfunction of equipment. No new accident scenarios will be created by the special procedure and the inspection described therein. The inspection will provide data on the acceptability of the splices. This special procedure does not impact the margin of safety in the basis of the technical specifications. The inspection of splices described by the special procedure is to verify the condition of the splices and would improve safety.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-024, RWCU Recirculation Pump Flow Optimization Procedure, SP-87-024, Revision 0

DESCRIPTION: This procedure increases RWCU flow while monitoring flow and existing brake horsepower on the RWCU pumps.

SAFETY SUMMARY: This test merely increases RWCU flow for monitoring purposes only. This test will not adversely affect any safety-related equipment.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-030, Motor-Operated Valve Actuator Diagnostic Test Special Procedure, SP-87-030, Revision 0

DESCRIPTION: Limitorque valve operators will be tested to determine operator capabilities.

SAFETY SUMMARY: No testing will occur which will cause an increase in probability or severity of an accident. Does not affect technical specification-related functions. Test only gathers information.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-063, Personnel Air Lock Temperature Monitoring, SP-87-063, Revision 0

DESCRIPTION: A RTD will be installed inside the drywell personnel air lock to detect temperature changes during the performance of PT-20.3b.

SAFETY SUMMARY: Procedure will only be used to monitor temperature of personnel air lock while performing PT-20.3b (LLRT).

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-065, Handling and Loading of NUCAC 10-142 Shipping Cask, SP-87-065, Revision 0

DESCRIPTION: The procedure describes the loading of irradiated hardware into a 10-142 cask staged to poolside on 117' elevation in the Reactor Building and the handling of the cask as stated in the Pacific Nuclear Cask Handling Manual.

SAFETY SUMMARY: The cask will not be transferred over fuel or safety equipment nor will it be loaded in the spent fuel pool. Hence, no danger to fuel or safety system exists. Stored fuel will not be handled nor will heavy loads (greater than 1600 lbs) be transported over stored fuel. The reliability of the safety systems will not be reduced. Plant safety systems are unaffected by this procedure. No heavy loads will be transported over safety systems. No heavy loads will be transported over fuel or safety systems. The cask will be staged poolside and not in the spent fuel pool. No margins to safety are decreased. No technical specification requirements are decreased secondary containment will be maintained.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-87-081, Service Water (SW) Pump Motor Cooling Test, Revision 0

DESCRIPTION: Special Procedure SP-87-081 tests selected service water pump motors ability to run at design conditions with the normal motor upper bearing cooling water isolated.

SAFETY SUMMARY: SW pump motor operation will be maintained within design limits (provided by GE) which will be monitored during the test; therefore, the probability of occurrence is unchanged. The consequences of a loss of a SW pump are not affected by this test since the system will be maintained within its design limits during the test. The test procedure requires that the test be terminated prior to exceeding any motor design limits. Therefore, the probability of malfunction is unchanged. This test has no affect on consequences of malfunction since the SW System will be maintained within its design limits during the test. This special test requires operating selected SW pumps under adverse conditions. However, the new higher operating temperature will be maintained below the normal operating limits provided by GE for this motor type. Therefore, the probability of an accident or possibility for malfunction of a different type is not created. The SW System will be maintained within its normal operating limits during this test, therefore, the margin of safety will not be affected.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OEOP-01-ACCP, ATWS Containment Control Procedure, Emergency Operating Procedure, EOP-01-ACCP, Revision 4

DESCRIPTION: Changes to this procedure to reflect as-built per PM 80-133, field revision 136, enhance nuclear safety by improving control of the initial primary containment venting process as required by this procedure. This is achieved through the relocation of 1-CAC-V172 to allow bypassing of the large exhaust piping altogether at the suppression pool which is the more favorable vent path under postaccident conditions since the suppression pool water is utilized as scrubbers for potential gaseous radioactivity. Changes resulting from PM 80-133 also make Unit 1 identical to Unit 2 in respect to the affected valves, isolation overrides, and jumpers required to execute this procedure thus these same changes for Unit 2 have been previously evaluated and approved.

SAFETY SUMMARY: The overriding of the main stack hi-hi rad CAC Group 6 isolation signal by this revision is allowed and supported by the EPG, revision 2, when suppression chamber pressure exceeds 58 psig, thus the consequences of this action have previously been evaluated.

All other changes by this revision are administrative in nature and have no detrimental affect on nuclear safety.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 1SP-87-022, Electrical Operation of RCIC Vacuum Breaker 1-E51-F062 Against its Maximum Calculated dP, SP-87-022

DESCRIPTION: Vacuum breaker 1-E51-F062 will be closed against its maximum expected differential pressure to confirm that it is capable of operating properly under worst case conditions.

SAFETY SUMMARY: This procedure will verify, under actual pressure conditions, that valve 1-E51-F062 capable of operating as designed against its maximum expected differential pressure. The plant will be in a shutdown condition throughout the test.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 1SP-87-025, Operation of RHR Service Water Booster Pumps 1B and 1D With Temporary Pump Motor Cooling Water Installed and Contingent Plans if Conventional Discharge Header Freeze Seal Fails

DESCRIPTION: Provide temporary instruction for operating 1B and 1D motor coolers and provide contingency plans if freeze seal fails.

SAFETY SUMMARY: RHR service water booster pumps are not required for accident analysis and no credit is taken for them in Chapter 15 of the FSAR. Credit is not taken for the RHR SW pumps in the accident analysis evaluated in Chapter 15 of the FSAR. RHR SW booster pumps are not evaluated for safety concerns per the FSAR. Consequences of malfunction of RHR SW booster pumps as related to safety per the FSAR will not be increased. Credit is not taken for RHR SW booster pumps in FSAR analysis. Service water is still available.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 1SP-87-028, Electrical Closure of HPCI CST Suction Valve Against Maximum Calculated dP

DESCRIPTION: HPCI suction valve 1-E41-F004 will be closed against its maximum expected differential pressure to confirm that it is capable of operating properly under worst-case conditions.

SAFETY SUMMARY: This procedure will verify, under actual pressure conditions, that valve 1-E41-F004 can operate as designed against its maximum expected dP. The plant will be in a shutdown condition during this test.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 1SP-87-055, Eddy Current Test Procedure on the 1B RHR Heat Exchanger
SP-87-055, Revision 1

DESCRIPTION: This procedure merely provides procedural guidance for performing Eddy current testing of the 1B RHR heat exchanger.

SAFETY SUMMARY: Conducting an Eddy current test on the 1B RHR heat exchanger will not adversely affect nuclear safety. The actual testing does not perform a safety-related function.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 1SP-87-059, HPCI System Vessel Injection Test SP-87-059, Revision 0

DESCRIPTION: The purpose of the HPCI System vessel injection test is to demonstrate the reliability and performance of the system.

The test involves a full flow functional test of the HPCI System while injecting into the reactor vessel and operating in Test Condition 3. HPCI pump suction will be via the CST. The HPCI System will be started using a simulated automatic initiation signal transmitted by test jack, E41A-J1, located in Panel H12-P620. The response time from the simulated automatic initiation signal until the HPCI System delivers a flow rate equal to or greater than 4250 gpm at rated pump discharge pressure will be determined.

Step changes in the HPCI injection rate will then be made by adjusting the HPCI flow controller setpoint tape setting to determine the response characteristics of the flow control loop. Adjustments to the controller will be performed, if required, to improve system stability.

The HPCI System will then be shut down and, after a period of approximately fifteen to thirty minutes, another simulated automatic initiation signal will be established to restart the HPCI System. System response time and flow controller response characteristics will again be determined. Critical parameters associated with operation of the HPCI System will be monitored and traced to establish a permanent record baseline using the ERFIS time-history plot analysis routine.

A similar test was performed on both Units 1 and 2 during initial plant startup testing. This test, SU-15, High Pressure Coolant Injection System, was performed in Test Condition 3 and involved injection to the reactor vessel. The startup test is discussed in the FSAR. Inadvertent HPCI injection is also discussed in the FSAR. The FSAR states that during inadvertent HPCI injection at power, the operator can control any reactor power changes in the normal manner of power control. A prerequisite to performing the test is to verify that Operations shift personnel have thoroughly reviewed the test procedure and this safety evaluation.

The following concerns have been considered in developing the HPCI test procedure.

Vessel Water Level Transient

HPCI vessel injection will result in an initial increase in reactor water level due to feedwater controller response time. In order to compensate for this transient, the vessel water level will be slightly lowered prior to HPCI injection in order to anticipate the initial reactor water level increase. In addition, in order to more effectively maintain reactor water level during HPCI injection, the feedwater flow controller will be switched from three element control to one element control prior to HPCI initiation. In three element control, during HPCI vessel injection, the feedwater flow controller would receive erroneous information since HPCI flow enters the feedwater line downstream of the feedwater flow element/transmitter, and steam is diverted to the HPCI turbine from main steam line A upstream of the main steam line flow element/transmitter. Single element control will provide the feedwater flow controller with reactor water level information only. Vessel level will therefore be more closely maintained to the controller setpoint tape setting. Consequently, the probability of a reactor scram due to reactor water level is minimal.

NOTE: HPCI vessel injection during SU-15 was also performed with the feedwater flow controller in single-element control.

Vessel Pressure Transient

The GE Supplemental Reload Licensing Submittal for BSEP Unit 1, Reload 5, predicts an increase in reactor pressure of less than ten psig due to inadvertent activation of HPCI at full power. Since the HPCI test will be performed at reduced power, the pressure increase will be even less. Representative pressure transient traces from startup test SU-15 show no discernable increase to reactor pressure during HPCI injection in Test Condition 3. Therefore, based on the above, the integrity of the nuclear system process barrier is not threatened by high internal pressure and the probability of a reactor scram due to high reactor pressure is unlikely.

Reactor Water Chemistry Transient

A prerequisite to the HPCI test is to verify that the CST (source of HPCI pump suction) water quality meets the limits of AI-81, Table 10, Action Level 0. Water meeting this requirement will not result in a significant water chemistry transient during HPCI vessel injection. In the unlikely event that HPCI pump suction is automatically transferred from the CST to the suppression pool, due to low CST water level or high suppression pool water level, HPCI injection to the vessel will be secured.

Feedwater Nozzle Thermal Transient

HPCI injection to the vessel will cause feedwater nozzle thermal transients. Feedwater nozzle thermal cycling has been previously analyzed in the FSAR with respect to loss of feedwater heating. Thermal transients occurring as a result of the HPCI test will be monitored and recorded by newly installed temperature monitoring instrumentation. Records of these transients will be available for incorporation into the next feedwater nozzle thermal cycling analysis.

CST Water Level

The CST is the preferred source of pump suction for the HPCI and RCIC Systems. The minimum CST capacity required for the HPCI and RCIC Systems is 100,000 gallons. This capacity corresponds to a CST water level of approximately 9 feet. A prerequisite to the HPCI test is to verify that the CST water level is greater than 25 feet. A limitation included in the test states that if the CST water level decreases below 15 feet, HPCI injection to the vessel shall be secured. An adequate margin is, therefore, maintained over the minimum CST capacity required in the event that either the HPCI or RCIC System receives an actual initiation signal during or after the HPCI test. In addition, either system remains capable of taking suction from the suppression pool.

Condenser Hotwell Water Level

Water level in the condenser hotwell will increase during HPCI vessel injection since feedwater suction from the hotwell will be decreased to compensate for water injected to the vessel from the CST. The HPCI test includes a precaution to monitor hotwell level and transfer water to the CST as required to maintain hotwell level and condenser vacuum.

Suppression Pool Water Level and Temperature

The HPCI test includes a prerequisite to ensure that the suppression pool water level is initially lower than the level resulting in a technical specification action statement. Since the durations of the test injections are expected to be relatively short, an increase in suppression pool water level to the technical specification action level is unlikely. However, should this level be approached, a precaution is included in the test to transfer water from the suppression pool to Radwaste. This precaution is in addition to the requirement for a second operator to be present to monitor suppression pool level during HPCI System operation. Precautions are also included to monitor suppression pool temperature at five-minute intervals during HPCI System operation and to secure the system if the average suppression pool temperature reaches the technical specification limit of 105°F.

Reactor Feedwater Flow

HPCI vessel injection will result in a decrease in feedwater flow equal to HPCI pump flow (approximately 4250 gpm). This decrease will be initiated and controlled by the feedwater flow controller responding to an increase in reactor water level caused by HPCI initiation.

At the minimum recommended test power of 65 percent, total feedwater flow without HPCI injection will equal approximately 16,700 gpm. During HPCI vessel injection, total feedwater flow will be reduced to approximately 16,700 - 4,250 = 12,450 gpm. Each reactor feed pump will deliver half this flow rate or 6,225 gpm. The reactor feed pump recirculation valves do not open until the corresponding reactor feed pump flow decrease to between 2,955 and 3,045 gpm. Therefore, the probability of a reactor feed pump entering the recirculation mode of operation is minimal. Both reactor feed pumps should continue to supply water to the vessel at the above reduced flow rates.

In addition, an adequate margin exists over the minimum feedwater flow rates at which the reactor recirculation pumps run back to 45 percent speed. The feedwater flow constraint of reactor recirculation pump speed limiter No. 1 limits pump speed if feedwater flow is less than 20 percent of rated flow, which equals approximately 5,150 gpm. Feedwater flow rate during HPCI injection at 65 percent power will equal approximately 12,450 gpm. Adequate margin is therefore provided against a reactor recirculation pump runback initiated by speed limiter No. 1. The feedwater flow constraint of speed limiter No. 2 limits pump speed if the flow from any individual reactor feed pump is reduced to 20 percent of its rated flow, coincident with reactor water level below the 182" alarm point. Individual reactor feed pump flow has been estimated to be 6,225 gpm which corresponds to 48 percent of rated pump flow. Adequate margin is therefore provided against a reactor recirculation pump runback initiated by speed limiter No. 2.

Power Flux Transient

An increase in reactor power will occur during HPCI vessel injection due to higher core inlet subcooling. The HPCI test will be performed in Test Condition 3 which specifies a minimum reactor recirculation flow of 80 percent. Furthermore, the test recommends that reactor power be between 65 and 70 percent. The maximum expected power increase due to HPCI vessel injection at the above conditions is 11.4 percent. (Refer to attached memo NF-87-434.) The APRM neutron upscale trip setpoint is 116 percent. Since the maximum expected power during HPCI vessel injection is 70 + 11.4 = 81.4 percent, an adequate margin to the scram setpoint is maintained. The APRM thermal upscale trip setpoint is 0.66W + 50%, where W = total recirculation flow in percent. Based on the minimum recirculation flow of 80 percent, the thermal trip setpoint equals 103 percent. An adequate margin to this scram setpoint is also maintained.

From a plant safety standpoint, should power levels of the reactor rise above that expected based on previous tests and analyses, the neutron monitoring trips will scram the reactor, thus performing their intended function.

Fuel Thermal Limits

The CP&L Nuclear Fuel Section has developed recommendations for initial conditions to the HPCI test. Refer to attached memo NF-87-434. These recommendations will be verified to have been met prior to the start of the test by on-site Nuclear Engineering. These recommendations provide adequate margins to the fuel thermal limits to minimize the possibility of exceeding any limit. However, if at any time during the HPCI test Nuclear Engineering determines that a thermal limit is being encroached or has been exceeded, HPCI injection will be terminated or other appropriate action will be taken. The condition of the reactor core (power distribution/thermal margins) will be monitored by the Nuclear Engineering staff throughout the HPCI test.

Connection of ECCS Test Switch to ERFIS Point

In order to provide ERFIS with a HPCI initiation signal, a temporary connection will be made from the ECCS test switch to one of the ERFIS MUX cabinets. The connection of this high impedance test device to the HPCI System will not affect accident probabilities or consequences because the failures discussed below would result in conditions bounded by existing FSAR analyses.

1. Either Conductor Shorting to Ground

   Because the dc power system at BSEP is floating, the HPCI System may continue to function normally should either conductor short to ground. The worst case failure is a blown fuse during the injection portion of the test, which would leave the system lined up for vessel injection. Vessel injection would continue and an annunciator for A logic bus power failure would alarm.

2. Conductors Shorting to Each Other

   This failure would result in the initiation from high drywell pressure to be sealed in and unresettable. The system would continue to operate and would be bounded by the positive reactivity insertion transient discussed in the FSAR.

This connection does not affect any environmental conditions, reduce redundancy, violate active failure criteria, nor increase common mode failure probabilities; therefore, it is acceptable to make the test connection on a temporary basis.

SAFETY SUMMARY: This test involves a deliberate automatic initiation of the HPCI System while operating at reduced power. Adequate margins to fuel thermal limits and scram setpoints are established prior to HPCI initiation. This test does not render the HPCI System inoperable nor does it decrease the capability of the HPCI System or any other safety-related system to perform its design function.

Inadvertent HPCI initiation at full power is a condition analyzed in the Supplemental Licensing Submittal for Unit 1, Reload 5. It is not the limiting transient at full power, and its consequences are further reduced by initiating the test at reduced power. The control rod pattern to be established prior to the test will provide adequate margin to the fuel thermal limits.

The HPCI test will not subject the system to conditions outside the system design base. Operation of the HPCI System will not compromise the ability of any other safety-related system to perform its design function.

This test does not render the HPCI System inoperable. If the HPCI System receives an automatic initiation signal from either low reactor water level or high drywell pressure during a test injection, all associated components would simply receive another signal to lineup for vessel injection, and vessel injection would continue as required.

The HPCI test will not change the method in which any safety-related equipment achieves its safety function.

Simulations summarized in Nuclear Fuel Section memorandum NF-87-434 indicated that the margins to fuel thermal limits remain adequate and that nonbarrier fuel is predicted to remain below the preconditioning threshold. Adequate precautions are included in the procedure to ensure that the test will be secured before any technical specification margin of safety is reduced.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 1SP-87-061, Special Procedure for Tube Leaks in Feedwater Heaters 3A, 3B, 4A, 4B, 5A, and 5B (for Unit 1 only), SP-87-061, Revision 0

DESCRIPTION: Isolate and perform a low-pressure pneumatic leak test on the shell of feedwater heaters 3A, 3B, 4A, 4B, 5A, and 5B in order to perform a soap bubble inspection on the heat exchanger tubes to identify tube-to-shell throughwall leaks for plugging.

SAFETY SUMMARY: Feedwater tubes are routinely plugged when throughwall leakage is identified. This test is used to identify leaks for plugging to increase efficiency and minimize further damage to adjacent tubes. This test does not affect the margin of safety in the technical specifications. This test does not affect any previously analyzed accident in the FSAR and does not affect safety-related equipment.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-87-019, Adjustment/Repair of the Uninterruptible Power Supply Static Transfer Switch, SP-87-019, Revision 0

DESCRIPTION: The UPS System is designed to maintain the quality and continuity of 120 Vac power to critical monitoring and control loads in the plant. These are nonsafety-related loads which require a source of uninterruptible power in order to maintain sustained plant operation during normal transients. In addition, this system is not required for safe shutdown of the reactor or for actuation, monitoring, or operation of any of the ECCS Systems.

This special procedure involves bypassing the UPS static transfer switch while the reserve (hard ac) source is supplying the UPS loads. This allows maintenance to be performed on the static switch or power converters without interrupting the supply of power to the UPS loads. A lamp bank load will be utilized to troubleshoot problems with the static switch.

SAFETY SUMMARY: Bypassing of the UPS static switch with the reserve source supplying the UPS loads (and the subsequent return of the system to its normal status) is covered under Infrequent Operations in Operating Procedure OP-52, Sections 8.5 and 8.6, respectively. Therefore, this special procedure serves as guidance for the use of the lamp bank load that will be used in the maintenance/repair of the static transfer switch. Since the static switch and the inverters will be bypassed during the completion of this work, the probability of occurrence or the consequences of an accident or malfunction of equipment will not be increased. Similarly, the margin of safety, as defined in technical specifications, will not be reduced in any way.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-87-019, Adjustment/Repair of the Uninterruptible Power Supply Static Transfer Switch, SP-87-019, Revision 1

DESCRIPTION: To correct typographical errors in Steps 6.2.2 and 6.2.5 (references to previous steps were incorrect).

SAFETY SUMMARY: The UPS System is designed to maintain the quality and continuity of 120 Vac power to critical monitoring and control loads in the plant. These are nonsafety-related loads which require a source of uninterruptible power in order to maintain sustained plant operation during normal transients. In addition, this system is not required for safe shutdown of the reactor or for actuation, monitoring, or operation of any of the ECCS Systems.

This special procedure involves bypassing the UPS static transfer switch while the reserve (hard ac) source is supplying the UPS loads. This allows maintenance to be performed on the static switch or power converters without interrupting the supply of power to the UPS loads. A lamp bank load will be utilized to troubleshoot problems with the static switch.

Bypassing the UPS static switch with the reserve source supplying the UPS loads (and the subsequent return of the system to its normal status) is covered under Infrequent Operations in Operating Procedure OP-52, Sections 8.5 and 8.6, respectively. Therefore, this special procedure serves as guidance for the use of the lamp bank load that will be used in the maintenance/repair of the static transfer switch. Since the static switch and the inverters will be bypassed during the completion of this work, the probability of occurrence or the consequence of an accident or malfunction of equipment will not be increased. Similarly, the margin of safety, as defined in technical specifications, will not be reduced in any way.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-87-027, Filling and Venting the RWCU System Using DW From the F/D Precoat Cycle SP-87-027, Revision 0

DESCRIPTION: This document is a special procedure used to allow an alternate RWCU fill path while the reactor is at power.

