



Carolina Power & Light Company

Brunswick Steam Electric Plant
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February 12, 1987

27 MAR 2 All: 18

10CFR50.59

FILE: B09-13510C
SERIAL: BSEP/87-0065

Dr. J. Nelson Grace, Administrator
U.S. Nuclear Regulatory Commission
Suite 2900
101 Marietta Street NW
Atlanta, GA 30323

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

Dear Dr. Grace:

In accordance with 10CFR50.59, the following annual report is submitted for 1986. 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 1986 which are not described in the FSAR.

Very truly yours,

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

MJP/mbh
MSC/87-014

Enclosure

cc: Director of Inspection and Enforcement
NRC Document Control Desk

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TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-85-110, Revision 0, Standby Liquid Control System (SLCS) Two Pump Flow to the Reactor Vessel Flow Test

DESCRIPTION: This test verifies the discharge pressure calculation expected during SLCS two pump flow and expected SLCS relief valve setpoint margin during two SLCS pump flow conditions.

SAFETY SUMMARY: Section 9.3.4.3 of the UFSAR indicates that the SLCS is a special safety system not required for unit operation. However, it is acknowledged that the system must be capable of operating during unit operation as included in the Technical Specifications. Chapter 15 of the UFSAR does not address the failure of the SLCS to operate as an accident. The test parameters will not exceed the design parameters of SLCS discharge piping. Therefore, the probability of a piping failure of the portion of the SLCS within the reactor coolant pressure boundary will not be increased. Section 15.6 of the UFSAR does not apply, the LOCA. The SLCS is a safety system which must be available to aid in accident mitigation when the Reactor is in Conditions 1, 2, and 5 per the Technical Specifications. SP-85-110 will be performed with the reactor in Conditions 3 or 4 when the SLCS is not required to be operable and no credit is taken for its accident mitigating capabilities during the test. After the test the SLCS will be restored to its original operating configuration and shown to be operable through the normal PT performance. This special procedure does not modify the SLCS in any way. It tests the SLCS in a manner which will not exceed any system design parameter. The regular performance of the SLCS PT's will verify operability of the SLCS.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-86-034, Revision 0

DESCRIPTION: This procedure was developed to facilitate troubleshooting on instrument lines and manifold valves to locate possible leaks associated with reactor level instrument transmitters 1-B21-N042A and 1-C32-N004A.

SAFETY SUMMARY: This special procedure does not constitute an unreviewed safety question. It provides instructions for troubleshooting of associated instrument lines and valves.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-86-044, Revision 0, Withdrawal of CRD 22-19 for Drive Mechanism Replacement

DESCRIPTION: This procedure provides for instructions to remove control rod drive 22-19 which was installed improperly.

SAFETY SUMMARY: This procedure does not affect the plant accident analysis or margin of safety. The plant refueling and control rod interlocks are not affected by this procedure.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-86-069, Replacement of Unit 2 Process Computer Change Detection Printed Circuit Board

DESCRIPTION: Use of this procedure is used to evaluate the Unit 2 process computer to determine if problems encountered with the circuit board are resolved.

SAFETY SUMMARY: Changes to the process computer will not increase the probability of a previously evaluated accident. The process computer cannot create the probability of an accident or the possibility for malfunction of equipment important to safety not already evaluated.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-86-074, Service Water Pump Balancing

DESCRIPTION: This test involves making balance shots to the Service Water Pumps of Units 1 and 2 to obtain data for improving the vibration characteristics of the pumps.

SAFETY SUMMARY: This test is to evaluate a way to improve the operation of the pump and did not increase the probability of an accident as evaluated in the FSAR.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-86-081, Revision 1, Revised Functional Testing Procedure and Excess Flow Setpoint Added for Main Steam Line Radiation Monitors on Unit 2 for Hydrogen Water Chemistry Test

DESCRIPTION: The Hydrogen Water Chemistry Mini-Test 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 the special procedure is temporary and will be removed from the plant at the conclusion of the test.

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).

SAFETY SUMMARY: 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 evaluated will not be increased. The consequences of any equipment malfunction of equipment important to safety will not be increased.