SAFETY SUMMARY: The RWCU, while this procedure is being performed, will be isolated. No safety-related or technical specification-related equipment will be adversely affected. No new accident scenarios are introduced with this procedure.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-86-056, Transfer of Spent Fuel Pool Filters Into High Integrity Containers/HICs, SP-86-056, Revision 0

DESCRIPTION: Low dose vacuum filters will be drained of water and transferred from the spent fuel pool to a high integrity container inside a waste shipping cask.

SAFETY SUMMARY: The cask transport will follow safe load paths. No stored fuel or safety equipment will be passed. Stored fuel will not be moved per this procedure. The cask transport will follow safe load paths. Safety equipment will not be affected by this procedure. Secondary containment will be maintained per this procedure. No safety system will be isolated per this procedure. Stored fuel is not moved per this procedure. Cask transport will follow safe load paths. Loads greater than 1600 pounds will not be transported over stored fuel. This procedure does not direct any work which would endanger the integrity of the spent fuel pool liner.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-86-081, SP-86-081, Hydrogen Water Chemistry Minitest, Revision 2

DESCRIPTION: The hydrogen water chemistry minitest is being performed to determine the feasibility for permanent installation of a hydrogen injection system and the effectiveness of hydrogen water chemistry in the mitigation of IGSCC in recirculation piping at Brunswick. The test consists of using temporary vendor-supplied equipment for injection of hydrogen-to-reactor feedwater, injection of oxygen to plant condensate and off-gas systems, and for monitoring of associated plant water chemistry and materials parameters. The procedure involves the addition of hydrogen to the reactor primary coolant at increasing increments over a range of approximately 0-70 scfm. As a result, the radiolysis of water is suppressed thereby lowering the free oxygen concentration in the reactor coolant. The reduction in oxygen eliminates one of the necessary causative agents of IGSCC of stainless steel piping. In addition, oxygen will be injected to the condenser air removal system upstream of the recombiners to ensure recombination of the test hydrogen. Oxygen will also be added in small amounts to the condensate system to determine the amount necessary to maintain the protective oxide film present in the carbon steel portion of this system. The test is to last for approximately one week. All equipment installed by this special procedure is temporary and will be removed from the plant at the conclusion of the test. The various test equipment systems and their effect on plant systems and plant equipment are discussed separately below.

SAFETY SUMMARY: Hydrogen Storage and Distribution System

Compressed hydrogen is to be supplied to the plant site in gaseous form via tube-tank trucks. During the test, the trucks will be located per Figure 4 at the southeast corner of the Unit 2 Reactor Building. The location meets or exceeds all the criteria of NFPA Code Number 50A, Gaseous Hydrogen Systems at Consumer Sites, Section 5.0, which delineates storage tank siting criteria to ensure plant and personnel safety are maintained. In addition, the area will be roped off and posted to prevent vehicular and pedestrian traffic and also to prevent smoking or other potential ignition sources from entering the area. In addition to the NFPA code, the EPRI document, Guidelines for Permanent BWR Hydrogen Water Chemistry Installations, was consulted for siting criteria. Three additional siting characteristics are imposed: 1) The minimum permissible distance of storage vessels from safety-related structure; 2) The minimum permissible distance of vessel attached piping from safety-related air intakes; and 3) The route of hydrogen delivery on site.

(It should be noted that the EPRI document is intended for permanent installations and as such many of the criteria contained in it are not practical nor applicable to temporary installations. The design of the system considered all criteria discussed in the document; however, credit for full compliance is only taken for Section 4.1, Safety Considerations, Gaseous Hydrogen. Refer to Attachment I.)

1. The tube-tank trailers consist of ten vessels of approximately 75 ft^3 each. While in transit, and after permanently locating the trucks, a distance of at least 60 ft will be maintained between the truck and all safety-related structures (i.e., the Reactor Building). Per Figure 4-3 of the EPRI document, at a distance of 60 ft, each hydrogen vessel shall be limited to 10,800 scf. This will be accomplished by limiting vessel pressure to 2000 psig.

   P1(V1) = P2(V2)
   14.7 psig (10,800 ft^3) = P2 (75 ft^3)
   P2 = 2116 psia

   NOTE: Vessel dimensions are: 34' -4" lg x 22" dia x 1" wall

2. The largest diameter vessel attached piping upstream of the excess flow protection is 3/4". Per the EPRI guidelines, Figure 4-4, 160 ft separation from the nearest safety-related air intake is required. The nearest safety-related air intake is for diesel generator 4 and, per Figure 4, is more than 160 ft away.

3. Figure 8 shows the delivery route of the hydrogen while on site. The 60 ft separation distance is maintained. During delivery, the hydrogen trucks will maintain a minimum distance of 300 ft from the chlorine tank car located in the northeast section of the site and will be approximately 420 ft from the chlorine car in their permanent test position. Based on the large separation distances and the controlled delivery route of the truck, there is no potential for a common-cause failure to result in an explosive hydrogen/chlorine chemical reaction. Due to the temporary nature of the hydrogen truck siting, no fences, or crash barriers will be erected.

   The hydrogen flow control and distribution system consists of three separate modules; the H_2 supply station, the H_2 flow control station, and the H_2 injection station. The three modules are connected by 1/2" diameter neoprene hose with flexible stainless steel jacketing. The hose is supplied in 30 ft lengths and is connected using flared and compression fittings. The hose and distribution system have been cleaned for oxygen service in accordance with Compressed Gas Association (CGA) guidelines and are, therefore, also suitable for hydrogen service. The distribution hose will be pneumatically pressure tested with helium gas prior to use. The hose is rated at 3000 psig.

   The H_2 supply station will consist of a valve manifold with connections for up to two hydrogen trucks and one nitrogen source (for purging). The hydrogen pressure regulator will reduce supply pressure to 350 psig for distribution to the H_2 flow control station. The supply station also contains an automatic isolation valve which will actuate on high H_2 flow (approximately 150 scfm), high line pressure (650 psig), or by depressing the STOP button located at the H_2 flow control station. The isolation valve is air-operated and will fail to the closed position on loss of air or loss of power. The supply station also contains a relief valve set at 750 psig. The supply station is located at the hydrogen truck site. The supply hose will be routed from the truck location to the south end of the Unit 2 breezeway by means of temporary supports erected along the south wall of the Unit 2 Reactor Building. The hose will be run overhead to the extent possible and will be roped off and posted to limit personnel access. The service road along the south side of the Unit 2 Reactor Building will be closed to all vehicle traffic while the system is filled with hydrogen.

A second automatic isolation valve is located just outside the breezeway entrance and will actuate on high injection line pressure (450 psig), low injection line pressure (60 psig), upon depressing a second STOP button located at the flow control station, or upon receipt of a signal from any one of seven H_2 leakage monitors located inside the breezeway and condensate booster pump rooms. The detectors are located at potential leakage sites such as piping connections and in overhead areas where H_2 gas might collect. The detectors will isolate flow at approximately 2% H_2 concentration.

The H_2 flow control station is located inside the breezeway at the entrance to the condensate booster pump room. A second pressure regulator reduces system pressure to approximately 220 psig for injection to the condensate system. A second relief valve is located downstream of the pressure regulator to prevent overpressurization of this portion of the system. The relief valve discharge is routed outdoors. Injection flow will be controlled from this location and can be quickly stopped by depressing either of two STOP buttons located on the flow control panel. This station will be continuously manned throughout the test and will have direct communication with the Control Room. Manual isolation of injection by depression of one of the two STOP buttons will occur by verbal command from the Control Room on any of the following:

- Feedwater H_2 concentration less than 50% of expected
- Recombiner inlet pressure high (7 psig) UA-44/45, 5-1
- Recombiner surface temperature high-high UA-44/45, 4-2
- Loss of one (or both) SJAE
- Reactor trip
- Loss of oxygen injection
- Off-gas flow 20% higher or 10 scfm higher than expected (whichever is greater)
- Off-gas H_2 concentration 2% increasing
- Request of Shift Foreman or Control Operator

The hydrogen injection station is located inside the condensate booster pump room and consists of three 1/2" hoses routed to each of the three condensate boster pump suction lines. Each hose is furnished with a manual isolation valve and a check valve and is connected to the existing 3/4" drain line just upstream of the pump suction. Permanent plant valves in each of these drain lines will provide for isolation during injection hose installation and removal or as required. Injection will occur into only the two operating pumps. The pump vibration will be monitoed at each H_2 flow rate to ensure no effect on pump operation. Injection flow will be decreased or terminated if abnormal vibration occurs. The volume flow of hydrogen will be insignificant compared to the feedwater flow, so abnormal vibration is not expected. At the highest expected flow and lowest possible condensate system pressure, the ratio of feedwater to hydrogen by volume will be approximately 350:1, and by mass will be approximately 3 ppm.

The hydrogen system design has been reviewed with respect to Branch Technical Position CMEB 9.5.1 (NUREG 0800), Section C.5.d, Control of Combustibles. The hydrogen system is located primarily outdoors and does not pass through any safety-related areas with the exception of the south breezeway, which contains some safety-related cables. This is permitted per paragraph 5), as the hydrogen lines are equipped with automatic isolation upon pipe break to prevent explosive levels from being obtained. Under worst case pipe break conditons, less than 10 cu-ft of H_2 would be discharged into the south end of the breezeway between the entrance and the H_2 injection station. This area is well ventilated and has a volume of approximately 20,000 cu-ft. Although local concentrations at the pipe break might be initially higher, the volume ratio indicates a diffusion concentration of 0.05%, assuming no ventilation. This is well below the 4% lower exposive limit (LEL) of hydrogen in air. The flexible hose minimizes the effect of seismic events and the automatic shutoff valves provide for compliance with Section C.5.d, (5). The use of neoprene lined hose was required due to lack of a suitable substitute, however, use of this material is not prohibited. The system design is in accordance with Branch Technical Position CMEB 9.5.1, Section C.5.d.

2. Oxygen Storage and Distribution System

   Compressed oxygen will be supplied to the plant site in gaseous form via tube-tank trucks. During the test the trucks will be located per Figure 5 in the southwest corner of the protected area near the entrance to the transformer yard. The location meets or exceeds all of the criteria of NFPA Code Number 50, Bulk Oxygen Systems at Consumer Sites, Sections 4 and 5, which delineate storage tank siting criteria to ensure plant personnel safety is maintained. The area will be roped off and posted to prevent vehicular and pedestrian traffic and also to prevent smoking or other ignition sources from entering the area.

The oxygen flow control and distribution system consists of four separate modules; the O_2 supply station, the O_2 flow control station, the condensate O_2 injection station, and the off-gas O_2 injection station. The modules are connected by a 1/2" diameter neoprene hose with stainless steel jacketing. The hose is supplied in 30 ft lengths and is connected using flared and compression fittings. The hose and distribution system have been cleaned for oxygen service in accordance with CGA guidelines and will be pneumatically pressure tested with nitrogen gas prior to use. The distribution hose is rated at 3000 psig.

The O_2 supply station will consist of an oxygen truck, a valve manifold with connections for two oxygen trucks and a nitrogen source for testing, an oxygen pressure regulation, and a relief valve. The oxygen regulator will regulate supply pressure to 450 psig for distribution to the O_2 flow control station. The supply station relief valve protects the distribution system from regulator failure and is set at 650 psig. The supply station also contains an automatic isolation valve which will isolate on high O_2 flow (approximately 75 scfm), high line pressure (550 psig), or by depressing the STOP button located at the O_2 flow control station. The isolation valve is air-operated and will fail to the closed position on loss of air or loss of power.

The supply hose will be routed across the transformer yard to the west entrance of the Unit 2 Turbine Building. The hose will be routed overhead to the extent possible, and to maintain maximum distance from the electrical equipment in the area. The area will be closed to vehicle traffic and posted while the system is filled with oxygen.

The hose is to be routed into the Turbine Building at the west entrance and due east to the condensate pump pit area where the O_2 flow control station is located. The hose will again be routed overhead to the extent possible. At the O_2 flow control station, a second pressure regulator is available for additional pressure reductions. A second relief valve provides overpressure protection downstream of the regulator. The discharge of the relief valve is routed outdoors. Downstream of the relief valve, flow is divided into off-gas injection flow and condensate injection flow. Oxygen will be injected into the off-gas system upstream of the recombiner vessels to provide for recombination of the test injected hydrogen. Oxygen will also be injected to the condensate system at the suction of the condensate pumps on a trial basis to determine the amount of oxygen required to maintain the protective oxide film in the carbon steel portion of the condensate system.

Inside the Turbine Building, the oxygen line passes within approximately 20 ft of the Unit 2 high voltage switchgear and the hydrogen seal oil coolers. None of the applicable codes prohibits this proximity, however, as conservative measure the oxygen line will be checked frequently for leakage. During the test, following each increase in flow, the portion of the oxygen line located inside the Turbine Building will be checked for leakage with the use of a hand-held oxygen detector. If O_2 concentrations of 25% or more are achieved, the hydrogen and oxygen flow will be stopped until the leakage is corrected. During the test, the oxygen flow control station will be provided with telephone communication to the Control Room. Whenever possible, O_2 will be manually isolated only after H_2 injection has first been isolated and off-gas flow rates begin to increase. This will ensure that all test hydrogen has completed its pass through the recombiner. The station will be manned in conjunction with the Turbine Building sample station, which is nearby and within hearing range of the telephone at the O_2 flow control station.

3. Off-Gas System

With hydrogen addition to the feedwater system, the radiolysis of water in the reactor vessel is suppressed and the net production of hydrogen and oxygen decreases. Figure 6 represents the <u>approximate</u> expected hydrogen flow rates at Brunswick based on test results from the Swedish reactor at Forsmark. The hydrogen flow to the recombiner initially decreases and passes through a minimum. At higher addition rates, the hydrogen flow to the recombiner increases and assymptotically approaches the addition rate line. At the recombiners, the radiolytcally produced hydrogen and oxygen are present in stoichiometrically equivalent amounts. The added hydrogen is matched by addition of a stoichiometrically equivalent amount of oxygen upstream of the recombiner. The net result is <u>no change</u> in off-gas flow downstream of the recombiner. Significant changes in off-gas flow during the test could indicate a recombiner problem, and test injection will be terminated. A significant change shall be defined as an unexplained increase of 20% or more or at the discretion of the Shift Foreman or Test Engineer. Recombiner bed temperatures will be monitored during the test and should have the same approximate shape as the net hydrogen flow to the recombiner in Figure 6. To ensure recombiner design conditions are not exceeded, hydrogen addition will be decreased or terminated should a recombiner high-high temperature alarm be received. Each recombiner train is capable of recombining up to 116 scfm hydrogen. Since addition rates will not approach this value, overheating of the recombiner beds is not expected. Hydrogen concentrations at the inlet to the recombiner must be maintained below the LEL of 4% H_2 by volume. During the test, both recombiner trains will be in service at 50% load. The 50% load dilution steam flow of 3875 lbs/hr per train permits up to 58 scfm H_2 flow at the inlet of each recombiner before achieveing a 4% concentration. This number does not include condenser air-inleakage which will act to dilute H_2 concentration even further. Since the combination of radiolytic and added hydrogen will always be lower than 116 scfm (2 x 58 scfm), explosive concentrations will not be achieved (Reference Figure 6). The monitoring of recombiner temperatures will ensure condenser gas removal rates are as expected. Significant deviations from expected temperatures (i.e., a steady immediate increase in temperature versus the expected initial decrease) will be readily observed. In conclusion, for an H_2 concentration of over 4% to be achieved at the recombiner inlet, the HWC concept of suppression of radiolysis of water would have to fail to occur. This would of course be detected before H_2 flow rates above 116 scfm would be achieved. It would result in no change in recirculating water dissolved oxygen, a dramatic increase in dissolved hydrogen, and no change in ECP. This has not occured in any HWC tests to date. Explosive concentrations at the recombiner inlets do not represent a concern which requires specific monitoring.

4. Radiation Effects

Hydrogen water chemistry will result in an increase in site radiation levels due to the increased carry-over of Nitrogen - 16. This occurs as oxygen concentrations are reduced and the nitrogen undergoes a change in form from soluble (nitrates and nitrites) to volatile (ammonia). As a result, a great portion of the test procedure is devoted to the monitoring of these changes in radiation levels. Qualified Health Physics personnel will be utilized during the test via Special Procedure SP-86-096 to perform surveys and limit personnel access to high radiation areas in accordance with approved site procedures.

The location of the Unit 2 Reactor Building ventilation effluent rad monitor and the Control Building ventilation intake rad monitor have been evaluated for effects from increased N-16 activity during the test. Both are located such that sufficient shielding is provided between the detectors and expected N-16 radiation sources. Spurious isolations of Reactor Building or Control Building HVAC Systems are not expected. Turbine Building area radiation monitors (ARM) are expected to detect increases in radiation levels and many are likely to alarm; however, these monitors do not provide any safety-related interlocks and their actuation is of no consequence.

The main steam line radiation monitors trip-point will be set higher than normal at the start of the test to account for the higher than normal N-16 activity expected in the main steam. The adjustment of the MSLRM setpoint will be performed in accordance with Unit 2 technical specifications (Reference TSC- 86-TSB-11).

The off-gas process effluent rad monitors setpoints already contain enough margin to allow for the expected increase in activity. These rad monitors will be read periodically throughout the test to ensure that the setpoints are not inadvertently exceeded.

5. Effects on Other Plant Systems

The hydrogen and oxygen injection systems are not connected to, nor do they interface directly with any plant safety-related systems. However, the changes in primary coolant hydrogen and oxygen concentrations must be evaluated for potential effects on other plant systems. The addition of hydrogen to the plant feedwater system will result in increased dissolved H_2 concentrations prior to the point of entry into the reactor vessel. After entry to the vessel, dissolved H_2 concentrations vary in accordance with the flow rates in Figure 6. Feedwater system equipment drains and leak-offs downstream of the injection point will contain water with progresively higher concentrations of H_2. Depressurization of the water as it collects in drains and sumps could result in some release of H_2 gas. The three possible collection points are; 1) the Turbine Building equipment drain sump, 2) the drywell equipment drain sump, and 3) the Reactor Building equipment drain tank. The first two locations are open sumps and therefore the accumulation of H_2 gas in explosive concentrations is not expected.

A. The Turbine Building equipment drain sump is locat d at the -3 elevation in the north-south pipe tunnel. The area directly over the sump will be monitored periodically during the test to ensure no accumulation of H_2 occurs.

B. The drywell equipment sump is open to the drywell atmosphere and any collection of gas would be diffussed into the drywell which is inerted with N_2 gas and monitored continuously for both hydrogen and oxygen. No additional monitoring is required.

C. The Reactor Building equipment drain tank is continuously vented to the Reactor Building exhaust ventilation system. No additional monitoring is required.

The condensate booster pumps and feedwater pumps will be monitored for effects of H_2 injection. Baseline vibration readings using hand-held probes will be obtained on the condensate booster pumps prior to injection and additional readings will be taken at each H_2 flow rate. Excessive vibration (as defined by the Test Engineer or Shift Foreman) will result in decreasing or terminating injection. The feedwater pumps will be monitored before and during the test from the permanently installed vibration monitors with readouts located in the breezeway.

The changes in dissolved gas concentrations will have no effect on core physics. The ratio of these gases compared to overall core flow is insignificant.

The effect on fuel rod materials and integrity will also be insignificant. Inspection of fuel bundle surface films was performed at Dresden 2 after 18 months of near-continuous HWC operation. Only small changes in surface chemistry and crud deposition were observed. (Reference GE NEDO-31163, November 1985.) HWC operation at Brunswick for a one-week period will have no effect on fuel bundle integrity.

RCI-03.1, Section 7

The test equipment has been designed, located, and tested to ensure that any possible failure of the test equipment before, during, or after the test will have no effect on any plant safety-related equipment. The probability of any accident previously evaluated is not increased by the performance of this test. The test equipment does not effect the design or operating parameters of any safety-related plant equipment. The implementation of the testing procedure does not alter any plant operating parameters other than to affect small changes in primary coolant water chemistry. The consequences of any accident previously evaluted will not be increased. The consequences of any equipment malfunction of equipment important to safety will not be increased.

The performance of this test does not introduce any new or different accident possibilities from those previously evaluated in the FSAR. Although the test introduces to the site potentially hazardous materials not previously evaluated, the test procedure provides adequate controls and precautions to ensure that the siting, storage, handling, and usage of these materials does not change the manner in which the reactor or any safety-related reactor plant system operates, or introduce failures of safety-related equipment which could create a different accident than those previously evaluated. The test does not create a reduction in any margin of safety as defined in the basis of any technical specification. The performance of this test does not constitute an unreviewed safety question.

ATTACHMENT I

Special Procedure SP-86-081 complies with the following codes or sections of codes as listed below:

NFPA 50, 1974, Bulk Oxygen Systems at Consumer Sites, Sections 4.0 and 5.0
NFPA 50A, 1973, Gaseous Hydrogen Systems at Consumer Sites, Section 5.0

NUREG 0800, Branch Technical Position CMEB 9.5.1, Section C.5.d, Control of Combustibles

EPRI NP-4500-SR-LD, March 1986, Guidelines for Permanent BWR Hydrogen Water Chemistry Installations, Section 4.1, Safety Considerations, Gaseous Hydrogen

CGA 4.1, 1985, Cleaning Equipment for Oxygen Service

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-87-090, Estimating Leakage Past the Circulating Water System (CW) Inlet Water Box Valves 2-CW-V13 and 2-CW-V12, SP-87-090, Revision 0

DESCRIPTION: The test will provide procedural guidance for specific valve manipulations which will provide a basis for estimating leakage past the A-south and B-north CW System debris filter (water box) inlet valves.