TITLE: SP-86-081 (Cont'd)

SAFETY SUMMARY (Cont'd)

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.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-86-082, Material Monitoring and Reactor Water Chemistry Monitoring During Hydrogen Water Chemistry Test

DESCRIPTION: In order to perform continuous material and reactor water chemistry monitoring during the Hydrogen Water Chemistry Mini-Test, as accomplished by sample inboard isolation valve 2B32-F019 and sample outboard isolation valve 2B32-F020 on reactor recirculation water sampling line 2RXS-1 will be left open. The valves will be returned to their normally closed positions upon completion of the test. Therefore, no revision to the FSAR is required.

SAFETY SUMMARY: This procedure does not increase the probability or consequences of any accident previously evaluated. This procedure will install test equipment and establish flow through it via reactor water recirculation sample line 2RXS-1, and tubing installed per PM 86-082. The function of sample line 2RXS-1 inboard and outboard isolation valves 2B32-F019 and -F020, including their ability to close on isolation signals, is not affected by this procedure. Discharge from the test equipment to the condenser will be via nonsafety-related reactor well drain line 2G41-45-10-157.

This procedure does not create the possibility of a new or different kind of accident or malfunction from any accident or malfunction previously evaluated. This procedure does not affect the design of any safety-related system, nor does it affect the performance of any safety functions. Test equipment operation per this procedure will not change the mode of operation of any other plant system. All tubing and equipment installed by this procedure is nonsafety-related.

This procedure does not involve a decrease in any margin of safety. All pressure retaining tubing and components installed by this procedure have been designed and tested in accordance with ANSI B31.1-1980 Edition.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: SP-86-096, Environmental and Radiological Control (E&RC) Activities During Hydrogen Water Chemistry Test on Unit 2

DESCRIPTION: In this test, hydrogen is injected into feedwater to reduce free oxygen, one of the necessary causative agents of intragranular stress corrosion cracking (IGSCC). As a by-product of the test carryover of N-16 increases. Data collected by E&RC during this test is for information only and is not related to the margin of safety as defined in the basis to any Technical Specification.

SAFETY SUMMARY: E&RC activities deal with monitoring plant chemistry and measuring radiation levels which do not increase the probability of an accident previously evaluated and does not affect FSAR safety equipment.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 85-061, Installation of Emergency Response Facility Information System Data Acquisition System on Unit 1

FUNCTIONAL SUMMARY: This plant modification, PM 85-061, covers the ERFIS Data Acquisition System (DAS) hardware installation; the conduit, cable tray, raised floor, multiplexer cabinets, remote electronics cabinets, and distributed DAS modules in the Electron Equipment Room (EER); the fiber-optic and power cables from the EOF/TSC Building to the DAS; and the signal wires/cables to the existing system panels. Input signal cable terminations and internal wiring in existing panels that affect plant systems are not included in PM 85-061.

PM 85-061 includes the total ERFIS DAS functional summary, design basis, safety evaluation, environmental qualification review, and final system test. All cable pull slips are included.

Input signal cable terminations and internal wiring required to complete the tie-in of the ERFIS signals from plant systems in existing panels will be done by separate associated plant modifications, hereafter referred to as mini mods. These separate mini mods will be developed by plant system and will include specific implementation procedures, clearances required, acceptance tests, and references for the system points being tied in. These mini mods will also include all applicable drawings, drawing revision sheets, bills of materials, and document revision sheets for affected maintenance PTs, MSTs, and MIs necessary to complete the tie-in. These mini mods will reference the generic sections of PM 85-061.

Upon completion of the final tie-in terminations, these mini mods will direct the documentation of the termination on the cable pull slips contained in PM 85-061.

Partial turnovers will be made after the input/output (I/O) point terminations are completed and checked to enable the Computer Support group to maintain the hardware after checkout. However, the ERFIS System cannot be considered functional or usable until all I/O points have been installed and checked out and the full system acceptance test has been completed as described in Section E of this plant modification.

The design of the process computer data link, the meteorological tower data link, and the ERFIS trouble annunciator in the Control Room is being finalized and will be added to the modification as the design is completed.

The basic function of this plant modification is to provide a computer-based information monitoring system to meet the requirements of NUREG-0737, Clarification of TMI Action Plan Requirements, and Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (generic letter No. 82-33).