SAFETY SUMMARY: Performance of this procedure will not adversely impact any design basis accident scenarios as evaluated in the FSAR. CW System and condenser effects will be minimal. The margin of safety as stated in the Unit 2 technical specifications will not be jeopardized. Plant system effects will be minimal.

MSC/88-099A

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 79-183, Unit 1 CRD Pump Suction Line Reroute; Provide a New Line From the Effluent of the Condensate Polishers (CDD and CFD) to the CRD Pump Suction Upstream of the Suction Filters and Leave the Existing Line From the Condensate Storage Tank (CST) in Place

FUNCTIONAL SUMMARY: The control rod drive (CRD) collet retainer tubes have had a history of cracking (reference SIL 139). As analyzed by General Electric (GE), this cracking is due "to thermal cycles during hot scrams, followed by exposure to oxygenated CRD cooling water which can be aggressive to sensitized materials." This cracking is significant in that "should the cracking progress to a point of complete circumferential separation, the CRD would be rendered inoperable." Besides redesigning these tubes, GE recommends in SIL 148, supplying the CRD with less oxygenated water. This would give a significant increase in time to crack formation. This modification deals with implementing this recommendation.

Presently, CRD water is supplied from the condensate storage tank (CST). This modification would install new piping and valves such that the new supply would come off the condensate system after the condensate filter demineralizers (CFDs) and the condensate deep-bed delmineralizers (CDDs), and connect to the present CRD suction header. A pressure control valve (PCV-4105) and two pressure indicators (PI-4106 and -4107) would be added to regulate CRD suction pressure. Also, with the use of two check valves (CO-V177 and COR-V51) the CST would be maintained as a backup supply should the condensate system be isolated.

SAFETY SUMMARY:

1. This modification does not directly affect any safety-related equipment. If does, however, directly affect the CRD system in that it alters the water supply to that system. An additional source of water is now available. Though this new source is the primary source for the system, the original source from the CST will remain as a backup water supply.

2. As stated in the Design Basis document, this "system operates without outside control to supply water to the CRD pump suctions whenever the condensate pumps are operating." The control valve has no outside power source other than instrument air. Should instrument air be lost, the control valve will close, and the system water will automatically be supplied from the CST. Check valves are installed in the two supply lines to prevent backflow from the CST to the condensate polisher and vice versa. Each source line isolation valve is normally open.

   Two modes of failure need to be addressed at this point: 1) failure of check valve V177, and 2) failure of PCV-4105 to the full open position.

   If check valve V177 failed, water would flow to the CST since the other end of the condensate supply header going to the torus is normally valved shut. The CST would eventually fill from this flow; however, preceding this event would be a water level drop in the hotwell. Level instrumentation would sense this drop in level and attempt to make up the difference. Hotwell makeup comes from the CST. There would, therefore, be a recirculation of water--hotwell to CST and back to the hotwell. Although not desirable, this in itself would not result in a degradation of plant operating conditions or potentially reduce the safety margin of any system. Furthermore, if the CST could be filled, it could not pressurize because there are two pressure outlets. A 7-inch vent pipe vents to the air at the top of the tank, and there's an overflow line from the CST to the auxiliary surge tank.

Failure of the PCV-4105 to the full open position would subject the piping downstream of the valve to the condensate system pressure. This could happen if the valve closing spring broke. From the Design Basis document, the system pressure would be 238 psig at 120°F. This pressurization would not damage any piping or components because of the presence of the relief valves F001A and F001B. The relief pressure setpoint of these valves is 250 psi. The pressure these pipes would see would not reach the relief valve setpoint.

3. This modification is designed to provide the rated system flow from either source line. As stated above, this modification does not directly affect a safety-related component or system.

Comments

This modification enhances the reliability of the control rod drives, because the low oxygenated water from the condensate train lengthens the time to initial cracking in the collet retainer tubes. This cracking has been prevalent in the past and could lead to a failure of a drive. Design and installation of this modification is in accordance with approved codes and standards, and the new piping has been hydro tested to ensure system integrity. During system tie-ins, the CRD pumps were shut down, thus a different flow of cooling water to the drives was necessary. A temporary mechanical jumper from the demineralized water system to just downstream of the drive water filters satisfied this need. The jumper was used when the unit was in cold shutdown and at 0 psig reactor pressure.

A rubber hose, rated for 200 psi, was sufficient to handle the 150 psi demineralized water. A check valve was used to prevent any contamination from entering the demineralized water system. The jumper was removed when the modification was completed.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 79-184, Unit 2 CRD Pump Suction Line Reroute; Provide a New Line From the Effluent of the Condensate Polishers (CDD and CFD) to the CRD Pump Suction Upstream of the Suction Filters and Leave the Existing Line From the Condensate Storage Tank (CST) in Place

FUNCTIONAL SUMMARY: The control rod drive (CRD) collet retainer tubes have had a history of cracking (reference SIL 139). As analyzed by General Electric (GE), this cracking is due "to thermal cycles during hot scrams, followed by exposure to oxygenated CRD cooling water which can be aggressive to sensitized materials." This cracking is significant in that "should the cracking progress to a point of complete circumferential separation, the CRD would be rendered inoperable." Besides redesigning these tubes, GE recommends in SIL 148, supplying the CRD with less oxygenated water. This would give a significant increase in time to crack formation. This modification deals with implementing this recommendation.

Presently, CRD water is supplied from the condensate storage tank (CST). This modification would install new piping and valves such that the new supply would come off the condensate system after the condensate filter demineralizers (CFDs) and the condensate deep-bed delmineralizers (CDDs), and connect to the present CRD suction header. A pressure control valve (PCV-4105) and two pressure indicators (PI-4106 and -4107) would be added to regulate CRD suction pressure. Also, with the use of two check valves (CO-V177 and COR-V51) the CST would be maintained as a backup supply should the condensate system be isolated.

SAFETY SUMMARY:

1. This modification does not directly affect any safety-related equipment. If does, however, directly affect the CRD system in that it alters the water supply to that system. An additional source of water is now available. Though this new source is the primary source for the system, the original source from the CST will remain as a backup water supply.

2. As stated in the Design Basis document, this "system operates without outside control to supply water to the CRD pump suctions whenever the condensate pumps are operating." The control valve has no outside power source other than instrument air. Should instrument air be lost, the control valve will close, and the system water will automatically be supplied from the CST. Check valves are installed in the two supply lines to prevent backflow from the CST to the condensate polisher and vice versa. Each source line isolation valve is normally open.

   Two modes of failure need to be addressed at this point: 1) failure of check valve V177, and 2) failure of PCV-4105 to the full open position.

   If check valve V177 failed, water would flow to the CST since the other end of the condensate supply header going to the torus is normally valved shut. The CST would eventually fill from this flow; however, preceding this event would be a water level drop in the hotwell. Level instrumentation would sense this drop in level and attempt to make up the difference. Hotwell makeup comes from the CST. There would, therefore, be a recirculation of water--hotwell to CST and back to the hotwell. Although not desirable, this in itself would not result in a degradation of plant operating conditions or potentially reduce the safety margin of any system. Furthermore, if the CST could be filled, it could not pressurize because there are two pressure outlets. A 7-inch vent pipe vents to the air at the top of the tank, and there's an overflow line from the CST to the auxiliary surge tank.

Failure of the PCV-4105 to the full open position would subject the piping downstream of the valve to the condensate system pressure. This could happen if the valve closing spring broke. From the Design Basis document, the system pressure would be 238 psig at 120°F. This pressurization would not damage any piping or components because of the presence of the relief valves F001A and F001B. The relief pressure setpoint of these valves is 250 psi. The pressure these pipes would see would not reach the relief valve setpoint.

3. This modification is designed to provide the rated system flow from either source line. As stated above, this modification does not directly affect a safety-related component or system.

Comments

This modification enhances the reliability of the control rod drives, because the low oxygenated water from the condensate train lengthens the time to initial cracking in the collet retainer tubes. This cracking has been prevalent in the past and could lead to a failure of a drive. Design and installation of this modification is in accordance with approved codes and standards, and the new piping has been hydro tested to ensure system integrity. During system tie-ins, the CRD pumps were shut down, thus a different flow of cooling water to the drives was necessary. A temporary mechanical jumper from the demineralized water system to just downstream of the drive water filters satisfied this need. The jumper was used when the unit was in cold shutdown and at 0 psig reactor pressure.

A rubber hose, rated for 200 psi, was sufficient to handle the 150 psi demineralized water. A check valve was used to prevent any contamination from entering the demineralized water system. The jumper was removed when the modification was completed.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 80-078, Wide Range Containment Water Level Monitoring; The Suppression Pool Indicating Range is Being Increased to (-)10 ft to (+)6 ft as a Result of Requirements Outlined in NUREG 0737.

FUNCTIONAL SUMMARY: The indication range for the suppression pool is being increased from (-)6' to (+)6' to the new range of (-)10' to (+)6' as a result of the requirements outlined in NUREG-0737.

Plant Modification 80-016 already completed installed two new 1/2" instrument lines along side the existing 24" RHR suction pipes at penetration X225A and X225B. These penetrations will be used for the new lower reference point (-10') of suppression pool for water level measurement.

The existing two lower reference connections to the suppression pool at (-)6', will be spared along with their associated RIP valves and control and indication circuits.

All existing piping to transmitter low and high sensing ports will be removed and new piping, tubing, and valves will be installed. The present configuration in which one transmitter initiates both a narrow and wide range indication will be changed. A new transmitter CAC-LT-4177 will be installed for narrow range indication and annunciation.

Transmitter loops CAC-LT-2601 and CAC-LT-2602 will now receive their loop power from the Foxboro isolation cabinets.

Excess flow check valves will be installed in the new lower measurement lines with position indication on the RTG board.

A normally isolated non-Q gage glass will be installed for local level indication in the north RHR room.

A surge volume reservoir will be installed at the top of each reference leg column to ensure a constant pressure to the transmitter reference port to eliminate the calibration problems the existing level system is experiencing.

SAFETY SUMMARY: The increased measurement range of the suppression pool level in conjunction with the more reliable instrumentation and new reference leg configuration will ensure a safer and more accurate overall measurement monitoring system, actually increasing the margin of safety as defined in the technical specifications. The probability or consequences of an accident or equipment malfunction will not be increased.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 80-134, Dedicated Hydrogen Control; Upgrade of the Containment Atmospheric Dilution (CAD) System to Meet the Requirements of NUREG-0737

FUNCTIONAL SUMMARY: The purpose of this modification was to install a dedicated containment atmospheric dilution (CAD) nitrogen injection system for containment hydrogen control during a postaccident mode to meet the requirements of NUREG-0737, Item IIE.4.1.

The upgraded system is designed to meet the above requirement by having a combined design that is single active failure proof for both containment isolation and operation of the purge system. In order to meet these requirements, the scope of work included the following:

The CAD nitrogen injection piping was relocated from the containment atmospheric control (CAC) inerting lines. A dual dedicated path designed to protect against single active failure for CAD nitrogen injection into both the drywell and the suppression chamber was provided.

The containment vent valves and their associated controls to the Standby Gas Treatment System were modified to provide single failure protection for containment isolation and venting.

CAC valves not required to operate after a DBE no longer have a containment isolation signal override feature.

The nitrogen supply from the CAD tank to the noninterruptible instrument air system were removed and the line capped.

In addition to the above work, this modification provided the following:

Flow control and indicating capability for both CAD injection and CAC makeup nitrogen flow to the drywell and suppression chamber to facilitate system operation.

All new control valves are solenoid operated.

High drywell pressure (40 psig) isolation interlock to inerting, makeup, and CAD valves was removed. Annuciation still exists.

Electrical cables associated with the CAC inboard and outboard isolation valve circuits were upgraded to Engineering Safeguard System (ESS) level. Wiring changes were made to provide the CAC inboard isolation valves with a Division I isolation signal only and the CAC outboard isolation valves with a Division II isolation signal only.

SAFETY SUMMARY: Addition of containment isolation valves for postaccident combustible gas control does not impact any accident analysis previously evaluated in Chapter 15 of the FSAR. The addition of containment isolation valves for the CAC System does not affect the consequences of any transient or accident event evaluated in Chapter 15 of the FSAR. The addition of new containment isolation valves will not impact the containment isolation capability of any existing valves. The consequences of a malfunction of equipment previously evaluated in the FSAR will be decreased by providing a post-LOCA Nitrogen Purge System (CAD) that is designed against single failure of an active component. The function and design requirements of the new containment isolation valves will meet the requirements for existing containment isolation valves. Therefore, an accident or malfunction of a different type will not be created. The margin of safety to control combustible gases post-LOCA increased by designing the system against single failure of components. The margin of safety for containment isolation will not be reduced.

CHANGE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-221F, Addition of Anticavitation Flow Control Valve to Reactor Building Closed Cooling Water (RBCCW) System Heat Exchanger (Hx) Service Water (SW) Outlet; Installs New Flow Control Valve 1-SW-V382 in the RBCCW HX SW Outlet Piping and Also Replaces RBCCW HX Inlet and Outlet Butterfly Valves.

FUNCTIONAL SUMMARY: This Plant Modification (PM) 82-221F provides instructions to install a new anticavitation flow control valve 1-SW-V382 in the common SW discharge line from the RBCCW Hx. Addition of this valve will eliminate the need to throttle flow with the Hx SW outlet butterfly valves and will serve to reduce cavitation of the piping system because of throttling with these valves.

Also included in the scope of this modification are replacement of the existing RBCCW Hx SW inlet (1-SW-V107, V108, and V109) and outlet (1-SW-V133, V134, and V135) butterfly valves with new aluminum bronze lug body butterfly valves. The flanged piping spools downstream of the Hx outlet valves will be replaced with new 70/30 Cu-Ni piping spools. Existing, spared 1-SW-RT-58-3, along with associated instrument lines, will be removed and electrical connections will be determinated and spared.

Electrical ripout of radiation analyzer will require coordination to be worked in conjunction with Plant Modification 82-219W.

This PM has been reviewed in accordance with procedure RP-1.97 with regard to impact upon instrumentation credited for postaccident monitoring capability and the conclusion is that no postaccident monitoring capabilities are affected.

SAFETY SUMMARY: Systems and components modified within the scope of this plant modification are not addressed in Chapter 15 of the UFSAR. This modification does not introduce a new failure mode to the system nor alter any safety margin as defined by the technical specifications for BSEP Unit 1.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-287W, Penetration X-206A-C and D, X206C-C and D, and X-225B RIP Valve Replacement; This Modification Replaces the Penetration X-206A-C and D, X206C-C and D, and X-225B Instrument Isolation Valves With Direct Acting Solenoid Valves or a Pipe Cap.

FUNCTIONAL SUMMARY: The purpose of this modification was to replace existing air-operated isolation valves on the instrument lines exiting from drywell penetrations 1-X-206A, 1-X-206C, and 1-X-225B with direct acting solenoid valves. Removal of the air-operated isolation valves and associated instrument air equipment, supports our commitment to comply with IEB-79-01B. Eliminating the air-operated valves also eliminated maintenance problems resulting from diaphragm leaks in the valve operators and frozen solenoid valves caused by moisture in the instrument air system.

The new design solenoid valves provide remote manual isolation capability of the drywell instrument lines from penetrations 1-X-206A, 1-X-206C, and 1-X-225B, which is identical to the original design function. Position switches will be utilized to provide remote indication of valve position in the Control Room.

Air-operated isolation valves 1-E41-PV-1218D, 1-E41-PV-1220D, 1-CAC-PV-1218C, and 1-CAC-PV-4345, along with their associated air system equipment has been removed. In addition, previously spared air-operated isolation valve 1-CAC-PV-1220C was removed. The existing equipment of the RNA System serving penetration 1-X-206A was removed and valves 1-RNA-IV-575 and 1-RNA-IV-908 were plugged.

Valve 1-E41-PV-1218D was replaced by solenoid valve 1-E41-SV-1218D. Valve 1-E41-PV-1220D was replaced by solenoid valve 1-E41-SV-1220D. Valve 1-CAC-PV-1218C was replaced by solenoid valve 1-CAC-SV-1218C. Valve 1-CAC-PV-4345 was replaced by solenoid valve 1-CAC-SV-4345. Valve 1-CAC-PV-1220C which was previously spared, has been removed and the line capped for future use.

SAFETY SUMMARY: No increase is expected in the probability of occurrence or consequences of an accident or malfunction of equipment as a result of this modification. This statement is based on the following:

- Current isolation valves rely on both electrical and pneumatic power for valve actuation. A failure or transient in either of these systems can cause spurious isolation valve actuation. Replacement valves are direct acting, reducing the dependence on any support systems.

- Current isolation valves are maintenance concerns generally resulting from moisture in instrument air and faulty diaphragms in the operators. Replacement valves should be less of a concern to maintenance.

- Basic design function remains unchanged.

An actual decrease in system failure is expected (increased reliability) based on the above.

The valves installed via this modification are qualified to operate in a postaccident environment per IEE 3?3-1974 standard.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-287X, Penetration 1-X206B, 1-X206D, and 1-X225A RIP Valve Replacement; This Modification Replaces the Existing Penetration 1-X-206B-B/C/D, 1-X206D-C/D, and 1-X225A Isolation Valves for Suppression Chamber Instruments Serving the E41 and CAC Systems With Direct Acting Solenoid Valves.

FUNCTIONAL SUMMARY: The purpose of this modification is to:

1. Replace existing air-operated suppression chamber isolation valves in the instrument lines exiting from penetrations 1-X206B-B/C/D and 1-X206D-D with direct acting solenoid valves.

2. Replace existing spared manual isolation valve in the instrument line exiting penetration 1-X206D-C with a cap.

3. Replace existing excess flow check valve in the instrument line exiting penetration 1-X225A with a direct acting solenoid valve.

Removal of existing air-operated isolation valves and associated instrument air equipment supports our commitment to comply with IEB 79-01B. Eliminating the air-operated valves will also eliminate present maintenance problems resulting from diaphragm leaks to the valve operators and frozen solenoid valves caused by moisture in the instrument air system. Air-operated valves 1-CAC-PV-1219B, 1-CAC-PV-1219C, and 1-E41-PV-1219D, and all instrument air instrumentation and tubing from these valves to the header which extends from root valve 1-RNA-V-226 will be removed. Air-operated valve 1-E41-PV-1221D and all instrument air instrumentation and tubing from this valve to 1-RNA-IV-835 will be removed.

The new solenoid valves will be numbered as follows:

Penetration	Isolation Valve
1-X206B-B	1-CAC-SV-1219B
1-X206B-C	1-CAC-SV-1219C
1-X206B-D	1-E41-SV-1219D
1-X206D-C	None (capped)
1-X206D-D	1-E41-SV-1221D
1-X225A	1-CAC-SV-4344

SAFETY SUMMARY: No increase is expected in the probability of occurrence or consequences of an accident or malfunction of equipment as a result of this modification. This statement is based on the following:

- Current air-operated isolation valves rely on both electrical and pneumatic power for valve actuation. A failure or transient in either of these systems can cause spurious isolation valve actuation. Replacement valves are direct acting, reducing the dependence on any support systems.

- Current air-operated isolation valves are maintenance concerns generally resulting from moisture in instrument air and faulty diaphragms in the operators. Replacement valves should be less of a concern to maintenance.

- Basic design function remains unchanged.

An actual decrease in system failure is expected (increased reliability) based on the above.

The solenoid valves installed via this modification are qualified to operate in a postaccident environment per IEE 323-1974 standard.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-003, Spent Fuel Pool Storage Expansion--Phase II; Install Two HDFSS Modules and Miscellaneous Storage Components

FUNCTIONAL SUMMARY:

1.1 The purpose of this plant modification is to increase the BWR spent fuel storage capacity of the Unit 1 spent fuel pool by the installation of two additional General Electric (GE) designed high density fuel storage system (HDFSS) modules. The first three HDFSS modules were installed under Plant Modification 82-205 (PM 82-205). The remaining modules to be installed consist of rectangular fuel storage cells combined in a 13 x 15 and a 13 x 19 matrix assembly.

1.2 In order to install the modules, it is necessary to remove the existing control rod storage hangers and the two existing sipping-can support tubes from the spent fuel pool by using underwater divers.

1.3 Before underwater diving activities can begin, the spent fuel pool must be reconditioned to reduce the radiation levels in the pool and create a "safe dive zone" for the divers.

1.3.1 To create a safe dive zone, spent fuel assemblies/fuel bundles must be moved from the first row of BWR and PWR racks (rows A1 through A6) to other racks in the pool.

1.3.2 Vacuuming and hydrolazing of the fuel pool work area prior to the start of initial diving (or as required by E&RC) will also be required to reduce the radiation levels in the pool to an acceptable level for divers.

1.4 The new HDFSS modules are free standing. They do not latch to the grid nor to the walls or floor of the fuel pool nor do they tie to each other. In order not to transmit structural loads to the grid, the modules rest on large support bases which are designed to bridge the grid, leak detection channels, diffuser pipes, swing bolts, bearing plates (from the original rack latch downs), and the lateral grid restraint system. The support bases do not tie to the pool floor. They are keyed and constructed so as to fit in only one location on the floor. The support bases are constructed of 304 stainless-steel plate with three lifting eyes on each base.