TITLE: PM 85-061 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

The Emergency Response Facility Information System (ERFIS) is an integrated, computer-based plant data collection, analysis, and display system. It is designed to improve the decision-making capability and response time of operating personnel during emergency situations. ERFIS is designed to display relevant plant information to operating personnel in the Main Control Room and to support staff in the Emergency Operating Facility/Technical Support Center (EOF/TSC). The Safety Parameter Display System (SPDS) function of the ERFIS is designed to provide Main Control Room operators with information formatted to support the BSEP Emergency Operating Procedures (EOP) and is designed to address certain Control Room human factors concerns. The EOF/TSC portion of the ERFIS functions is designed to relieve Control Room operators of duties and communications not directly related to reactor system manipulations. The ERFIS is designed to also display essential Regulatory Guide (RG) 1.97 variables that will be used for evaluating incident sequence, determining mitigating actions, and evaluating damages. In addition, the EOF/TSC portion of ERFIS is designed to provide radiological and meteorological data for determining post-accident measures.

A hybrid (i.e., distributed and centralized) Data Acquisition System (DAS) was configured.

In addition, a Nuclear Measurement, Analysis, and Control (NUMAC) System will be installed to monitor scram configuration and control rod position. The NUMAC System is not part of ERFIS but is being added under this modification and will provide outputs to ERFIS. NUMAC is a microcomputer that is constructed of standard plug-in modules in a chassis designed for rack mounting.

The NUMAC output to ERFIS will consist of the following items:

1. An "all rods in" or "all rods not in" signal will be sent to ERFIS.
2. A "scram confirmed" or "scram not confirmed" signal will be sent to ERFIS. This will be based on the plant rules for confirmation of scram.
3. Rod identification and position for all rods will be periodically sent (about every second) to ERFIS. This will occur during normal operation and after a scram.
4. After a scram, a list of rods and their positions that do not conform to the plant shutdown criteria will be prepared by NUMAC computer and sent to ERFIS.

The balance of the input signals are wired to five centralized multiplexer (MUX) cabinets, located in the Unit 1 EER and in the Unit 1 Computer Room.

TITLE: PM 85-061 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

The ERFIS DAS, as presently configured, will obtain input signals from the following BSEP Systems:

<u>System</u>	<u>Designation</u>	<u>Mini Mod NO.</u>
Nuclear Steam Supply Shutoff/ Nuclear Boiler	A71/B21	W
Reactor Recirculation	B32	K
Feedwater/Condensate	CO/C32	S
Control Rod Drive	C11	X
Standby Liquid Control	C41	L
Neutron Monitor	C51	U
Reactor Protection	C71	M
Process Radiation Monitor	D12	B
Area Radiation Monitor	D22	C
Ventilation/Drywell Steam Detection	VA	D
Reactor Building Sampling	RXS	N
Electrical Distribution	ED	T
Electro Hydraulic Control/ Main Steam	EHC/MS	E
Residual Heat Removal	E11	R
Core Spray	E21	Q
High-Pressure Core Injection	E41	G
Reactor Core Isolation Cooling	E51	J
Transverse Incore Probe	TIP	P
Reactor Water Cleanup	G31	F
Service Water	SW	O
Off-Gas/Radwaste	OG/G16	A
Meteorological Tower	MET.TWR.	H
Containment Atmospheric Control	CAC	V
Suppressing Pool Temperature Monitoring System	SPTMS	I

NOTE: "Mini mods" is a term used to indicate brief plant modifications that will be developed to cover the termination of the ERFIS wiring at the tap panels, where the ERFIS ties in the various existing plant systems to gather input information.

TITLE: PM 85-061 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

In summary, ERFIS equipment is divided into three component groups that will perform the following functions:

- Date Acquisition System (DAS)
 - isolation of Class 1E inputs
 - analog and digital input signal processing
 - remote communication with computers
 - error detection
 - diagnostics
 - analog and digital outputs
- Computer System
 - process, log, and alarm (PLA)(i.e., point processing, alarm logging, etc.)
 - communications interfaces
 - error detection/correction
 - diagnostics
 - transient recording and analysis (TRA)
 - real-time analysis and display system (RTAD)
- Control Consoles and Display System
 - color graphic displays
 - hard-copy reports, both graphic and alphanumeric
 - man-machine interface
 - on-line program modification/development
 - on-line/off-line diagnostics
 - graphics modification or addition
 - report modifications
 - security, password changes

SAFETY SUMMARY: ERFIS does not increase the probability of occurrence of any accident previously evaluated in the Final Safety Analysis Report (FSAR). ERFIS is a passive plant information gathering and display system that has no control function. This plant modification is installed to meet the requirements of 10CFR50.47(b); NUREG-0737, Supplement 1 (generic letter 82-33); and Standard Review Plan 18.2, Safety Parameter Display System, Revision 0, 84/12.

ERFIS is a new system composed of two dedicated computer systems, one for each Brunswick Station unit, Units 1 and 2. In certain areas, data is shared between these computer systems for the two units; however, no potential safety concern is created for either unit because of the optical interface existing between safety-related circuits of both units and ERFIS.

TITLE: PM 85-061 (Cont'd)

SAFETY SUMMARY (Cont'd)

ERFIS is a plant information system with no control function, and it does not increase the consequences of any previously evaluated accident. In the event of an accident, it will provide Main Control Room personnel with timely information on plant conditions and thus give them the informational capability to make appropriate decisions and actions on a timely basis.

The existence of approved isolation devices connecting input modules to safety-related circuits precludes an increase in the occurrence of malfunctions. Additionally, the ERFIS display of data trends overtime can alert plant operators to potential degraded equipment conditions or incipient failure modes, thus permitting timely correction of the situation. Therefore, the probability of occurrence of malfunction may decrease due to better and timely information of plant conditions.

The consequences of malfunction of equipment important to safety may be decreased due to the availability of earlier, better formatted information on ERFIS. ERFIS, due to its isolation from this equipment, can not detrimentally affect the consequences.

This modification does not create a different type of probability for an accident (or a possibility for malfunction of equipment important to safety) than already evaluated in FSAR because:

- a. ERFIS is a passive information gathering, retrieval, and display system with no control functions. Isolation of safety-related signals from ERFIS is provided by qualified safety-related isolators located in the multiplexer (MUX) cabinets and in distributed DAS modules, and through fiber-optic cables.
- b. The inputs to ERFIS of plant parameters and conditions that require information from safety-related circuits are provided through qualified components (e.g., dropping resistors, cables extending the circuits to the digital or analog input modules, and the input modules themselves). Optical isolation is also provided to isolate each safety-related circuit from the ERFIS system. Those qualified components that are inserted in safety-related circuits and that require power will be powered from Class 1E sources. Those circuits powered from Class 1E sources are electrically isolated from the ERFIS system, since the only ties between the safety-related circuits and the ERFIS system circuits are by fiber-optic cables in the plant data input links.

TITLE: PM 85-061 (Cont'd)

SAFETY SUMMARY (Cont'd)

- c. ERFIS will provide plant operators with timely information on a large number of diverse parameters from plant process circuits that are isolated from redundant, dedicated computers. There is, however, a new, potential, human-interaction failure mode, if plant operators become too dependent on the SPDS and then become deprived of information by a failure of both computers. This, however, is not a concern as operators are required to be able to operate the plant with or without ERFIS. Also, the probability of this occurrence is expected to be low because of the redundant computers with high system reliability, periodic operator training, and prompt corrective action.

The margin of safety as defined in the basis to any Technical Specification is not reduced by this modification because ERFIS is a passive, computer-based information system with no control function. The ERFIS input is isolated from safety-related circuits by qualified isolators, and optical isolation (fiber-optic cable) is also provided to isolate each safety circuit from the ERFIS system. Those qualified components that are inserted in safety-related circuits and that require power will be powered from Class 1E sources. Those circuits powered from Class 1E sources are electrically isolated from the ERFIS, since the only ties between the safety-related circuits and ERFIS circuits are by fiber-optic cables.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-028, Upgraded Replacement of Unit 1 Moisture Separator Reheater Pocket Shell Drain Line

FUNCTIONAL SUMMARY: This plant modification will replace the moisture separator reheater (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 main condenser. These lines are severely eroded and are being replaced with an upgraded material. The piping configuration will remain unchanged. Supports and support locations will remain unchanged.