1.4.1 The modules sit on the support bases and are designed to slide and rock in a seismic event. The upper surface of the support base is ground smooth and flat. The load-bearing pads on the bottom of the modules are made of a special low-friction material. There are four load-bearing pads on each module.

1.4.2 The HDFSS modules are constructed of 304 stainless steel with an aluminum/boron (Boral) layer sandwiched into most cell walls as a neutron multiplication poison. This poison allows a much closer spacing of the fuel assemblies (high density) without exceeding the neutron multiplication factor (k/eff) listed in the design features of the technical specification (k/eff less than 0.95).

1.5 The effectiveness of the Boral poison in the modules was tested by source/detection/instruments prior to loading fuel assemblies into the modules installed by PM 82-205. The required monitoring of the Boral is now accomplished by the periodic testing of Boral coupons (PT-90.11) stored in the fuel pool.

1.6 There are several tools supplied with the modules for their installation. They include:

1.6.1 The level verification tool
1.6.2 The wet lifting fixture
1.6.3 The support base lifting fixture
1.6.4 The uprighting fixture
1.6.5 The tolerance verification fixture (dummy fuel assembly)
1.6.6 The wet lifting fixture attach/release tool
1.6.7 The module wall positioner
1.6.8 The module spacing tool

All of these tools are necessary for the installation of any module and must be kept until all ten HDFSS modules are installed in the two pools.

1.7 In addition to the modules and tools, there are covers supplied for the support bases. These are to be used in the event a support base is installed on the pool floor, and the module which rests on that support base is not installed. The cover protects the support base polished surface from damage. The base cannot be recovered easily from the pool floor after installation for touch-up polishing without the use of divers.

1.8 Up to 130 new Type D control rod hangers will be installed on the curb of the spent fuel pool.

1.9 A control rod transfer station is attached to the 17 x 15 module to allow for the transfer of a control rod between the existing control rod hanger facilities and the new control rod hangers. This transfer is necessitated by removal of the existing control rod storage structure to make room for the two HDFSS modules.

1.10 New sipping-can support tubes and a rack for trash cask and transuranic (in-core detector) liners, in addition to the modules, are also to be installed under this plant modification.

1.11 The two new sipping-can support tubes are to be installed along the south wall of the spent fuel pool. The tubes are supported at the base by attachment to the 18 x 10-inch thrust beam and laterally restrained at the top from an existing attachment to an embedded plate in the liner.

1.11.1 The sipping-can support tubes are fabricated from 10-inch (Sch 40) 304 stainless-steel pipe.

1.12 Racks for trash cask and transuranic (in-core detector) liners will be installed north of the spent fuel shipping cask storage area.

1.12.1 The racks are free standing. They do not latch to the grid nor to the walls or floor of the fuel pool. The support base of the racks is notched to bridge the 8-inch diameter thrust beam at the base of the pool and also the shipping cask shear ring. The racks are fabricated from 304 stainless-steel. The racks will provide no interference with the use of the IF-300 shipping cask.

1.13 The sipping can support tubes and the racks for trash cask and transuranic (in-core detector) liners are all designed to withstand the effects of the design basis earthquake.

SAFETY SUMMARY: There are no unreviewed safety questions as the result of this plant modification.

This modification increases the size of the storage capacity of the pool which places additional loads on the structural systems of the pool and on the cooling systems as well. The result of the additional weight brings the shear stress in the pool slab to within 12 percent of allowable. This will require an administrative limit of 75 tons on the fuel cask's maximum weight. The additional heat loads, as a result of added fuel, change the time to boiling in the event of a loss of cooling accident and change the system's configuration necessary to maintain the pool temperature below 150°F during different fueling evolutions (i.e., either the fuel pool gates must remain removed or RHR spent pool cooling assist in operation for a longer period of time). As both the additional structural and heat loads represent increases to accident probability or consequences, they were unreviewed initially. The additional loads on the structural and cooling systems created by the addition of five HDFSS modules were reviewed by the NRC and were licensed effective December 15, 1983. (Amendment Nos. 61 and 87 to Facility Design License Nos. DPR-71 and DPR-62). The limit on the cask was included as an FSAR revision (Section 9.1.4.2.1), as part of the Unit 2, Phase 1, modification (PM 82-204).

The new HDFSS modules for BWR fuel storage hold the fuel assemblies as much as 10 inches higher in the water than the present grid-mounted fuel storage racks. One consequence of this involves the level of water which would remain in the fuel pool after a loss of water accident during a time in which the pool gates were removed and the reactor well flooded. Certain component failures in the reactor well or dryer/separator pool (such as the reactor well bellows seal) would allow the spend fuel pool to drain down to the highest elevation in the fuel transfer channel. As originally designed, this highest elevation would keep some water cover over the active fuel region of the fuel assemblies stored in the grid-mounted racks. In the new modules, the low water level would have allowed from 3 to 6 inches of the active fuel region of the assemblies to be exposed (above the water level). As postulated accidents, which expose stored spent fuel to air, are not addressed in the FSAR, a modification to install a 10-inch Class I seismic barrier in the refueling channel (to raise the minimum level to which the fuel pool could be drained by 10 inches) was completed under PM 83-005. The FSAR was changed under PM 82-204 to identify the top of the barrier as the lowest level that the fuel pool can drain to with the fuel pool gates removed. There is now no possibility of the occurrence of an unanalyzed accident as a result of this modification.

Another consequence of the increased elevation of the stored spent fuel deals with the minimum water cover which is present over the fuel during normal water levels. The concern here is the decontamination factor (DF) for fission gases released from leakers, or in the worst case, from a damaged fuel assembly stored in the module. The result of the fuel in the HDFSS modules being at a higher elevation naturally results in a reduction of the water cover for any fuel in them. For this reason, the technical specification was changed from the requirement of at least 22 feet, 3 inches of water over the active fuel region of the fuel assemblies stored in the spent fuel storage racks to 20 feet, 6 inches of water over the top of irradiated fuel rods of the fuel assemblies stored in the spent fuel storage pool racks. In addition, this change lowered the design basis requirement of removal of 99 percent of the iodine that would be released in the event of a refueling accident to 98 percent of the iodine. These changes were reviewed by the NRC and were licensed effective October 25, 1984 (Amendment Nos. 77 and 104 to Facility Design License Nos. DPR-71 and DPR-62).

A third consequence of the increased elevation of the fuel assemblies in the HDFSS modules lies in the fact that fuel carried by the fuel grapple will have to clear the lifting bails of fuel stored in the modules. The present MI-10-2AA setting for the normal up limit on the fuel grapple will result in the proper clearance if the grapple is a 731E635 Group 3 type (original to Unit 1) but not if the grapple is a 731E635 Group 1 type (original to Unit 2). If this lift height setting is maintained for transferring the fuel assembly through the channel and into the vessel, the bottom of the assembly will be 31 feet, 7 inches above the core (as measured to the top of the fuel channels). As the present FSAR limit is 30 feet, a change to the FSAR was submitted with PM 82-204 as required. An analysis for a 32-foot drop has been completed and shows that the conclusion presented in the present FSAR on a fuel handling accident will not be impacted as a result of the fuel handling equipment modification. A modification to install a grapple on Unit 1, which has similar lifting capabilities to the Group 3 grapple (the original Unit 1 grapple was moved to Unit 2), was completed under PM 83-194. This new grapple is a GE NF-400 type grapple, Model 769E521 Group 1. The FSAR change to reflect a 32-foot drop for the refueling accident was submitted as part of Unit 2 modification, Phase I (PM 82-204).

The use of poisoned fuel storage modules in the spent fuel storage pool and the limits of criticality for poisoned storage were not originally addressed in the FSAR or technical specification; therefore, they were unreviewed. The unreviewed safety questions were submitted in the Fuel Storage Design Features License Amendment Request approved by the NRC on December 15, 1983. All FSAR changes were handled as part of the Unit 2, Phase I, modification (PM 82-204). The affected technical specification sections were:

3.9.9 Minimum Water Depth to be Maintained Over Irradicated Fuel (approved October 25, 1984); and

5.6 Fuel Storage Design Features (approved December 15, 1983).

Construction and Operations Activities

This modification will involve the movement of fuel within the fuel pool, the removal and installation of miscellaneous spent fuel pool components in the pool, and the installation of HDFSS modules in the pool. The principal concerns are carrying loads which weigh more than 1600 lbs over fuel storage racks which contain spent fuel assemblies and maintaining strict control of spent fuel assembly locations stored in the pool. The modification procedures will ensure that no loads in excess of 1600 lbs will be moved over fuel stored in any spent storage racks or modules. The modification procedure will include safe load plans for all evolutions where a load in excess of 1600 lbs is moved within the pool. This document will be a prerequisite to these moves. The approved fuel handling procedures for the plant will be utilized to move the spent fuel, and these will ensure that close track is kept over the fuel locations.

ALARA

Features to keep radiation exposure as low as reasonably achievable (ALARA) have been incorporated in the design of the equipment being installed. ALARA concerns with the actual installation work are primarily associated with contaminated components which will be removed and the direct radiation hazards in and around the spent fuel pool. Typical exposure rates around the spent fuel pool range from 5 to 10 mR/hr. Radiation exposure rates in the pool at the diver's work area are expected to be between 20 to 75 mR/hr after vacuuming the pool bottom and relocating active fuel elements to the maximum distance from the area. Much of the work will be performed in lower radiation areas away from the spent fuel pool area. E&RC will provide necessary radiological assessment, monitoring, and direction to assist in minimizing radiation exposure to the workers. The highest doses for the planned work will likely be to the diver and Reactor Building crane operator. Exposure conditions in these positions should be carefully monitored.

Control of Heavy Loads

Among other requirements, the Safety Evaluation Report (SER) issued with the approved license specifically addressed the special lifting equipment with regard to its design limits and testing in accordance with ANSI N14.6. All instructions and requirements for movement of equipment are contained in the modification package installation procedure. The SER issued with the approved license amendments for these modifications addressed the testing of the fixture. The fixture will be load tested to 150 percent of the heaviest load to be lifted (load rating of the fixture is 32,000 lbs) prior to use. This equipment was evaluated on December 15, 1983, in the SER as "adequate, and therefore, acceptable." The modification procedure ensures that the safe load path requirements are satisfied and also ensures that no heavy loads are moved over stored spent fuel.

BCU will operate the Reactor Building crane per their Work Procedure WP-10. Both the engineer(s) and all crane operators working this plant modification will sign off one time on the safe load path diagram prior to moving either the support bases, grid-mounted racks, or any of the new storage racks within this load path area. This signature will remain in effect as proof of that operator's awareness of the safe load area and need not be repeated for each move or evolution. The operators will be qualified to standards equal to those required by the latest revision of MP-06.

Unlimited movement of all heavy loads required by this modification (grid racks, support bases, HDFSS modules, lifting fixture) are permitted within the safe load path area defined on the safe load path diagram in conjunction with the following clarification. Movement of all heavy loads into or out of the spent fuel pool will be controlled by a safe load path drawn by the Responsible BESU Engineer specifically for that piece of equipment. Each of the specific safe load path diagrams will be signed and dated by the Responsible BESU Engineer and the crane operator(s) making the particular move.

The October 12, 1984, amendment of the December 15, 1983, Safety Evaluation Report also affirms a commitment in the license submittal that handling procedures will be such that the storage racks immediately adjacent to the rack being moved will have a 2-foot, 6-inch buffer area void of fuel. (Movement of HDFSS modules only. Movement of smaller grid-mounted racks shall be performed per MP-28.)

Environmental Impact Evaluation

This modification increases the spent fuel pool storage capacity, replaces the existing sipping-can station, and installs a rack for the storage of trash cask and transuranic (in-core detector) liners. There will be no interaction or changes to circulating water and, therefore, the maximum allowed thermal rise or chlorine concentration of the circulating water will not be exceeded, and there will be no adverse affect of the monitoring of same.

This modification will not affect the ability of the meteorological monitoring system to accurately sense atmospheric conditions.

This modification will not cause the radiological limits of Technical Specification Section 3/4.11 to be exceeded or adversely affect the ability to monitor same. See Section 4.0, Unreviewed Safety Questions, for justification.

This modification will not cause a change in the quantity or size of entrained organisms in the cooling systems.

This modification will only use chemicals approved by the site and, therefore, will not increase the likelihood of an uncontrolled oil or chemical spill.

This modification will not create an adverse environmental impact.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-004, Spent Fuel Pool Storage Expansion--Phase II; Install Two HDFSS Modules and Miscellaneous Storage Components

FUNCTIONAL SUMMARY:

1.1 The purpose of this plant modification is to increase the BWR spent fuel storage capacity of the Unit 2 spent fuel pool by the installation of two additional General Electric (GE) designed high density fuel storage system (HDFSS) modules. The first three HDFSS modules were installed under Plant Modification 82-204 (PM 82-204). The remaining modules to be installed consist of rectangular fuel storage cells combined in a 13 x 15 and a 13 x 19 matrix assembly.

1.2 In order to install the modules, it is necessary to remove the existing control rod storage hangers and the two existing sipping-can support tubes from the spent fuel pool by using underwater divers.

1.3 Before underwater diving activities can begin, the spent fuel pool must be reconditioned to reduce the radiation levels in the pool and create a "safe dive zone" for the divers.

1.3.1 To create a safe dive zone, spent fuel assemblies/fuel bundles must be moved from the first row of BWR and PWR racks (rows A1 through A6) to other racks in the pool.

1.3.2 Vacuuming and hydrolazing of the fuel pool work area prior to the start of initial diving (or as required by E&RC) will also be required to reduce the radiation levels in the pool to an acceptable level for divers.

1.4 The new HDFSS modules are free standing. They do not latch to the grid nor to the walls or floor of the fuel pool nor do they tie to each other. In order not to transmit structural loads to the grid, the modules rest on large support bases which are designed to bridge the grid, leak detection channels, diffuser pipes, swing bolts, bearing plates (from the original rack latch downs), and the lateral grid restraint system. The support bases do not tie to the pool floor. They are keyed and constructed so as to fit in only one location on the floor. The support bases are constructed of 304 stainless-steel plate with three lifting eyes on each base.

1.4.1 The modules sit on the support bases and are designed to slide and rock in a seismic event. The upper surface of the support base is ground smooth and flat. The load-bearing pads on the bottom of the modules are made of a special low-friction material. There are four load-bearing pads on each module.

1.4.2 The HDFSS modules are constructed of 304 stainless steel with an aluminum/boron (Boral) layer sandwiched into most cell walls as a neutron multiplication poison. This poison allows a much closer spacing of the fuel assemblies (high density) without exceeding the neutron multiplication factor (k/eff) listed in the design features of the technical specification (k/eff less than 0.95).

1.5 The effectiveness of the poison in the modules was tested by source/detection/instruments prior to loading fuel assemblies into the modules installed by PM 82-204. The required monitoring of the Boral is now accomplished by the periodic testing of Boral coupons (PT-90.11) stored in the fuel pool.

1.6 There are several tools supplied with the modules for their installation. They include:

1.6.1 The level verification tool
1.6.2 The wet lifting fixture
1.6.3 The support base lifting fixture
1.6.4 The uprighting fixture
1.6.5 The tolerance verification fixture (dummy fuel assembly)
1.6.6 The wet lifting fixture attach/release tool
1.6.7 The module wall positioner
1.6.8 The module spacing tool

All of these tools are necessary for the installation of any module and must be kept until all ten HDFSS modules are installed in the two pools.

1.7 In addition to the modules and tools, there are covers supplied for the support bases. These are to be used in the event a support base is installed on the pool floor, and the module which rests on that support base is not installed. The cover protects the support base polished surface from damage. The base cannot be recovered easily from the pool floor after installation for touch-up polishing without the use of divers.

1.8 Up to 130 new Type D control rod hangers will be installed on the curb of the spent fuel pool.

1.9 A control rod transfer station is attached to the 17 x 15 module to allow for the transfer of a control rod between the existing control rod hanger facilities and the new control rod hangers. This transfer is necessitated by removal of the existing control rod storage structure to make room for the two HDFSS modules.

1.10 New sipping-can support tubes and a rack for trash cask and transuranic (in-core detector) liners, in addition to the modules, are also to be installed under this plant modification.

1.11 The two new sipping-can support tubes are to be installed along the south wall of the spent fuel pool. The tubes are supported at the base by attachment to the 18 x 10-inch thrust beam and laterally restrained at the top from an existing attachment to an embedded plate in the liner.

1.11.1 The sipping-can support tubes are fabricated from 10-inch (Sch 40) 304 stainless-steel pipe.

1.12 Racks for trash cask and transuranic (in-core detector) liners will be installed north of the spent fuel shipping cask storage area.

1.12.1 The racks are free standing. They do not latch to the grid nor to the walls or floor of the fuel pool. The support base of the racks is notched to bridge the 8-inch diameter thrust beam at the base of the pool and also the shipping cask shear ring. The racks are fabricated from 304 stainless-steel. The racks will provide no interference with the use of the IF-300 shipping cask.

1.13 The sipping can support tubes and the racks for trash cask and transuranic (in-core detector) liners are all designed to withstand the effects of the design basis earthquake.

SAFETY SUMMARY: There are no unreviewed safety questions as the result of this plant modification.

This modification increases the size of the storage capacity of the pool which places additional loads on the structural systems of the pool and on the cooling systems as well. The result of the additional weight brings the shear stress in the pool slab to within 12 percent of allowable. This will require an administrative limit of 75 tons on the fuel cask's maximum weight. The additional heat loads, as a result of added fuel, change the time to boiling in the event of a loss of cooling accident and change the system's configuration necessary to maintain the pool temperature below 150°F during different fueling evolutions (i.e., either the fuel pool gates must remain removed or RHR spent pool cooling assist in operation for a longer period of time). As both the additional structural and heat loads represent increases to accident probability or consequences, they were unreviewed initially. The additional loads on the structural and cooling systems created by the addition of five HDFSS modules were reviewed by the NRC and were licensed effective December 15, 1983. (Amendment Nos. 61 and 87 to Facility Design License Nos. DPR-71 and DPR-62). The limit on the cask was included as an FSAR revision (Section 9.1.4.2.1), as part of the Unit 2, Phase 1, modification (PM 82-204).

The new HDFSS modules for BWR fuel storage hold the fuel assemblies as much as 10 inches higher in the water than the present grid-mounted fuel storage racks. One consequence of this involves the level of water which would remain in the fuel pool after a loss of water accident during a time in which the pool gates were removed and the reactor well flooded. Certain component failures in the reactor well or dryer/separator pool (such as the reactor well bellows seal) would allow the spend fuel pool to drain down to the highest elevation in the fuel transfer channel. As originally designed, this highest elevation would keep some water cover over the active fuel region of the fuel assemblies stored in the grid-mounted racks. In the new modules, the low water level would have allowed from 3 to 6 inches of the active fuel region of the assemblies to be exposed (above the water level). As postulated accidents, which expose stored spent fuel to air, are not addressed in the FSAR, a modification to install a 10-inch Class I seismic barrier in the refueling channel (to raise the minimum level to which the fuel pool could be drained by 10 inches) was completed under PM 83-005. The FSAR was changed under PM 82-204 to identify the top of the barrier as the lowest level that the fuel pool can drain to with the fuel pool gates removed. There is now no possibility of the occurrence of an unanalyzed accident as a result of this modification.

Another consequence of the increased elevation of the stored spent fuel deals with the minimum water cover which is present over the fuel during normal water levels. The concern here is the decontamination factor (DF) for fission gases released from leakers, or in the worst case, from a damaged fuel assembly stored in the module. The result of the fuel in the HDFSS modules being at a higher elevation naturally results in a reduction of the water cover for any fuel in them. For this reason, the technical specification was changed from the requirement of at least 22 feet, 3 inches of water over the active fuel region of the fuel assemblies stored in the spent fuel storage racks to 20 feet, 6 inches of water over the top of irradiated fuel rods of the fuel assemblies stored in the spent fuel storage pool racks. In addition, this change lowered the design basis requirement of removal of 99 percent of the iodine that would be released in the event of a refueling accident to 98 percent of the iodine. These changes were reviewed by the NRC and were licensed effective October 25, 1984 (Amendment Nos. 77 and 104 to Facility Design License Nos. DPR-71 and DPR-62).

A third consequence of the increased elevation of the fuel assemblies in the HDFSS modules lies in the fact that fuel carried by the fuel grapple will have to clear the lifting bails of fuel stored in the modules. The present MI-10-2AA setting for the normal up limit on the fuel grapple will result in the proper clearance if the grapple is a 731E635 Group 3 type (original to Unit 1) but not if the grapple is a 731E635 Group 1 type (original to Unit 2). If this lift height setting is maintained for transferring the fuel assembly through the channel and into the vessel, the bottom of the assembly will be 31 feet, 7 inches above the core (as measured to the top of the fuel channels). As the present FSAR limit is 30 feet, a change to the FSAR was submitted with PM 82-204 as required. An analysis for a 32-foot drop has been completed and shows that the conclusion presented in the present FSAR on a fuel handling accident will not be impacted as a result of the fuel handling equipment modification. A modification to install a grapple on Unit 1, which has similar lifting capabilities to the Group 3 grapple (the original Unit 1 grapple was moved to Unit 2), was completed under PM 83-194. This new grapple is a GE NF-400 type grapple, Model 769E521 Group 1. The FSAR change to reflect a 32-foot drop for the refueling accident was submitted as part of Unit 2 modification, Phase I (PM 82-204).