SAFETY SUMMARY: This pipe is not safety-related. The original stream analysis is adequate and the system remains unchanged. The system function remains unchanged. The piping is non-Q and located in the Turbine Building. Equipment important to safety is not affected. No safety-related equipment is affected by this pipe installation. The pipe is not referenced in Technical Specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-034, Standby Liquid Control System Upgrade on Unit 1

FUNCTIONAL SUMMARY: PM 86-034 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 1,450 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 valves 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.

TITLE: PM 86-034 (Cont'd)

SAFETY SUMMARY (Cont'd)

The relief valves are capable of operating at a setpoint of 1,450 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 84-106, Installation of a Quick Responding Fire Suppression Sprinkler Head in the Unit 2 Emergency Core Cooling System Room

FUNCTIONAL SUMMARY: This modification installs a single quick-responding sprinkler head to provide coverage for both the RCIC steam isolation valve (2-E11-F008) and the RHR common suction valve (2-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.

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 performed 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 steam 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:

TITLE: PM 84-106 (Cont'd)

SAFETY SUMMARY (Cont'd)

1. 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.
2. 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 affects 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 affects 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 83-113, Addition of Redundant Reactor Wide Range Flood-up Level Transmitter and Indicator on Unit 1

FUNCTIONAL SUMMARY: In "position paper on RG 1.97" and "Brunswick Response to NUREG-0737, Supplement 1 - RG 1.97, Application to Emergency Response Facilities," both dated August 9, 1983, 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: To add the redundant indicator to Section 7.5.1-1, B21-LI-R605B, revise 6.2.4-1 to add excess flow check valve B21-IV-2149. The equipment being installed and the lines being extended are not required to mitigate the occurrence of an accident. This modification adds supplement indication only and adds no control functions. Except for the extension of instrument lines, there is no change to existing equipment. New equipment is procured, installed, and tested as Q-List 1E equipment. The new instrument loop being added is for post accident monitoring but is not required to perform any control function which would mitigate the consequences of a malfunction of equipment. This modification adds no control functions. This modification adds supplement indication for an existing instrument loop. The new instrument loop being added 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 86-041, Cutting and Removal of Instrument Line on Reactor Recirculation System Loop to Facilitate Chemical Decontamination on Unit 1

FUNCTIONAL SUMMARY: Each 12-inch reactor recirculation piping riser has a one-inch instrument line attached approximately three 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: Figure 5.4.1-2 requires a change to show the instrument lines capped inside of primary containment. Instrument lines are presently capped outside of containment. Capping these lines inside containment does not change the probability of any accidents evaluated in the FSAR. Subject instrument lines serve no safety function. Capping these lines inside primary containment does not change the operation of any safety system, therefore, the consequences of any accident evaluated in the FSAR is unchanged. This modification eliminates several unused lengths of stagnant instrument lines and is not related to safety equipment. By eliminating stagnant instrument lines, the probability of equipment malfunction is unchanged. Removal and capping of these unused stagnant instrument lines does not affect the consequences of equipment failure. Removal and capping of stagnant instrument lines will not change the probability of accident or equipment failure and may reduce these probabilities since the unused lengths of pipe are being removed thus reducing the surface area of the pressure boundary. Stagnant instrument lines are not used and do not affect system operation.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-143, Installation of Off-Flow Indication for Loops A and B Discharge of the Service Water System Vital Header on Unit 1

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. These new loops are required to meet RG 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 monitoring instrumentation is being added to the service water instrument application description (paragraph 9.2.1.5). This PM adds additional post accident monitoring instrumentation and does not affect any previously evaluated accident. This PM adds additional monitoring instrumentation only. There are no control functions. New flow indication is for monitoring system operation. This PM adds postaccident monitoring instrumentation only. The new equipment is procured and installed as Q-List Class 1E. This PM does not affect equipment previously evaluated in the FSAR. This PM adds supplemental indication for monitoring system operation. The new equipment has no control functions. System operation and function is unchanged. Equipment is procured, installed, and tested as Q-List Class 1E. This equipment is not Technical Specification related. There are no control or setpoint changes.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-145, Appendix R, Communications Upgrade on Unit 1