The use of poisoned fuel storage modules in the spent fuel storage pool and the limits of criticality for poisoned storage were not originally addressed in the FSAR or technical specification; therefore, they were unreviewed. The unreviewed safety questions were submitted in the Fuel Storage Design Features

License Amendment Request approved by the NRC on December 15, 1983. All FSAR changes were handled as part of the Unit 2, Phase I, modification (PM 82-204). The affected technical specification sections were:

3.9.9 Minimum Water Depth to be Maintained Over Irradicated Fuel (approved October 25, 1984); and

5.6 Fuel Storage Design Features (approved December 15, 1983).

Construction and Operations Activities

This modification will involve the movement of fuel within the fuel pool, the removal and installation of miscellaneous spent fuel pool components in the pool, and the installation of HDFSS modules in the pool. The principal concerns are carrying loads which weigh more than 1600 lbs over fuel storage racks which contain spent fuel assemblies and maintaining strict control of spent fuel assembly locations stored in the pool. The modification procedures will ensure that no loads in excess of 1600 lbs will be moved over fuel stored in any spent storage racks or modules. The modification procedure will include safe load plans for all evolutions where a load in excess of 1600 lbs is moved within the pool. This document will be a prerequisite to these moves. The approved fuel handling procedures for the plant will be utilized to move the spent fuel, and these will ensure that close track is kept over the fuel locations.

ALARA

Features to keep radiation exposure as low as reasonably achievable (ALARA) have been incorporated in the design of the equipment being installed. ALARA concerns with the actual installation work are primarily associated with contaminated components which will be removed and the direct radiation hazards in and around the spent fuel pool. Typical exposure rates around the spent fuel pool range from 5 to 10 mR/hr. Radiation exposure rates in the pool at the diver's work area are expected to be between 20 to 75 mR/hr after vacuuming the pool bottom and relocating active fuel elements to the maximum distance from the area. Much of the work will be performed in lower radiation areas away from the spent fuel pool area. E&RC will provide necessary radiological assessment, monitoring, and direction to assist in minimizing radiation exposure to the workers. The highest doses for the planned work will likely be to the diver and Reactor Building crane operator. Exposure conditions in these positions should be carefully monitored.

Control of Heavy Loads

Among other requirements, the Safety Evaluation Report (SER) issued with the approved license specifically addressed the special lifting equipment with regard to its design limits and testing in accordance with ANSI N14.6. All instructions and requirements for movement of equipment are contained in the modification package installation procedure. The SER issued with the approved license amendments for these modifications addressed the testing of the fixture. The fixture will be load tested to 150 percent of the heaviest load to be lifted (load rating of the fixture is 32,000 lbs) prior to use. This equipment was evaluated on December 15, 1983, in the SER as "adequate, and therefore, acceptable." The modification procedure ensures that the safe load path requirements are satisfied and also ensures that no heavy loads are moved over stored spent fuel.

BCU will operate the Reactor Building crane per their Work Procedure WP-10. Both the engineer(s) and all crane operators working this plant modification will sign off one time on the safe load path diagram prior to moving either the support bases, grid-mounted racks, or any of the new storage racks within this load path area. This signature will remain in effect as proof of that operator's awareness of the safe load area and need not be repeated for each move or evolution. The operators will be qualified to standards equal to those required by the latest revision of MP-06.

Unlimited movement of all heavy loads required by this modification (grid racks, support bases, HDFSS modules, lifting fixture) are permitted within the safe load path area defined on the safe load path diagram in conjunction with the following clarification. Movement of all heavy loads into or out of the spent fuel pool will be controlled by a safe load path drawn by the Responsible BESU Engineer specifically for that piece of equipment. Each of the specific safe load path diagrams will be signed and dated by the Responsible BESU Engineer and the crane operator(s) making the particular move.

The October 12, 1984, amendment of the December 15, 1983, Safety Evaluation Report also affirms a commitment in the license submittal that handling procedures will be such that the storage racks immediately adjacent to the rack being moved will have a 2-foot, 6-inch buffer area void of fuel. (Movement of HDFSS modules only. Movement of smaller grid-mounted racks shall be performed per MP-28.)

Environmental Impact Evaluation

This modification increases the spent fuel pool storage capacity, replaces the existing sipping-can station, and installs a rack for the storage of trash cask and transuranic (in-core detector) liners. There will be no interaction or changes to circulating water and, therefore, the maximum allowed thermal rise or chlorine concentration of the circulating water will not be exceeded, and there will be no adverse affect of the monitoring of same.

This modification will not affect the ability of the meteorological monitoring system to accurately sense atmospheric conditions.

This modification will not cause the radiological limits of Technical Specification Section 3/4.11 to be exceeded or adversely affect the ability to monitor same. See Section 4.0, Unreviewed Safety Questions, for justification.

This modification will not cause a change in the quantity or size of entrained organisms in the cooling systems.

This modification will only use chemicals approved by the site and, therefore, will not increase the likelihood of an uncontrolled oil or chemical spill.

This modification will not create an adverse environmental impact.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-109, Suppression Pool Pressure Instrumentation; Provide Additional Suppression Pool Pressure Indication in the Main Control Room

FUNCTIONAL SUMMARY: This modification will provide additional suppression pool pressure indication in the main Control Room. This modification provides redundant indication of suppression pool pressure per Regulatory Guide 1.97, revision 2, as modified by Brunswick position paper on RG 1.97, revision 2, dated April 19, 1984.

This instrumentation will be divisionalized, redundant to the existing instrument loop (CAC-PI-1257-3). Equipment will be qualified to IEEE-323-1974 and IEEE-344-1975 (where applicable), powered by a 24-volt 1E power supply (located in the Foxboro isolator cabinet, XU-76, to provide future SPDS and TSC hookup), and meet separation and isolation criteria per IEEE-279-1771.

The existing loop components are to be retagged and designated loop A (Division I). Currently the existing loop utilizes Divisions I and II cables. This will be corrected by this plant modification and the loop will be entirely Division I. PM 82-256 has provided changeout of the existing transmitter so that environmental qualification is met. Range of 0 to 75 psig was not changed. PM 83-109 will provide same range and be designated loop B (Division II).

The process line to the Rosemount pressure transmitter will come from pipeline CAC-720. The pressure head in the surge volume reservoir will be zeroed out in the calibration of the transmitter.

This instrument loop is for indication only and will not have any trip functions or setpoints associated with it.

This modification will also modify the containment water level instrument racks, H21-P022-01 in south RHR and TS-2602-1 in north RHR, to resolve seismic concerns associated with differential movement between the Reactor Building and the torus. The modification involves disconnecting these racks from the torus wall.

SAFETY SUMMARY: This plant modification does not affect equipment previously evaluated in FSAR. It adds a redundant suppression pool pressure indication (Division II) and divisionalizes an existing instrument loop (Division I). The new instrument loop does not have any control functions, and both loops' equipment is procured, installed, and tested as Q-list, Class 1E, equipment. The equipment is not technical specification related; there are no control or setpoint changes.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-113, Rx Water Level Instrumentation Upgrade; Addition of New Redundant Wide Range Flood-Up Level Transmitter and Indicator

FUNCTIONAL SUMMARY: In "Position Paper on Regulatory Guide 1.97" and "Brunswick Response to NUREG 737, Supplement 1--Regulatory Guide 1.97 Application to Emergency Response Facilities," both dated 8/9/83, CP&L has agreed to make certain modifications to instrumentation to meet the NRC requirements. This PM provides the changes required to partially implement variable A2-RPV level measurement.

This PM adds a new flood-up level transmitter, B21-LT-N027B, on instrument rack H12-P005. This transmitter will be redundant with B21-LT-N027A. A new flood-up level indicator, B21-LI-R605B, on Control Room panel H12-P601 is also added.

To support this change, reference line B21-701 must be routed to instrument rack H21-P005 in addition to its present routing to rack H21-P004. New cables are also required.

Isolation modules in panel XU-76 in the Control Room provide loop power and an isolated signal for future ERFIS use.

SAFETY SUMMARY: The new instrument loop being added is for postaccident monitoring but is not required to perform any control function which would mitigate the consequences of a malfunction of equipment. An extension of instrument lines is the only change to existing equipment and new equipment is procured, installed, and tested as Q-list 1E equipment. The new instrument loop is not technical specification related nor is it electrically connected to any technical specification equipment.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-131, Drywell Pressure-Narrow Range; Add Narrow Range Pressure Transmitter and Indicator to Indicate -5 to +5 psig in Drywell

FUNCTIONAL SUMMARY: This modification adds a new drywell pressure instrumentation loop to measure the range -5 to +5 psig.
1-CAC-PT-5113/1-CAC-PI-5113

The transmitter is located in the Reactor Building. The indicator is located in the Control Room on panel H12-P601.

The loop receives power from isolation cabinet CB-XU-76. Cabinet XU-76 also provides an isolated signal for future use by the ERFIS.

This new instrument loop is required to meet Regulatory Guide 1.97 as agreed by CP&L in the BSEP position paper on RG 1.97. This modification implements RG 1.97 variable D4.

Existing instrumentation remains unchanged except for a new tee in one instrument line.

SAFETY SUMMARY: The instrumentation installed by this modification is qualified as 1E and does not affect any installed equipment. It provides additional indication only, has no control functions, is not technical specification related nor electrically connected to any technical specification related equipment. Considering all of the former conditions, it is concluded that a safety hazard is not introduced by this plant modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-143, Cooling Water Flow to ESF Components Instruments; Addition of Flow Indication for Loop A and B Discharge of Vital Header

FUNCTIONAL SUMMARY: This modification adds two new flow measurement loops to measure service water flow in the vital header.

Loop 1-SW-FE/FT/FI-5114 measures the flow in that half of the vital header which receives water from the conventional header. Loop 1-SW-FE/FT/FI-5115 measures the flow in that half of the vital header which receives water from the nuclear header.

The transmitters and flow elements are located in the Reactor Building. The indicators are located in the Control Room. Isolation modules in Cabinets XU-76 and XU-77 provide loop power and an isolated signal for ERFIS.

These new loops are required to meet Regulatory Guide 1.97, as agreed by CP&L in the BSEP position paper on RG 1.97. This modification implements RG 1.97 variable D22.

The existing Service Water System and instrumentation remain unchanged except for the addition of the new flow elements.

SAFETY SUMMARY: This plant modification does not affect equipment previously evaluated in the FSAR. System operation and function are unchanged by the installment of this postaccident monitoring instrumentation as it has no control functions. The equipment is procured, installed, and tested as Q-list, Class 1E.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-249, Transmitter Recalibration; Recalibrate Transmitters 1-B21-LT-N036 and -N037 to New Range -150 to +150 Inches

FUNCTIONAL SUMMARY: This modification recalibrates the following reactor vessel level instrument loops from the present range of -100 to +200 to -150 to +150 inches.

2-B21-LT-N036/2-B21-N036-1/2-B21-LI-R610
2-B21-LT-N037/2-B21-N037-1/2-B21-LR-R615

The transmitters LT-N036/N037 are located in the Reactor Building. Trip units N036-1/N037-1, indicator LI-R610, and recorder LR-R615 are located in the Control Room.

This recalibration is required to meet Regulatory Guide 1.97 as agreed by CP&L in the BSEP position paper on R.G. 1.97.

New scales will be required for the indicators and recorder. New chart paper will be required for the recorder.

The existing logic will not be modified and system operation will remain as is.

SAFETY SUMMARY: The equipment used and mode of usage is not changed by this plant modification. Due to the new range, the trip unit setpoint was recalibrated to meet technical specifications of greater than or equal to 53 inches above TAF, but technical specifications have not been altered, and there is no effect on safety as a result of this modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-005, Provide Alternate Feed for Battery Changer 1B-1; Install Alternate 480 Vac Feed to Battery Charger 1B-1 from MCC 1XB Compt. D3A and Associated Transfer Switch

FUNCTIONAL SUMMARY: This modification shall accomplish the following items. It shall provide alternate source of power to the 125/250 Vdc battery charger 1B-1. It will install a nonautomatic transfer switch to accomplish switching from the normal 480 Vac Source, MCC 1CB, to the alternate 480 Vac source, MCC 1XB, and install fuses in the shunt trip circuit of the normal 480V breaker. The modification shall also provide recommended procedure changes to the plant Operating Manual and the FSAR. The modification is provided as a part of CP&L compliance with the requirements set forth in 10CFR50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.

SAFETY SUMMARY: Providing an alternate source and transfer switch for 10CFR50, Appendix R, requirements does not change the basic design intent and will not cause a malfunction of equipment important to safety that has been previously evaluated in Chapter 15, nor does it increase the probability or possibility of an accident or malfunction of a different type.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-007, Battery Charger 1B-2, Alternate Feed; Install Alternate Feed for Battery Charger 1B-2 and Associated Transfer Switch

FUNCTIONAL SUMMARY: This modification shall accomplish the following items as part of CP&L's commitment to the requirement of 10CFR50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979. It shall provide an alternate source of power to the 125/250 Vdc battery charger 1B-2. It will install a nonautomatic transfer switch to accomplish switching from the normal 480 Vac source, MCC 1CB, to the alternate 480 Vac source, MCC 1XB. It shall install fuses in the shunt trip circuit of the normal 480V source to prevent battery charger control circuit degradation during the loss of its normal 480 Vac source.

SAFETY SUMMARY: Providing an alternate source and transfer switch for 10CFR50, Appendix R, requirements does not change the basic design intent and will not cause a malfunction of equipment important to safety that has been previously evaluated in Chapter 15, nor does it increase the probability or possibility of an accident or malfunction of a different type.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-102, Unit 2 Install Water Suppression System (Rx 5.0'); This Modification Installs a Line of Closely Spaced Closed Head Sprinklers Adjacent to Draft Stops in the Unit 2 Reactor Building, Elevation -17.0'

FUNCTIONAL SUMMARY: In accordance with the requirements of 10CFR50, Appendix R, "...licensees should reexamine those previously approved configurations of fire protection that do not meet the requirements specified in Section III.G to Appendix R..." A detailed reexamination and reanalysis of the Brunswick safe shutdown capability has been performed and is contained in the alternative shutdown capability assessment (ASCA report). Based on this, the Brunswick Steam Electric Plant concluded it can meet the requirements of 10CFR50, Appendix R, Section III.G, through a combination of alternative shutdown capability in accordance with Section III.G.3 requirements and separation of redundant functions in accordance with Section III.G.2.

10CFR50, Appendix R, Section III.G, requires fire protection features capable of limiting fire damage so that one train of systems necessary to achieve and maintain hot shutdown conditions is free of fire damage, and systems necessary for cold shutdown can be repaired in 72 hours.

One of the requirements of Appendix R, and as committed to by the BSEP, is fulfilled by installing water curtains in conjunction with draft stops along the periphery of the separation zone to prevent propagation of combustion products across the zone.

The Reactor Buildings are divided into two halves along an east/west line with one train of safe shutdown systems located in the northern half of the building and the other located in the southern half. Physical separation between the two halves is provided by the inerted drywell, torus, steam tunnel, ECCS room, and the HPCI room. Where such physical separation does not exist on the 5-foot elevation, separation zones of 20-foot width and free of significant quantities of intervening combustible are provided.

These separation zones will be provided with sprinklers utilizing the guidance of NFPA-13, Section 4-4.8.2.3. This section is concerned with the prevention of fire spread through large floor openings. To achieve this objective, lines of closely spaced, closed head sprinklers located along the periphery of the opening are to be installed with a draft stop provided between the opening and the sprinklers. Although this area is not a floor opening, the objective of limiting fire spread past a given vertical plane is the same. The direction of fire spread cannot be determined in advance, thus, a sprinkler/draft stop configuration will be provided to limit fire spread from north to south and south to north. Existing concrete beams are utilized as a draft stop on the south side of the separation zone, and the existing 20" RHR piping, in conjunction with the existing concrete beam on the north side of the separation zone, acts as the draft stop. Baffle will be installed to prevent one sprinkler from spraying or "cold-soldering" the adjacent sprinklers. Baffles will be affixed to the pipe in most cases, and to the ceiling or wall in others. The existing RHR pipe supports on the north side function as baffles, and the existing vertical cable trays on the south side of the zone at the east wall function as a baffle. The sprinkler heads are temperature actuated closed heads (165°F) and will supply a minimum of 3 gpm/linear foot. Sprinkler spacing is at a maximum of 6 ft (18 gpm), and no sprinkler will discharge less than 15 gpm.

The piping system is an extension of the existing area suppression system in the area. The tie-in is downstream of existing flow switch FS-3968. The water curtain is hydraulically calculated and designed. All pipe components are FP-Q. The piping will be supported by seismically designed (nonsafety) supports. Support material is procured as FP-Q per plant practice. The additional loads from these supports will not adversely affect the structural integrity of the existing slabs, walls, to which they are attached.

SAFETY SUMMARY: The additional water suppression system has no effect on accidents evaluated in the FSAR, Chapter 15. There is not a direct or an indirect interface. This modification reduces the potential for fire to propagate from one safety train to the other through spatial separation zones and water curtains, thus, the margin of safety is increased.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-105, Unit 1 ECCS Quick Response Head; This Modification Installs a Quick Responding Sprinkler Head in the Unit 1 ECCS Room Above Valves 1-E11-F008 and 1-E51-F008.

FUNCTIONAL SUMMARY: In accordance with the requirements of 10CFR50, Appendix R, "...Licensees should reexamine those previously approved configurations of fire protection that do not meet the requirements specified in Section III.G to Appendix R...." A detailed reexamination and reanalysis of the Brunswick safe shutdown capability has been performed and contained in the ASCA Report. Based on this report, Brunswick Steam Electric Plant concluded it can meet the requirements of 10CFR50, Appendix R, Section III.G through a combination of alternative shutdown capability in accordance with Section III.G.3 requirements and separation of redundant functions in accordance with III.G.2.

10CFR50, Appendix R, Section III.G requires fire protection features capable of limiting fire damage so that one train of systems necessary to achieve and maintain hot shutdown conditions is free of fire damage, and systems necessary for cold shutdown can be repaired in 72 hours.

One of the requirements of Appendix R, and as committed to in the BSEP ASCA Report, is fulfilled by installing one quick-responding sprinkler head in the unit's ECCS room above the RHR common suction valve (1-E11-F008) and the RCIC steam isolation valve (1-E51-F008). This one head provides sprinkler coverage for both of these valves.

The hot shutdown components of concern are the electrical cables and motor operators for the HPCI and RCIC steam isolation valves (1-E41-F003 and 1-E51-F008), respectively. These valves are separated horizontally by at least 11 1/2 feet of space which is free of exposed fixed combustible material. The cold shutdown components of concern are the outboard RHR shutdown cooling common suction valve (1-E11-F008), and the inboard isolation valves for the RHR shutdown cooling injection lines for both A and B RHR loops (1-E11-F015A and B). The cold shutdown valves are provided with approximately 26 feet of separation with the exception of the common RHR suction valve (1-E11-F008). A single fire in the vicinity of the RHR common suction valve (1-E11-F008) could potentially render both trains of RHR inoperable.

This modification installs a single quick-responding sprinkler head to provide coverage for both the RCIC steam isolation valve (1-E51-F008) and the RHR common suction valve (1-E11-F008). The combination of the passive cable protection (other modifications) and the increased speed of suppression of the quick-responding head provides reasonable assurance that at least one train of the hot shutdown components and one train of the cold shutdown components will be maintained in event of a fire.

The quick-responding sprinkler head is temperature activated (200°F) and will supply a minimum of 0.15 gpm per square foot of coverage. This results in an area coverage of 130 square feet. This head is UL listed for use in commercial applications and utilizes state-of-the-art technology for fast response (5-10 times faster than conventional heads). This increased speed of suppression meets the intent of the design as contained in the ASCA Report.

The piping to this head is a continuation of the piping to be installed in PM 84-103, which originates downstream of flow switch FS-3971 and continues along the east wall outside of the ECCS room. This modification ties into PM 84-103 piping, penetrates the east wall and continues inside. The design flows for the quick-responding head in PM 84-105 are not expected to change significantly due to any upstream piping changes that may occur for PM 84-103 and will meet the design intent and criteria. Should any significant changes become necessary, this modification will be revised to reflect the changes.

All pipe components are FP-Q. The piping will be supported by seismically designed (nonsafety) supports. Support material is procured as FP-Q per plant practice. The additional loads from these supports will not adversely affect the structural integrity of the existing slabs/walls to which they are attached. The grout for the penetration seal in the east wall of the ECCS room is procured as OTSQ for use in the FP-Q application, since the wall is a fire barrier. The core bore is through the safety-related wall and will have no adverse effects on the structure.

NOTE: This modification is not to be installed until installation of the following piping in PM 84-103 is complete. (Line 1-FP-371-3-J-2, 1-FP-371-2 1/2-J-2, and 1-FP-370-1 1/4-J-2). The piping system contained in this modification is tied into the piping system previously installed in PM 84-103. The Responsible Engineer may elect to perform acceptance testing for PM 84-103 and PM 84-105 concurrently or individually, based on plant conditions and engineering judgement.

CAUTION: If the PM 84-103 piping is not completely installed, assure that all open piping is temporarily capped prior to acceptance testing.

SAFETY SUMMARY: FSAR, Section 7.3.1.1.6.7, will be changed to reflect that a quick-responding sprinkler is added in the RCIC steam line tunnel area which would actuate in the event of a significant steam line break occurring in the room. Fire sprinkler actuation will be annunciated in the Control Room. Manual operator action is then required to determine the cause of sprinkler actuation and isolate the RCIC steam line as required. FSAR, Section 9.5, will be revised to include the new installation to add locational and descriptive information.

The addition of a quick-responding sprinkler head in the RCIC steam tunnel area will not increase the probability of occurrence of any accident previously evaluated in FSAR, Chapter 15. Main steam line break accidents outside the drywell and Reactor Building are addressed (15.6.3), but not steam line breaks in the Reactor Building such as could occur in the RCIC steam tunnel area.

Further, the sprinkler head provides additional protection for safety components in the event of a fire in the area, which provides reasonable assurance that operability of these components is maintained. A flooding analysis has been performed to verify safety systems will not be affected.

The consequences of accidents previously evaluated in the FSAR (Chapter 15) will not be increased. The ability of an operator to obtain information regarding the presence of an abnormal occurrence in the RCIC steam tunnel is enhanced by the presence of the sprinkler system which would actuate only if substantial steam leakage is occurring or in the event of a fire.

The probability of occurrence of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The function of the equipment important to safety in the RCIC steam line tunnel area is unaffected by the actuation of the sprinkler system unless either a significant steam leak (> 3,067 lb/hr) or a fire exists. Should the sprinkler system actuation be caused by a steam leak of this size, it will be required that operator actions be initiated to isolate RCIC.

The installation provides additional fire protection as required by the ASCA Report. A flooding analysis has been preformed to verify safety systems will not be affected.

The consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The RCIC stream leak detection instruments in the area will maintain their function as described in the FSAR until a substantial (> 3,067 lb/hr) postulated steam leak exists.

The consequences of sprinkler actuation at steam leak rates above this value are acceptable to the RCIC steam leak detection equipment since:

A. Sprinkler actuation will be annunciated in the Control Room providing operator knowledge of abnormal conditions in the area. Prior to sprinkler actuation, RCIC steam leak detection instrumentation will have initiated Control Room alarms and control logic functions.

B. Sprinkler actuation is indicative of an abnormal event being either a fire or a substantial steam leak in which case prompt operator actions will be required.

The installation will also provide increased fire protection capability as required by ASCA and provide a reasonable assurance that operability of the components to be protected is maintained. A flooding analysis was performed to verify safety systems will not be affected.

The probability of an accident or possibility for malfunction of equipment important to safety of a different type than already evaluated in the FSAR will not be created.

This installation provides additional fire protection and does not increase the possibility of an accident.

The RCIC steam tunnel area steam leak detection instrumentation will operate as designed unless a substantial (> 3,067 lb/hr) steam leak exists in the room. Should this occur, operator action is required. Annunciator procedures will be revised to address the effects of sprinkler head actuation on the RCIC steam leak detection system. A flooding analysis has been performed to verify safety systems will not be affected.

The margin of safety as defined in the basis to technical specifications is not reduced. The RCIC steam leak detection instrumentation is technical specification related, but it has been shown to be unaffected by the sprinkler installation unless a substantial (> 3,067 lb/hr) steam leak is present in the room. The annunciator procedures will be revised to address the effects of sprinkler actuation on the RCIC steam leak detector instruments in the area. The margin of safety is increased, due to enhancement of fire protection for components in the area.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-111, Unit 1 Fire Stops and Cable Coating (Reactor Building); Install Fire Stops and Coat Cables in Unit 1 Reactor Building, Elevation 50.0'

FUNCTIONAL SUMMARY:

In accordance with the requirements of 10CFR50, Appendix R, ... Licensees should reexamine those previously approved configurations of fire protection that do not meet the requirements specified in Section III.G to Appendix R..." A detailed reexamination and reanalysis of the Brunswick safe shutdown capability has been performed and contained in the ASCA Report. Based on this report, Brunswick Steam Electric Plant concluded it can meet the requirements of 10CFR50, Appendix R, Section III.G, through a combination of alternative shutdown capability in accordance with Section III.G.3 requirements and separation of redundant functions in accordance with Section III.G.2.

10CFR50, Appendix R, Section III.G, requires fire protection features capable of limiting fire damage so that one train of systems, necessary to achieve and maintain hot shutdown conditions, is free of fire damage, and systems necessary for cold shutdown can be repaired in 72 hours.

One of the requirements of Appendix R, and as committed to in the BSEP ASCA report, is fulfilled by installing fire stops for cables that are fixed intervening combustibles between redundant safe shutdown trains.

The following vertical cable trays penetrating the slab between the 50' and 80' elevations of the Reactor Building were inspected and found to be adequately sealed:

Tray No.	Location
44Q/CA	Col. S-4R
44P/BA	Col. S-4R
44N/DA	Col. S-4R
43P/DB	Col. L-7R, 8R
43P/CB	Col. L-7R, 8R
43P/BB	Col. L-7R, 8R

Redundant safe shutdown systems are located on the 50' elevation of the Unit 1 Reactor Building. These systems are represented by trains A and B reactor instrument racks located on the northeast and southeast sides of the drywell wall. With the 20' separation provided on the east side of the 50' elevation, the only feasible propagation path for a fire which could threaten both trains, is across the south side of elevation 50' 0". To further reduce the possibility of a fire involving both redundant trains, a fire stop will be installed for the exposed cables in the south center zone of elevation 50'. This modification package will provide for the installation of a fire stop consisting of one-inch thick marinite board secured to a unistrut channel frame and cut to fit around the several levels of horizontal cable trays. The voids between the marinite board, cables, and cable tray will be packed with ceramic fiber (kaowool). All cables will be coated for a length of 5' on each side of the marinite board with a flame retardant material (flame safe).

The support frame for the fire stop assembly is considered non-Q and has been designed in accordance with seismic criteria in order to preclude the possibility of the barrier being dislodged and causing damage to adjacent safety-related systems or components.

SAFETY SUMMARY:

The fire stop installed in this modification package does not raise any unresolved safety questions. The FSAR will be updated to document the installation of the new fire stop. Existing technical specifications are adequate with no changes required.

Derating of cable ampacity due to the thermal insulating qualities of the flame retardant coating is not a factor based on the Factory Mutual Test Report for Thomas and Batts flame safe fire retardant coating.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-129, Unit 1 Sprinkler Installation (50.0' Elevation, Reactor Building); Installation of a Closely Spaced Line of Closed Head Sprinklers Adjacent to Draft Stops in the Unit 1 Reactor Building, Elevation 50.0'

FUNCTIONAL SUMMARY: In accordance with the requirements of 10CFR50, Appendix R, "...Licensees should reexamine those previously approved configurations of fire protection that do not meet the requirements specified in Section III.G to Appendix R..." A detailed reexamination and reanalysis of the Brunswick safe shutdown capability has been performed and is contained in the alternative shutdown capability assessment (ASCA) report. Based on this, the Brunswick Steam Electric Plant concluded it can meet the requirements of 10CFR50, Appendix R, Section III.G, through a combination of alternative shutdown capability in accordance with Section III.G requirements and redundant functions in accordance with Section III.G.2.

10CFR50 Appendix R, Section III.G, requires fire protection features capable of limiting fire damage so that one train of systems necessary to achieve and maintain hot shutdown conditions is free of fire damage, and systems necessary for cold shutdown can be repaired in 72 hours.

One of the requirements of Appendix R, and as committed to in the BSEP ASCA report, is fulfilled by installing water curtains in conjunction with draft stops within the separation zone to prevent propagation of combustion products across the zone.

The Reactor Buildings are divided into two halves along an east/west line with one train of safe shutdown systems located in the northern half of the building and the other located in the southern half. Physical separation between the two halves is provided by the inerted drywell, torus, steam tunnel, ECCS room, and HPCI room. Where such physical separation does not exist on the 50-foot elevation, separation zones of 20-foot width free of significant quantities of intervening combustibles are provided.

These separation zones will be provided with sprinklers utilizing the guidance of NFPA-13, Section 4-4.8.2.3. This section is concerned with the prevention of fire spread through large floor openings. To achieve this objective, lines of closely spaced, closed head sprinklers are utilized in conjunction with draft stops within the bounds of the separation zone.

Although this area is not a floor opening, the objective of limiting fire spread past a given vertical plane is the same. The direction of fire spread cannot be determined in advance, thus, a sprinkler/draft stop configuration will be provided to limit fire spread from north to south and south to north. Existing concrete beams are utilized as draft stops. One sprinkler line/draft stop combination will be oriented to prevent fire spread from north to south. The other sprinkler line/draft stop combination will prevent fire spread from south to north. Existing concrete beams and structural members will function as baffles between the heads to prevent one sprinkler from spraying or "cold soldering" the adjacent sprinklers. The sprinkler heads are temperature actuated closed heads (165°F) and will supply a minimum of 3 gpm/linear foot of water curtain. Sprinkler spacing is at a maximum of 6 ft (18 gpm), and no sprinkler will discharge less than 15 gpm.

The piping system is an extension of the existing area suppression system in the area. The tie-in is downstream of existing flow switch FS-3973. The water curtain is hydraulically calculated and designed. All pipe components are FP-Q. The piping will be supported by seismically designed (nonsafety) supports. Support material is procured as FP-Q per plant practice. The additional loads from these supports will not adversely affect the structural integrity of the existing slabs, walls, to which they are attached.

Existing plant drawing F-24003 is being changed to correct a valve number. The valve number was field verified to be 1-FP-V389, and this drawing change will agree with the valve number in the OP and other plant documents.

SAFETY SUMMARY: The additional water suppression system has no effect on the probability of occurrence or consequences of any accident or equipment malfunction previously evaluated in the FSAR, nor does it affect the margin of safety as defined in the basis to any technical specification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-130, Unit 2 Sprinkler Installation (50.0' Elevation, Reactor Building); Installation of a Line of Closely Spaced Closed Head Sprinklers Adjacent to Draft Stops in the Unit 2 Reactor Building, Elevation 50.0'

FUNCTIONAL SUMMARY: In accordance with the requirements of 10CFR50, Appendix R, "...Licensees should reexamine those previously approved configurations of fire protection that do not meet the requirements specified in Section III.G to Appendix R..." A detailed reexamination and reanalysis of the Brunswick safe shutdown capability has been performed and is contained in the Alternative Shutdown Capability Assessment (ASCA) Report. Based on this, the Brunswick Steam Electric Plant concluded it can meet the requirements of 10CFR50, Appendix R, Section III.G, through a combination of alternative shutdown capability in accordance with Section III.G requirements and redundant functions in accordance with Section III.G.2.

10CFR50, Appendix R, Section III.G, requires fire protection features capable of limiting fire damage so that one train of systems necessary to achieve and maintain hot shutdown conditions is free of fire damage, and systems necessary for cold shutdown can be repaired in 72 hours.

One of the requirements of Appendix R, and as committed to in the BS[illegible]CA Report, is fulfilled by installing water curtains in conjunction with [illegible]raft stops within the separation zone, to prevent propagation of combustion products across the zone.

The Reactor Buildings are divided into two halves along an east/west line with one train of safe shutdown systems located in the northern half of the building, and the other located in the southern half. Physical separation between the two halves is provided by the inerted drywell, torus, steam tunnel, ECCS room, and HPCI room. Where such physical separation does not exist on the 50-foot elevation, separation zones of 20-foot width free of significant quantities of intervening combustibles are provided.

These separation zones will be provided with sprinklers utilizing the guidance of NFPA-13, Section 4-4.8.2 3. This section is concerned with the prevention of fire spread through large floor openings. To achieve this objective, lines of closely spaced, closed head sprinklers are utilized in conjunction with draft stops within the bounds of the separation zone.

Although this area is not a floor opening, the objective of limiting fire spread past a given vertical plane is the same. The direction of fire spread cannot be determined in advance, thus, a sprinkler/draft stop configuration will be provided to limit fire spread from north to south and south to north. Existing concrete beams are utilized as draft stops. One sprinkler line/draft stop combination will be oriented to prevent fire spread from north to south. The other sprinkler line/draft stop combination will prevent fire spread from south to north. Existing concrete beams and structural members will function as baffles between the heads to prevent one sprinkler from spraying or "cold soldering" the adjacent sprinklers. The sprinkler heads are temperature actuated closed heads (165°F) and will supply a minimum of 3 gpm/linear foot of water curtain. Sprinkler spacing is at a maximum of 6 ft (18 gpm), and no sprinkler will discharge less than 15 gpm.

The piping system is an extension of the existing area suppression system in the area. The tie-in is downstream of existing flow switch FS-J973. The water curtain is hydraulically calculated and designed. All pipe components are FP-Q. The piping will be supported by seismically designed (nonsafety) supports. Support material is procured as FP-Q per plant practice. The additional loads from these supports will not adversely affect the structural integrity of the existing slabs, walls, to which they are attached.

SAFETY SUMMARY: The addition of protective water curtains has no negative impact on safety-related equipment, nor does it increase the probability or consequences of any accident previously evaluated in the FSAR. The margin of safety as defined in the basis to the technical specifications will actually be increased due to the enhancement of the separation zone between safety trains with water suppression.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-135, Appendix R, Emergency Lighting, Unit 1 Turbine Building; Emergency Lights Will be Added or Relocated to Provide Additional Lighting for Access/Egress Routes for Compliance With CP&L's Commitment to 10CFR50, Appendix R

FUNCTIONAL SUMMARY: The purpose of this modification is to partially fulfill the requirements of 10CFR50, Appendix R, Section III.J, concerning emergency lighting as defined by CP&L's Alternative Shutdown Capability Assessment (ASCA). The Unit 1 Turbine Building breezeway and controlled access corridor require additional emergency lighting for access/egress routes upon loss of normal lighting. Emergency lighting units will sense the loss of normal 120 Vac lighting power and shall be automatically energized. The emergency lighting units will be driven by a 6 Vdc self-contained battery pack and shall be able to maintain power for a minimum of eight hours. Each lamp assembly consists of two 12 watt halogen lamps, or where required, four 6 watt weatherproof fixtures.

The design for this modification conforms to the original design/installation codes, standards, and specifications. Therefore, no degradation from the original system design/installation requirements is being introduced as a result of this modification. This modification will provide the required emergency lighting for the areas described below. The adequacy of illumination and lamp positioning will be verified by Operations.

Emergency lighting units will be installed as follows:

A. Unit 1 Turbine Building Breezeway, Elevation 20' -0"; See Sketch SK-E-84-135-01

   1. One new emergency lighting unit, EL-B7, fed from panel 1T2, circuit 9, shall be located as shown on sketch.

   2. One existing lighting unit, EL-B5 (lamps only), fed from panel 1T2, circuit 11, shall be relocated as shown on sketch.

B. Unit 1 Turbine Building Access Corridor, Elevation 45' -0"; See Sketch SK-E-84-135-01

   1. One new emergency lighting unit EL-TAC1, fed from panel 1T6, circuit 13, shall be located as shown on sketch. One core bore through a non-Q exterior wall is required for the remote mounting of four weatherproof fixtures.

   2. One new emergency lighting unit, EL-TAC2, fed from panel 1T6, circuit 18, shall be located as shown on sketch.

   3. One new emergency lighting unit, EL-TAC3, fed from panel 1TE2, circuit 1, shall be located as shown on sketch.

All conduits shall be field run. Emergency lighting and conduit supports will be designed and installed in accordance with drawing 9527-6-3576, Appendix A, Nonseismic Conduit and Electrical Box Supports and Seismic, Self-Contained DC Emergency Lighting System Supports, revision 1 and later.

SAFETY SUMMARY: The emergency lighting units do not affect operation of any safety-related equipment evaluated in the FSAR. The units would not increase the probability of occurrence or the consequences of any accident previously evaluated in the FSAR. This modification would not reduce the margin of safety as defined in the basis to any technical specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-137, Appendix R Emergency Lighting, Unit 1 Reactor Building; Emergency Lighting Units and Hand Held Units Will be Added to Provide for Access/Egress Lighting for Compliance With CP&L's Commitment to 10CFR50, Appendix R

FUNCTIONAL SUMMARY: The purpose of this modification is to partially fulfill the requirements of 10CFR50, Appendix R, Section III.J, concerning emergency lighting as defined by CP&L's alternative shutdown capability assessment (ASCA). The Unit 1 Reactor Building contains safe shutdown equipment which requires additional emergency lighting for access/egress routes upon loss of normal lighting. Emergency lighting units will sense the loss of normal 120 Vac lighting power and shall be automatically energized. The emergency lighting units will be driven by a 6 Vdc self-contained battery pack and shall be able to maintain power for a minimum of eight hours. Each lamp assembly consists of two 12 watt halogen lamps.

The design for this modification conforms to the original design/installation codes, standards, and specifications. Therefore, no degradation from the original system design/installation requirements are being introduced as a result of this modification. This modification will provide the required emergency lighting for the areas described below. Adequacy of illumination and lamp position will be verified by Operations.

Emergency lighting units will be installed as follows:

A. Unit 1 Reactor Building, Elevation (-)17'-0"; See Sketch SK-E-84-137-01

1. One new emergency lighting unit, EL-1R15, fed from panel 1RE1, circuit 16, shall be located as shown on sketch.

2. One new emergency lighting unit, EL-1R16, fed from panel 1R2, circuit 12, shall be located as shown on sketch.

3. One new emergency lighting unit, EL-1R17, fed from panel 1R2, circuit 10, shall be located as shown on sketch.

4. One new emergency lighting unit, EL-1R18, fed from panel 1R2, circuit 16, shall be located as shown on sketch.

5. One new emergency lighting unit, EL-1R19, fed from panel 1R2, circuit 2, shall be located as shown on sketch.

6. One new emergency lighting unit, EL-1R35, fed from panel 1RE1, circuit 5, shall be located as shown on sketch.

B. Unit 1 Reactor Building, Elevation 20'-0"; See Sketch SK-E-84-137-02

1. One new emergency lighting unit, EL-1R20, fed from panel 1RE1, circuit 11, shall be located as shown on sketch.

2. One new emergency lighting unit, EL-1R21, fed from panel 1R1, circuit 27, shall be located as shown on sketch.

3. One new emergency lighting unit, EL-1R22, fed from panel 1R2, circuit 22, shall be located as shown on sketch.

4. One new emergency lighting unit, EL-1R23, fed from panel 1RE1, circuit 16, shall be located as shown on sketch.

5. One new emergency lighting unit, EL-1R24, fed from panel 1RE1, circuit 16, shall be located as shown on sketch.

6. One new emergency lighting unit, EL-1R25, fed from panel 1RE1, circuit 23, shall be located as shown on sketch.

C. Unit 1 Reactor Building, Elevation 50'-0"; See Sketch SK-E-84-137-03

1. One new emergency lighting unit, EL-1R26, fed from panel 1RE2, circuit 3, shall be located as shown on sketch.

2. One new emergency lighting unit, EL-1R27, fed from panel 1R4, circuit 5, shall be located as shown on sketch.

3. One new emergency lighting unit, EL-1R28, fed from panel 1R4, circuit 6, shall be located as shown on sketch.

4. One new emergency lighting unit, EL-1R29, fed from panel 1R4, circuit 4, shall be located as shown on sketch.

5. One new emergency lighting unit, EL-1R30, fed from panel 1R4, circuit 4, shall be located as shown on sketch.

6. One new emergency lighting unit, EL-1R31, fed from panel 1R3, circuit 11, shall be located as shown on sketch.

7. One new emergency lighting unit, EL-1R32, fed from panel 1RE2, circuit 3, shall be located as shown on sketch.

8. One new emergency lighting unit, EL-1R33, fed from panel 1RE1, circuit 19, shall be located as shown on sketch.

9. One new emergency lighting unit, EL-1R34, fed from panel 1R3, circuit 3, shall be located as shown on sketch.

In addition to the emergency lighting units, five flashlights housed in a tool box will be located at the Unit 1 remote shutdown panel. During a fire or postfire condition, it may be necessary for an Auxiliary Operator to go to the east yard area. Due to the excessive number of emergency lights it would require for the operators' route and the lack of self-contained eight-hour emergency lighting units available for outdoor use, CP&L requested an exemption from the requirements of 10CFR50, Appendix R. The exemption stated that portable hand lights (flashlights) will be provided in the Control Room and at each remote shutdown panel. This modification will partially fulfill this commitment.

SAFETY SUMMARY: The emergency lighting units do not directly or indirectly affect operation of any safety-related systems evaluated in the FSAR. This modification would not increase the probability of occurrence or the consequences of any accident previously evaluated in the FSAR. The modification would not impact any equipment or increase the probability of any equipment malfunctions different that those previously evaluated in the FSAR.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-138, Appendix R, Emergency Lighting, Unit 2 Reactor Building; Emergency Lighting Units Added, Relocated, or Lamps Adjusted to Provide Lighting for Access/Egress Routes for Compliance With CP&L's Commitment to 10CFR50, Appendix R

FUNCTIONAL SUMMARY: The purpose of this modification is to partially fulfill the requirements of 10CFR50, Appendix R, Section III.J, concerning emergency lighting as defined by CP&L's alternative shutdown capability assessment (ASCA). The Unit 2 Reactor Building contains safe shutdown equipment which requires additional emergency lighting for access/egress routes upon loss of normal lighting. Emergency lighting units will sense the loss of normal 120 Vac lighting power and shall be automatically energized. The emergency lighting units will be driven by a 6 Vdc self-contained battery pack and shall be able to maintain power for a minimum of eight hours. Each lamp assembly consists of two 12 watt halogen lamps.

The design for this modification conforms to the original design/installation codes, standards, and specifications. Therefore, no degradation from the original system design/installation requirements are being introduced as a result of this modification. This modification will provide the required emergency lighting for the areas described below. Adequacy of illumination and lamp position will be verified by Operations.

Emergency lighting units will be installed as follows:

A. Unit 2 Reactor Building, Elevation (-)17'-0"; See Sketch SK-E-84-138-01

1. One new emergency lighting unit, EL-2R14, fed from panel 2RE1, circuit 16, shall be located as shown on sketch.

2. One new emergency lighting unit, EL-2R15, fed from panel 2R2, circuit 12, shall be located as shown on sketch.

3. One new emergency lighting unit, EL-2R16, fed from panel 2R2, circuit 10, shall be located as shown on sketch.

4. Lamp on one unit, EL-2R12, will be redirected up the ladder as shown on sketch.

B. Unit 2 Reactor Building, Elevation 20'-0"; See Sketch SK-E-84-138-02

1. One new emergency lighting unit, EL-2R17, fed from panel 2RE1, circuit 11, shall be located as shown on sketch.

2. One new emergency lighting unit, EL-2R18, fed from panel 2R1, circuit 25, shall be located as shown on sketch.

3. One new emergency lighting unit, EL-2R19, fed from panel 2R2, circuit 22, shall be located as shown on sketch.

4. One new emergency lighting unit, EL-2R20, fed from panel 2RE1, circuit 16, shall be located as shown on sketch.

5. One new emergency lighting unit, EL-2R21, fed from panel 2R6, circuit 9, shall be located as shown on sketch.

6. One new emergency lighting unit, EL-2R22, fed from panel 2RE1, circuit 16, shall be located as shown on sketch.

7. One new emergency lighting unit, EL-2R23, fed from panel 2RE1, circuit 23, shall be located as shown on sketch.

8. One existing emergency lighting unit, EL-2R9 (lamp only), fed from panel 2R6, circuit 9, shall be relocated as shown on sketch.

9. Lamps for two units, EL-2R2 and EL-2R6, will be redirected as shown on sketch.

C. Unit 2 Reactor Building, Elevation 50'-0"; See Sketch SK-E-84-138-03

1. One new emergency lighting unit, EL-2R24, fed from panel 2RE1, circuit 9, shall be located as shown on sketch.

2. One new emergency lighting unit, EL-2R25, fed from panel 2R4, circuit 5, shall be located as shown on sketch.

3. One new emergency lighting unit, EL-2R26, fed from panel 2R4, circuit 4, shall be located as shown on sketch.

4. One new emergency lighting unit, EL-2R27, fed from panel 2R4, circuit 4, shall be located as shown on sketch.

5. One new emergency lighting unit, EL-2R28, fed from panel 2R4, circuit 4, shall be located as shown on sketch.

6. One new emergency lighting unit, EL-2R29, fed from panel 2R3, circuit 11, shall be located as shown on sketch.

7. One new emergency lighting unit, EL-2R30, fed from panel 2RE2, circuit 3, shall be located as shown on sketch.

8. One new emergency lighting unit, EL-2R31, fed from panel 2RE2, circuit 5, shall be located as shown on sketch.

In addition to the emergency lighting units, five flashlights housed in a tool box will be located at the Unit 2 remote shutdown panel. During a fire or postfire condition, it may be necessary for an Auxiliary Operator to go to the east yard area. Due to the excessive number of emergency lighting units it would require for the operators' route and the lack of self-contained eight-hour emergency lighting units available for outdoor use, CP&L requested an exemption from the requirements of 10CFR50, Appendix R. The exemption stated that portable hand lights (flashlights) will be provided in the Control Room and at each remote shutdown panel. This modification will partially fulfill this commitment.

SAFETY SUMMARY: The emergency lighting units do not directly or indirectly affect operation of any safety-related systems evaluated in the FSAR. This modification would not increase the probability of occurrence or the consequences of any accident previously evaluated in the FSAR. The modification would not impact any equipment or increase the probability of any equipment malfunctions different than those previously evaluated in the FSAR.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-145, Appendix R Communications Upgrade; Installation of Three Sound-Powered Phone Circuits to Provide Reliable Communication Between Areas of the Reactor, Diesel, Service Water, and Control Buildings

FUNCTIONAL SUMMARY: This modification installs two additional sound-powered phone circuits to provide communication between Unit 1 Reactor, Diesel, Service Water, and Control Buildings in the event a fire disables the existing system. A common circuit linking Unit 1 and Unit 2 remote shutdown panels will also be installed.

Preliminary evaluation has shown that the existing sound-powered phone system is routed across both safe shutdown Trains A and B and thus, a fire could disable the entire system. The new installation will ensure that one train of the phone system will be available during alternative safe shutdown activities.

This modification is being implemented to comply with the requirements of 10CFR50, Appendix R.

SAFETY SUMMARY: A dedicated phone system for remote shutdown in the event of a fire reduces the probability and consequences of any previously evaluated accident. This modification has no affect on the probability or consequences of any equipment malfunction previously evaluated in the FSAR, nor does it affect any safety-related equipment. The margin of safety as defined in the technical specification would be increased by improved reliability.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-272, Fire Detector Modification CB Elevation 23' and 49'; Relocates Detector in Cable Spread Room (Unit 2), Installs New Detectors in AO Room and Southwest Corner 49' (Unit 2)

FUNCTIONAL SUMMARY: This modification makes changes in Control Building Fire Detector Zones C2, C5, and C9. Because of LCOs involved during installation and the extent of work to be performed, modification work will be divided into three separate parts (one per zone) and provisions provided for interim operability as required. Changes to be made are as per the following:

1. Control Building Fire Detector Zone 9

   1.1 There is an ionization detector (connected between C2-12 and C2-31) hanging from the overhead in the southwest corner of the Unit 2 cable spread room. This detector was utilized in a temporary structure located at the 23-ft level which has been removed. This detector will be removed along with its associated conduit and a new flame detector installed at ceiling level. The existing detector is not shown on any plant drawings. Therefore, a new detector number will be assigned (C2-35) and drawings marked-up to indicate changes.

2. Control Building Fire Detector Zone 2

   2.1 The following changes, to correct existing detector and conduit supports and to provide proper protection of the corner offices, will be made near the southwest corner of the EER area (49-ft level).

      2.1.1 Detector C2-4, which is suspended between electrical equipment cabinets with flex conduit, will be removed.

      2.1.2 Detector C2-5, which is located too close to the corner office wall, will be relocated on the EER room ceiling over the two corner office spaces.

      2.1.3 An additional ionization (smoke) detector will be installed at ceiling level in each corner office space.

      2.1.4 Existing detector C2-6 will be redesignated as C2-4.

3. Control Building Fire Detector Zone C5

   3.1 The Control Building area originally described as the electronic work room (49-ft level) was divided into two AO rooms with 8-ft solid ceilings by PM 84-379. The solid ceiling tiles have been removed because of an insufficient number of fire detectors in the area. This modification will install an additional detector at the 8-ft ceiling level of each room and also provide remote indication of detector C2-16. Installation of the new detectors in this area will provide fire detection coverage required for reinstallation of the solid ceiling tiles.

4. Fire detector wiring and conduit are installed independent of other plant systems and are not part of or tracked by BSEP CASP. Therefore, the following will apply:

   4.1 Cable pull slips are not provided. Routing will be per modification instructions.

   4.2 Cable and conduit are not identified per Specification 048-5 (ID per Specification 048-5 is assigned by CASP). Identification will be per modification instructions.

   4.3 With the exception of identification markings and documentation, detectors and conduit supports will be designed and installed per Specification 048-10 fire protection Q, seismic requirements (support identification and documentation is based on CASP data). Support identification inspections and documentation will be per modification instructions.

SAFETY SUMMARY: The equipment being installed meets all FP-Q requirements. The function of the system is not changed nor is any other system function affected. The CKTs in which the new detectors are added are not functionally changed. Detectors are the same type now used at BSEP. Maintenance nor surveillance requirements are not reduced for safety-related components. Redundancy of safety-related systems is not affected by this modification. Equipment used is consistent with present equipment. No single activity failure nor separation criteria violation will result because of this modification. The function of the system is not changed and equipment used is the same type used at BSEP now. Therefore, no new type accident could be created by this modification. An increase in the number of fire detectors will improve fire detection capability. No new type equipment is introduced.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-337, HVAC Battery Room, Unit 1; Supply Duct Heating to Battery Rooms 1A and 1B

FUNCTIONAL SUMMARY: This modification introduces a duct heater into the HVAC supply duct for each battery room. The heater will, through thermostatic control, maintain battery room temperature above 77°F. The heater will also have a flow control switch which will trip the heater on loss of flow.

Temperature control for each room is accomplished by temperature-controlled vortex dampers located in the supply duct. IEEE 484-1981 recommends 77°F as an optimum temperature for battery life and capacity. Below an electrolyte temperature of 77°F, a correction factor must be applied to the hydrometer reading for the battery to obtain the specific gravity. The current technical specification limits for specific gravity are being increased and a 77°F temperature must be maintained to ensure technical specification operability. The vortex dampers alone are not enough to maintain the temperature. A duct heater is being placed in the supply duct downstream of the supply fan for each battery room. The heaters are sized at 25 kW to accommodate a rise in temperature of 45°F.

This equipment to be used in conjunction with the existing battery room ventilation system. It will augment the vortex dampers in ensuring room temperature is maintained at a sufficient level to meet technical specification requirements for battery specific gravity.

SAFETY SUMMARY: This modification does not affect any equipment or raise any safety concern on a possible accident or malfunction of equipment previously evaluated in the FSAR. This modification does not change any technical specification margin of safety.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-005, CAC Purge and Vent Valve Rework; Add Automatic CAC Purge and Vent Valve Isolation From Main Stack High Radiation Monitor

FUNCTIONAL SUMMARY:

The purpose of this plant modification is to provide drywell vent and purge valve isolation on primary containment high radiation signal in accordance with NUREG-0737, Item II.E.4.2.7.

The existing main stack radiation monitor will be used to originate the isolation signal. The Unit 2 modification 86-006 will be operable or partially operable before the Unit 1 modification 86-005 is completed. Unit 2 modification 86-006 will remove the alarm circuit from the main stack radiation monitor hi-hi trip relay and connect a control relay (3-55) to the hi-hi trip relay N.O. contacts. Unit 2 relay 3-55 will provide the hi-hi alarm and control the trip circuit for Unit 2 (relay 3-56) and Unit 1 (relay 3-55). Since the radiation monitor is common to both units, an override switch is provided on each unit to bypass the trip after it has been determined that the drywell purge and vent valves on that unit are not the cause of the high radiation trip.

The Unit 1 trip relay (3-55) will be mounted in Unit 1 Control Room cabinet XU-53 and will be powered from a separately fused circuit off of the CAC isolation trip override circuit. Unit 1 trip relay 3-55 is controlled by Unit 2 relay 3-55. Unit 1 relay 3-55 contacts are normally open. The relay will be energized during normal operation so the contacts will be closed. Contacts 1 and 2 will be wired in series with upscale trip contact of Reactor Building exhaust radiation monitor K609A and auxiliary trip unit C51A-22A relay K83 contacts 6 and 10. If any of these three contacts open, relay K82 in the auxiliary trip unit will deenergize, which initiates a primary containment group 6 isolation. Contacts 3 and 4 of relay 3-55 are connected to the B loop radiation monitor in the same manner as 1 and 2 are connected to the A loop.

Units 1 and 2 CAC Purge and Vent Valve Isolation Logic From OG Stack High Radiation

During normal operation, the main stack radiation monitor hi-hi contacts are closed and the following relays are energized: Unit 2--3-55, 3-56, K83A and B, K82 in A and B loops; Unit 1--3-55, K83A and B, K82 in A and B loops.

If the stack radiation monitor reaches the hi-hi setpoint, the contacts will open deenergizing the following relays: Unit 2--3-55, 3-56, K82 in A and B loops; Unit 1--3-55, K82 in A and B loops. This will give both units an OG vent pipe hi-hi alarm and a group 6 isolation. The isolation can be bypassed on Unit 2 by placing the override switch in override which will energize relay 3-56 and K82 in A and B loops. The isolation can be bypassed on Unit 1 by placing the override switch in override which will energize relay 3-55 and K82 in A and B loops.

SAFETY SUMMARY:

This modification gives an additional trip input to a safety-related system. It does not change the operation or function of the system. It would not affect the probability or consequences of an accident or equipment malfunction and would not decrease the margin of safety as defined in the basis to any technical specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-011, Unit 1, Replace Obsolete UPS System; Replace Existing 37.5 KVA UPS System With New 50 KVA Equipment.

FUNCTIONAL SUMMARY: The Unit 1 Uninterruptible Power Supply (UPS) System feeds various loads throughout the plant, including equipment located in the electronic equipment room, the computer room, the circulating water area, and the filter house.

The purpose of this modification is to upgrade the existing Unit 1 UPS System equipment as a result of operational concerns regarding the reliability of the UPS System and its impact on the integrity of the vital loads that it serves. The existing UPS equipment is obsolete and beyond reasonable maintenance. Spare parts from the original vendor (Static Products) are no longer available, which puts an unusual burden on the maintenance staff.

The work in this modification represents a one-for-one direct replacement for existing UPS equipment in the battery rooms. The provisions for ready access to spare parts will reduce maintenance time such that equipment availability will be increased.

The existing 37.5 KVA UPS System, which includes a primary power converter unit, a standby power converter unit, and a switching module, will be replaced with new equipment rated at 50 KVA and with the same functional capabilities. Much of this modification will be performed during a Unit 1 outage period while the primary power converter is acting as the in-service unit. The existing conduit and wiring will be disconnected from the existing equipment and then be reconnected to the new units.

Procurement Specification No. 106-003 for the UPS equipment includes a requirement that the incoming alternate feed breaker on the standby unit be supplied as a GE mag break type TEC 150 amp trip with an adjustable instantaneous pick-up which will be set at a value of approximately 1520 amps as recommended under calculation set 82125-E-160-F. This calculation was prepared (by NELD) in order to meet the requirements of 10CFR50, Appendix R, Section III.G, in that adequate protective device coordination must be provided for the portion of the electrical distribution system that serves dedicated shutdown loads. Plant Modification 84-123 was prepared in order to implement the recommendations of this calculation. Because of the provisions made in the procurement specification for the new UPS equipment, the implementation of Plant Modification 84-123 is no longer required.

Various system changes that will be incorporated in this modification are as follows:

A. The existing 70 amp, 480-volt feeder breaker, which supplies the normal power to the primary and standby units from MCC 1CA Compt. C07 and MCC 1CB Compt. C55 respectively, will be replaced with new 150 amp breakers based on the recommendations made in the calculation contained in the Design Analysis section of this modification.

B. The existing 60 amp 480-volt feeder breaker which supplies the alternate feed to the UPS System from MCC 1CB Compt. C69 will be replaced with a new 70 amp breaker based on the recommendation made in the calculation contained in the Design Analysis section of this modification.

C. The existing 4/C #6 480-volt feeder cables from MCC 1CA and 1CB to the primary and standby units will be replaced with new 4/C #2 feeder cables based on the recommendations made in the calculation contained in the Design Analyses section of this modification.

D. The existing 4/C #4 480-volt feeder cable to the 45 KVA alternate feed transformer will be replaced with a new 4/C #2 feeder cable based on the recommendation made in the calculation contained in the Design Analysis section of this modification. The feeder from the transformer to the primary UPS unit will also be replaced and rerouted to the new standby UPS unit.

Replacing the existing 37.5 KVA UPS equipment with new equipment rated at 50 KVA will impose an additional load on the existing 125/250-volt battery system due to the reduced efficiency factor associated with this larger unit. This additional load was evaluated by BESU against the BSEP Battery Load Study (reference: BEM-14562, dated [illegible]/20/86), and it was concluded that there was an adequate capacity margin to accommodate this additional load.

The installation of the new UPS equipment will be performed in several stages and in a sequence that will minimize the impact on the existing plant systems. All work, except for the modifications to the existing equipment pads and installation of the conduit supports, must be performed during an outage period. A summary of the construction sequence is as follows:

A. Replace Existing Standby Unit

During this period of construction, there will be no planned service interruption of the vital AC bus (non-Q), and the 45 KVA transformer hard source is available as an alternate feed with automatic transfer capability.

Upon completion of the acceptance for the new standby unit and after interim operability signatures have been obtained, this unit will be available for service during the remaining portions of this modification.

B. Install Temporary Feeder

A temporary power feeder will be installed from the new standby UPS Unit 1B to the UPS distribution panel 1A in accordance with ENP-14. A spare 100 amp branch circuit switch at UPS distribution panel 1A will be provided with a set of fuses and be utilized to serve the temporary incoming feeder cable.

During this period of construction, there will be no planned service interruption of the vital (non-Q) AC bus. The 45 KVA transformer hard source will continue as an alternate source to the primary unit with automatic transfer capability.

C. Reroute Existing Alternate Feed Cable

During this period of construction, there will be no planned service interruption to the vital (non-Q) AC bus. While the permanent alternate feed circuit is being modified, the new standby UPS unit is available as an alternate source, via the temporary feeder cable, to the in-service primary unit with no automatic transfer capability. On a failure of the existing primary unit, the load would be transferred manually to the standby unit by opening the permanent incoming main breaker at distribution panel 1A and closing the temporary incoming fused switch that was provided earlier.

Upon completion of the installation of the new alternate source cable, interim operability signatures will be obtained so that the new standby unit will have the capability of automatic transfer to the hard source for the balance of the work to be performed in this modification.

D. Transfer UPS Load From Primary Unit to New Standby UPS Unit

At UPS distribution panel 1A, the permanent incoming main breaker is opened and the fused switch serving the temporary incoming line will be closed.

The UPS loads are now being supplied by the new standby unit through the temporary feeder so that the primary unit can be replaced. This transfer will result in a planned service interruption with a duration of only a few seconds. The installation procedures provide for appropriate notification prior to this planned service interruption.

E. Replace Existing Primary UPS Unit

During this period of construction, there will be no planned service interruption to the vital (non-Q) AC bus. The UPS loads will be supplied from the new standby UPS unit through the temporary feeder with the hard source available with automatic transfer capability.

F. Transfer UPS Load to New Primary Unit

At UPS distribution panel 1A, the fused switch serving the temporary incoming line will be opened and the permanent incoming main breaker will be closed.

At this point, there will be a temporary loss of power to the UPS loads (with a duration of approximately 4 hours) while the final transfer to the primary unit is being made and acceptance testing is completed. Appropriate notification is provided for prior to this planned service interruption. On completion of the transfer, the UPS loads will be supplied from the new primary unit. However, the alternate feed source will not be available until the temporary feeder cable is disconnected at the standby unit, and the cross-tie cable between the primary and standby unit is connected at the standby unit.

G. Remove Temporary Feeder

At this point in the construction sequence, the temporary feeder cable will be removed and the final tie-in between the primary and standby units will be made. The opening in the existing fire barrier that was provided so the temporary feeder could be routed will be resealed and the final operability, including the previous interim operabilities, will be initiated.

Interim operabilities will be declared throughout the installation/construction process. This will enable the completed portions of the system to be put back into service while work continues on the rest of the system. This is being done in order to minimize the service interruptions to the UPS loads. Interim operabilities will be declared at the following steps:

1. Upon completion of the installation and testing of the new standby unit.

2. Upon completion of the installation and testing of the new UPS System alternate feed circuit.

3. Upon completion of the installation and testing of the new primary unit.

The installation of the new 3-inch conduit using an existing conduit penetration in the common battery room wall represents Fire Protection-related work since the existing cap on the conduit sleeve must be removed in order to extend the conduit on both sides of the fire barrier wall. The temporary power feeder through the cable spreading room fire barrier wall will also represent fire protection-related work.

The safety-related work in this modification includes the circuit breaker replacement and wiring terminations at safety-related MCC 1CA Compt. C07 and MCC 1CB Compt. C55 and C69. The feeder breakers at the MCCs represent the Q/non-Q boundary between the safety-related MCCs and the nonsafety-related UPS System. Equipment and material as listed on the Bill of Material will be procured as Q-list material.

All other work contained in this modification package is nonsafety-related.

SAFETY SUMMARY: No new anticipated operationa. occurrences or postulated accidents will be introduced as a result of the work contained in this modification. The new UPS equipment represents a direct one-for-one replacement of the existing outdated equipment. The limits of this system are being neither expanded nor reduced.

Since the new UPS equipment is a direct replacement of the existing system, no additional consequences of an accident other than those previously evaluated in the FSAR could be introduced.

The UPS equipment, which is fed from safety-related MCC buses, supplies regulated uninterruptible power to nonsafety-related loads. The feeder breakers at the MCCs represent the Q/non-Q boundary between the safety-related MCCs and the non-Q UPS System. The occurrence of a malfunction of this equipment will not affect any portion of a system that is important to the safe operation of the plant. The new UPS equipment represents a direct one-for-one replacement of the existing outdated equipment.

The consequences of a malfunction of the new UPS equipment have been reduced by providing protective device coordination in the alternate feed circuit and resizing the normal AC power feeders so that these feeders, which are fed from the emergency buses, are protected against thermal damage due to short circuit conditions.

Replacing the existing obsolete equipment with new state-of-the-art components enhances the reliability of the system, thereby reducing the overall possibility of any equipment malfunction.

Because the basic design and operational capabilities remain unchanged, the margin of safety as defined in the basis to the technical specifications is not reduced. The BSEP DC Load Study was reviewed to determine that adequate margin exists.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-021, Feedwater Temperature Monitoring; Installation of a Fast Time Response Temperature Element Into Each Feedwater Line to Monitor System Temperatures Downstream of RWCU Return to Reactor

FUNCTIONAL SUMMARY: This plant modification, 86-021, provides the means to monitor Unit 1 feedwater temperature downstream of RWCU injection. A thermowell will be installed on vertical risers for both feedwater lines inside the drywell at 35 ft elevation. Conduit will be run from these thermowells/temperature elements through a penetration to a recorder located on 20 ft elevation of the Reactor Building.

The recorder will be multipoint and will constantly monitor feedwater temperature at a chart speed of 2 in/hr. The recorder will also be capable of recording at faster chart speeds, if required. The chart paper will be marked with date and time by an auxiliary operator periodically during his shift rounds. The data will be collected and stored for use by BSEP for the feedwater nozzle fatigue analysis (reference NUREG 0619 and NRC Generic Letter 81-11). A procedural draft revision is included in this plant modification package to detail the data collection.

The temperature will be measured by thermowell mounted, fast time response resistance temperature detectors (RTDs). The RTDs will be capable of measuring temperature from 0°F to 750°F.

This PM has been reviewed in accordance with procedure RP-1.97 with regard to impact upon instrumentation credited for postaccident monitoring capability, and the conclusion is that no postaccident monitoring capabilities are affected.

SAFETY SUMMARY: No equipment important to safety is affected by this plant modification.

The installed monitoring instrumentation penetrates the reactor coolant pressure boundary and is designed for and installed in accordance with appropriate specifications and procedures for this service.

The instrumentation does not provide any system parameter control function. The feedwater temperature indication, which is provided by this plant modification, will be used by BSEP to analyze feedwater nozzle cracking as required by NUREG 0619 as amended by the NRC Generic Letter 81-11.

This plant modification does not pose any new failure mode to the Feedwater System nor alter any safety margin as defined by the Technical Specifications for BSEP Unit 1.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-028, MSR Pocket Shell Drain Line Replacement; Replace MSR Pocket Shell Drain Lines With an Upgraded Material

FUNCTIONAL SUMMARY: This plant modification will replace the MSR pocket shell drain line piping downstream of the flow orifices to the TEE where the east and west MSR lines connect upstream of the condenser. These lines are severely eroded and are being replaced with an upgrade material. The piping configuration will remain unchanged. Supports and support locations will remain unchanged.

SAFETY SUMMARY: The piping this modification is upgrading is located in the Turbine Building and is non-Q, nonsafety-related piping. No safety-related equipment is affected by this pipe replacement, and this line is not referred to in the technical specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-034, ATWS-SLCS Upgrade; Modify SLC as follows: Pump Control Switch Changeout, Relief Valve Setpoint Change; Addition of F033A and B Check Valve Test Conn; Flow Indication Loop on Test Line; Relief Valve Discharge Vent

FUNCTIONAL SUMMARY: Plant Modification 86-084 provides the requirements and guidelines to perform the following tasks on the Unit 1 Standby Liquid Control System.

1. Existing control switch for SLC pumps allows operation of only one pump at a time. This control switch will be replaced by a new control switch which will allow simultaneous operation of both the pumps. Tests performed on the SLC system during the 1986 Unit 2 outage demonstrated that "Two Pump Operation" is a viable option and exceeds the requirements imposed by ATWS Rule 10CFR50.62, paragraph C.4. The ATWS rule requires a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 wt percent sodium pentaborate solution for a 251-inch diameter vessel plant. The equivalent flow rate required for a 218-inch diameter vessel plant like BSEP is 66 gpm.

2. Relief valves 1-C41-F029A and B settings will be increased to 1450 psig to ensure adequate margin for two-pump operation.

3. A flow indicating loop will be installed in the SLCS test tank return line. This is not an ATWS requirement, but was requested by the ISI group to remove Code Exception PR-03 to ASME, Section XI, requirements (reference ENP-17) for the SLCS pumps. This flow indicating loop does not perform any safety-related function.

4. Check valve test connections will be installed upstream of valves F033A and F033B. This is not an ATWS requirement, but was requested by the ISI group in response to NRC Inspection IER 86-11/12. This test connection does not perform any safety-related function.

5. Vent connections will be installed on relief valve F029A and B discharge lines 17-1-155 and 18-1-155, to provide a means of flooding/venting these lines with the relief valves installed. This is added to satisfy FSAR requirements.

SAFETY SUMMARY: These changes do not increase the probability of occurrence or consequences of malfunction of the SLC System, or of any other safety-related equipment evaluated in the UFSAR. The increased boron injection rates created by dual pump operation is intended to increase safety margins of an accident scenario requiring the use of the SLC System, while remaining within the 6-25 ppm per minute injection rate limits, FSAR 9.3.4.3.

The relief valves are capable of operating at a setpoint of 1450 psig, per the manufacturer, without damage or malfunction. The remainder of the SLC System affected by the relief valve setpoint increase was evaluated and determined to be capable of performing at the increased pressure without compromising the integrity or function of the components in the system.

The change in solution concentration in the SLC System involves maintaining the percent by weight concentration at a minimum level of 13%. This level of concentration is within the region of required volume concentration that has been previously evaluated for the SLC System and equipment.

The Boron injection rate increase (dual pump operation) relief valve setpoint increase and the change in solution concentration for the SLC System are required to meet the requirements of the ATWS rule. The ATWS rule requirements provide a more conservative basis for the performance of the SLC System by increasing the injection flow rates and maintaining the solution concentration at or above 13%. The design and implementation of the ATWS requirements are intended to increase safety margins of an accident scenario requiring the use of the SLC System.

The remaining changes to the SLC System are designed to increase the reliability and safety of the system by verifying pump and pump check valve operation and to vent the relief valve discharge line. These changes are required to satisfy ASME and FSAR requirements.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-037, Reactor Feed Pump Minimum Flow Line and Main Steam Valve Modifications; This Modification Repairs the Spargers for the Reactor Feed Pump Minimum Flow Lines and Replaces Existing Supports. It Also Deletes MSR Blanket Steam Valves and Turbine Gland Seal Valves.

FUNCTIONAL SUMMARY: During an inspection of the Unit 2 condenser in 1986, several problems were noticed. It was discovered that the spargers for the reactor feed pump minimum flow lines (4 each) were cracked near the condenser wall and the pipe supports were damaged. Also, miscellaneous supports in the condenser were discovered to be damaged. Since the Unit 1 design is identical to the Unit 2 design, the Unit 1 pipe supports are to be replaced and the pipe will be replaced if determined to be necessary during inspection. Also, any miscellaneous damaged supports will be replaced as required.

This modification will also mechanically and electrically remove the following valves:

1-MS-2BSV1, V70	1-MS-1BSV2, V69
1-MS-1BSV1, V67	1-MS-2BSV2, V68

These valves are the motor-operated valves and check valves installed on the moisture separator reheater (MSR) that were designed to be used for applying a steam blanket on the MSR internals for corrosion protection when the unit is not in service. The auxiliary boiler system was designed to be the source of steam. The steam blanket system has never been used, and the valves have deteriorated to the point where frequent maintenance is required when the unit is in operation. This results in excessive costs and unnecessary exposure, since the valves are in a high radiation area. Therefore, the valves are to be removed, the pipe is to be capped, and the wiring is to be spared.

Valves 1-MVD-S6 and 1-MS-V52 are used in the turbine steam system. These valves have been used in the past; however, are no longer used for steam sealing since this valve lineup creates the possibility for contaminating the new auxiliary boiler. These valves are to be mechanically and electrically removed for the same reasons as stated above.

The electrical components affected by the removal of these valves include the following:

VALVE	MCC/COMPT	REMARKS
MS-1BSV1	1TH D34	MCC Spared
MS-2BSV1	1TH DZ0	"
MS-1BSV2	1TL/MM9	"
MS-2BSV2	1TL/MN2	"
MVD-S6	1TM/MQ1	"

VALVE	MCC/COMPT	REMARKS
MS-V66	1TH/DZ2	Circuit Affected Only
MS-V65	1TL/MM5	" " "
MS-RSCV1	1TH/D33	" " "
MS-RSCV2	1TL/MM8	" " "
MS-EBPV1	1TH/DZ1	" " "
MS-EBPV2	1TL/MN3	" " "
MS-1SRDCVC1	1G-CB HQ9 17	" " "
MS-2SRDCVC1		
MS-1SRDCVC2		
MS-2SRDCVC2		
MS-MSDCVC1		
MS-MSDCVC2		

MISCELLANEOUS

BOP Termination Cabinet XU-23	Work in Cabinet Only
BOP RTG Benchboard XU-2	Work in Cabinet Only

SAFETY SUMMARY: The following responses address the modifications to the reactor feed pump minimum flow lines:

The accident evaluation for the Feedwater System concerns the loss of or increase in feedwater control and the loss of feedwater heaters. Since this modification restores the Feedwater System to its original state, the probability of an evaluated accident occurrence nor the consequences of the accident are increased.

Since the Feedwater System is restored to its original design, a safety system is not adversely affected. Thus, neither the probability of the occurrence of safety equipment malfunction nor the consequences of equipment malfunction is increased. Also, since safety equipment is not affected, different types of safety equipment failure are not created.

Modifications to feedwater are not addressed in the technical specifications, thus the margin of safety as defined in the technical specifications is not affected.

The following responses address the removal of the auxiliary steam valves.

The accident evaluation for the turbine gland seal system concerns radiation releases into the atmosphere following a seal failure. The removal of the auxiliary steam valves does not increase this occurrence nor the consequences of the accident since the pipe is capped in place of the valves.

The removal of the auxiliary steam valves does not affect the operation of the gland seal system or any safety-related system. Thus, neither the probability of the occurrence of safety equipment malfunction nor the consequences of equipment malfunction is increased. Also, since safety equipment is not affected, different types of safety equipment failure are not created.

Modifications to the turbine gland seals are not addressed in any technical specifications, thus the margin of safety is not affected.

The following responses address the removal of the MS   ket steam valves:

The MSRs are not safety-related components, thus modifica   s removal of the blanket steam valves) do not change the FSAR evaluation of   ent consequences.

Failure of the MSRs to perform its function would only affect turbine performance; it would not affect any safety-related component. Therefore, the FSAR evaluation of equipment malfunction probability and equipment malfunction consequences is unchanged. Also, since no safety equipment is affected, different types of safety equipment failure are not created.

Modifications to the MSR are not addressed in any technical specifications, thus the margin of safety is not affected.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-040, Replacement of Extraction Piping From Deaerator to No. 3 FWH; Replace Extraction Piping From Deaerator to No. 3 FW Heaters With Chrome/Moly Piping, Along With Miscellaneous Turbine Building Support Work

FUNCTIONAL SUMMARY: This plant modification will replace the A106, Grade B, carbon steel piping from the deaerator to the No. 3 feedwater heaters with A335 P22 chrome moly piping. The pipe is presently eroded and has throughwall failures. The modification will also repair and modify pipe supports with design deficiencies in the Turbine Building. The piping configuration will remain the same. The eroded nozzles of the deaerator and No. 3 feedwater heaters will be built up to design by weld metal deposition as required.

SAFETY SUMMARY: The changeout of the piping to a new material does not affect the probability or consequences of any accident previously evaluated in the FSAR. Located in the Turbine Building, the piping is not safety related, has no effect on nuclear safety systems, and is not addressed in the technical specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-041, Recirculation Chemical Decon; Cut Instrument Lines Off of RCR Riser Piping to Facilitate Chemical Decon Injection. Remove Instrument Lines Back to Penetration and Cap at Penetration and RCR Risers

FUNCTIONAL SUMMARY: Each 12-inch reactor recirculation piping riser has a 1-inch instrument line attached approximately 3 feet below the center line of the riser elbow. The instrument lines reduce to 3/4 inch and exit the primary containment at various penetrations. Each line is capped at its penetration outside of primary containment. These instrument lines were capped by previous plant modifications and are no longer required for plant operation.

This plant modification cuts the instrument lines at the reactor recirculation riser piping such that the lines can be utilized for injection points for the chemical decontamination of the reactor recirculation piping. After the chemical decontamination, the instrument lines will be capped at the recirculation riser piping. The remaining instrument piping will be removed back to the primary containment penetrations where the penetrations will be capped.

SAFETY SUMMARY: Subject instrument lines serve no safety function. Instrument lines are presently capped outside of containment. Capping these lines inside of primary containment does not change the probability nor the consequences of any accidents evaluated in the FSAR.

This modification eliminates several lengths of stagnant instrument lines. Removal and capping of stagnant instrument lines will not change the probability of accident or equipment failure, or the consequences of equipment failure, and may reduce these probabilities since the unused lengths of pipe are being removed reduces the surface area of the pressure boundary.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-052, 1-B32-F043A and B, and F044A and B Permanent Operator Disconnect; Disconnect Operators, Spare Cables, MCC Relays, and Remove Switches.

FUNCTIONAL SUMMARY: This modification removed all electrical connections from the valve operators for valves 1-B32-F043A and B, Reactor Recirc Pumps 1A and 1B discharge equalizer valves, and 1-B32-F044A and B, reactor recirc discharge equalizer bypass valves. The valve operators were left in place due to the negative cost impact associated with their removal.

This modification was required because:

1. BSEP technical specifications do not permit cross-tying of the recirc loops during power operation or startup (reference Technical Specification 3.4.1.1)

2. All four valves are under permanent clearance for administrative control purposes. Permanent clearances are not an acceptable means for permanent electrical disablement and are an added burden on Operations personnel.

3. Some of the motors have been robbed for use on other motor-operated valves.

4. Operator handwheels are locked in place.

Additionally, all logic, indication lights, control switches, cables, and motor control centers have been removed or spared as appropriate.

Associated with the logic removal, systems other than the recirc system are affected by this modification. They are:

System	Major Component Affected	Cabinet Numbers
RHR Sys A	Relay Logic	H12-P617
RHR Sys B	Relay Logic	H12-P618
Recirc MG Sets	Relay Logic	B32-P002A and B
Jet Pump Instr Sys	Relay Logic	H12-P619

Other cabinets affected by this modification are:

H12-P603, P624, and C91-P608.

SAFETY SUMMARY: Because the valve operators are not included in any of the accident sequences in Chapter 15 and reliability and redundancy of this portion of the recirc system is not affected with the motor operators disabled, no malfunction probability is affected.

Although the valves have had logic to close them due to drywell high pressure and reactor low level (LOCA signal from the LPCI System), both units have operated since 1982 with the B32-F043A, F043B, and F044B locked closed which is their safest position. The F044A is locked open due to thermal considerations. This is acceptable considering that this is a two-inch line and that if a line break were to occur on the downstream side of the open bypass valve (F044B), it would be no different than a break upstream of either bypass valve (F044A or B) which is bounded by the double-ended guillotine line break already evaluated.

No new equipment is added, and operator disconnect does not violate single active failure criteria or change common mode failure probabilities because the previously evaluated locked valve positions are unaffected by the operator disconnect. The valve positions were accepted based on the premise that if a line break were to occur on the downstream side of the open bypass valve (F044B), it would be no different than a break upstream of either bypass valve (F044A or B), which is bounded by the double-ended guillotine line break already evaluated.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-076, Modify Recirculation MG Sets Control Loops; Remove Master Controller Speed Demand Limiter, Error Limiting Network, Speed Controller, Replace M/A Transfer Stations and K23 Timers. Install Manual Reset for K2 Relays.

FUNCTIONAL SUMMARY: This modification performed the following within the reactor recirculation pump/MG sets control system for Unit 1:

1. Removed the master controller (1-B32-R620) and speed demand limiter (1-B32-K615).

2. Removed the error limiting networks (1-B32-K620A and B) and speed controllers (1-B32-R622A and B).

3. Replaced the M/A transfer stations with manual control stations (1-B32-R621A and B).

4. Replaced the feedwater interlock timers (1-B32-K23A and B).

5. Installed manual resets for the speed limiter No. 1 bypass relays (1-B32-K2A and B).

6. Corrected drawing errors on 1-FP-5572, sheets 3 and 5, and 1-FP-50614, sheet 6.

The Unit 2 modification that performed the recirculation control system changes is PM 86-012.

This modification was in response to EWR 84-167.

MODIFICATION CHANGES: This modification simplified each recirculation pump/MG set control system by establishing one current loop for control of generator speed. It was accomplished by removing the master controller, speed demand limiter, error limiting networks, and speed controllers. Speed control of each generator is maintained by its respective manual control station. The current loop includes the manual control station, speed limiter No. 1, speed limiter No. 2, signal failure alarm unit, and scoop tube actuator. The current supplied by the MV/I transmitter feeds the generator speed meter. The manual control stations are modified GEMAC 542-08 controllers and are installed where the M/A transfer stations were located. The GEMAC 542-08 controllers were supplied with 5-turn control potentiometers. These are replaced with 10-turn potentiometers to provide finer control. The replacement of the potentiometers was necessary due to the elimination of rate limiting within the loop. No change occurred in the function of the signal failure unit, speed limiter No. 1 or speed limiter No. 2.

In the startup mode, the signal generator supplies current to the loop at the 45% level. The startup current loop includes the signal generator, speed limiter No. 2, signal failure alarm unit, speed demand meter, and scoop tube actuator. After the field breaker closes, current control reverts to the manual control station.

The majority of recirculation pump runaways and consequent scrams have been attributed to failures within the error limiting networks and speed controllers. The primary purpose of these components is to allow operation in the automatic mode (load following). At BSEP this mode is not utilized. With removal of these components from the control loops, circuit load for the master controller exceeded its design limit, making it ineffectual. Consequently, the speed demand limiter was no longer needed. The removal of these components simplified control by establishing independent manual control of each MG set.

At least one scram has been attributed to the automatic reset of the speed limiter No. 1 bypass relay [K2A(B)]. This modification has rewired the ac auxiliary circuit to require a manual reset, thus providing complete operator control. The addition of the manual reset was accomplished by using a spare contact on the runback reset switch. This contact, in parallel with a normally open contact of the K2A(B) relay, provides a seal in and reset for the K2A(B) relay.

In conjunction with this, a contact of the K2A(B) relay is wired into the recirculation runback light circuit providing operators with an indication of the speed limiter No. 1 bypass relay status.

Prior to a reset, manual control will have to be established below the 28% level due to the elimination of rate limiting within the control loop.

This modification replaced the feedwater interlock timers K23A(B) (time-delay pickup and time-delay drop out) with time-delay drop out timers. The time delay pickup was necessary to prevent the K2A(B) prematurely resetting when feedwater flow exceeded 20%. However, with the installation of the manual reset for K2A(B), the time-delay pickup function of the timers is no longer needed.

Drawing errors on 1-FP-5572, sheets 3 and 5, and 1-FP-50614, sheet 6, were corrected per this modification. No functional changes occurred as a result of drawing corrections.

Implementation of this modification provided reliable pump/MG set control.

SAFETY SUMMARY: The analysis in Section 15.3.2 addresses a recirculation failure (speed controller failure) resulting in decreased recirculation flow. It states that the transient resulting from the single MG set failure (zero speed failure) is similar to but less severe than the trip of a recirculation pump. This plant modification replaced the speed controller with a manual controller. However, failure of the manual controller would result in the same transient generated by a speed controller failure and would still be less severe than a recirculation pump trip.

This PM removed the master controller so analysis of simultaneous pump failure is no longer necessary.

The recirculation flow controller failure analysis in Section 15.4.4 is based on the operating parameters of the MG set fluid coupler. This modification improved reliability of the control system thus reducing the probability of its failure, but had no affect on the operating parameters of the fluid coupler itself.

The modification does not affect Q-list equipment, required preventive maintenance, or inspections. It does not violate the single active failure or separation criteria and common mode failure is not increased.

Present technical specifications requirements and assumptions are not changed.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-094, 1-B21-PI-R612 and 1-B21-PDT-N035 Core D/P Loop Removal; Remove Above Instruments and Associated Wiring Spared.

FUNCTIONAL SUMMARY: This modification permanently removed the following instruments from the plant: 1-B21-PDT-N035 and 1-B21-PI-R612.

These comprised an instrument loop that monitored jet pump developed head. This parameter is essentially duplicated by the core plate D/P on recorder B21-PDR-R613.

The purpose for removal is two fold:

1. This transmitter was believed to have been the cause of a scram in 1986 and has been isolated on both units since then. According to the site GE Operations Engineer, this problem has been encountered at other BWRs.

2. The indicator was rarely, if ever, used. It is not required for any PTs or any other procedures and was recommended for removal by the Human Factors Engineering group. The Operations Engineers and the Operations Manager also concurred with removal of the loop.

The instruments were physically removed. The three-valve manifold at the transmitter was plugged, all permanently closed, and the low pressure rack isolation valve from line B21-704 was also permanently closed. The indicator was removed and a blank plate mounted in its place. The power supply was part of a multi-unit power supply B21-K604. This unit of the K604 supply was spared. Other interconnecting conductors/cables were spared as appropriate.

SAFETY SUMMARY: The instrument loop had no function in any of the accident sequences in Chapter 15 of the FSAR; therefore, its removal had no impact on the previous evaluations.

This equipment did provide indication that was essentially redundant to a core differential pressure indication. Removal of the equipment did not compromise system reliability or reduce redundancy assumed in the FSAR. Single active failure criteria and common mode failure were not affected. These items were unaffected because the transmitter was Q-list for pressure boundary only and the balance of the loop is non-Q.

This parameter was not used in the basis for any technical specification; therefore, no margin was reduced.