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: Description added to FSAR, Section 9.5.2, of a dedicated redundant sound powered phone system for remote shutdown, installed to satisfy the requirements of 10CFR50, Appendix R. Addition of a dedicated phone system for remote shutdown in the event of fire improves plant reliability thus reducing the probability of a previously evaluated accident. Addition of a more reliable communication system to ensure remote shutdown capability decreases the consequences of any previously evaluated accident. Installation of a sound powered phone system in no way affects any safety-related equipment to the point of changing the probability of malfunction. Addition of an independent sound powered phone system does not alter the design basis of any safety-related equipment, therefore, does not change the consequences of a malfunction. The design basis of any affected equipment important to safety is not altered, therefore, the change neither creates or increases the probability of any previously unevaluated malfunction. Improved reliability to provide remote shutdown increases the margin of safety as defined in the Technical Specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-094, Removal of Unit 1 Reactor Core Differential Pressure Instruments 1-B21-PI-R612 and 1-B21-PDT-N035

FUNCTIONAL SUMMARY: This modification permanently removes the following instruments from the plant:

1-B21-PDT-N035
1-B21-PI-R612

These instruments currently comprise an instrument loop that monitors "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 is believed to have been the cause of a scram in 1986, and has been isolated on both units since then. According to the Site General Electrical Operations Engineer, this problem has been encountered at other Boiling Water Reactors.
2. The indicator is rarely, if ever used. It is not required for any PTs or any other procedures, and was recommended for removal by the Human Factor Engineering group. The Operations Engineers and the Operations Manager also concurred with removal of the loop.

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

SAFETY SUMMARY: This instrument loop has no function in any of the accident sequences in Chapter 15 of the FSAR, therefore, its removal can have no impact on the previous evaluations.

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

This parameter is not used in the basis for any Technical Specification, therefore, no margin is reduced.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-021, Installation of Fast-Time Response Temperature Element Into Each Feedwater Line on Unit 1

FUNCTIONAL SUMMARY: This plant modification provides the means to monitor Unit 1 feedwater temperature downstream of Reactor Water Cleanup System (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 multi-point and will constantly monitor feedwater temperature at a chart speed of two 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 RG 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: Monitoring instrumentation installed by this PM measures feedwater temperature only and serves no system parameter control function. Installation of thermowells penetrates reactor coolant pressure boundary. Design and installation of these items is in accordance with appropriate specifications and procedures for this service. Design and installation of pressure retaining components is suitable for end use service conditions. No equipment which is important to safety is affected by this plant modification. No equipment which is important to safety is affected by this PM. This modification does not introduce any equipment or design which pose any new failure mode to the system. This modification does not alter any safety margin as defined by the Technical Specifications for BSEP Unit 1.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-011, Upgrade of Unit 1 Uninterruptible Power Supply (UPS) System

FUNCTIONAL SUMMARY: The Unit 1 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 the 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 new 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 pickup which will be set at a value of approximately 1,520 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.

SAFETY-SUMMARY: No new anticipated operational 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

TITLE: PM 86-011 (Cont'd)

SAFETY SUMMARY (Cont'd)

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 has 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 system 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 82-221F, Addition of Anti-Cavitation Flow Control Valve to the Reactor Building Closed Cooling Water System Heat Exchanger Service Water Outlet on Unit 1

FUNCTIONAL SUMMARY: This plant modification installs anti-cavitation flow control valve 1-SW-V382 in the common service water (SW) discharge line from the RBCCW heat exchangers. Addition of this valve will eliminate the need to throttle flow with the heat exchanger 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 determined and spared.

SAFETY SUMMARY: This modification does not alter safety margins as defined by the Technical Specifications for Unit 1. It does not introduce equipment or designs which may pose any new failure mode into the system.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-052, Permanent Disconnection of Valve Operators Associated With Unit 1 Reactor Recirculation System Pumps 1A and 1B Discharge Equalizer Valves, 1-B32-F043A and B, and the Pumps Discharge Equalizer Bypass Valves 1-B32-F044A and B

FUNCTIONAL SUMMARY: This modification will remove all electrical connections from the valve operators for valves 1-B32-F043A and B, reactor recirculation pumps 1A and 1B discharge equalizer valves, and 1-B32-F044A and B, reactor recirculation discharge equalizer bypass valves. The valve operators are to be left in place due to the negative cost impact associated with their removal.

This modification is required because:

1. BSEP Technical Specifications do not permit the cross-tying of the recirculation loops during power operation or startup (reference TS 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 hand wheels are locked in place.

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

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 Recirculation 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 are bounded by the double ended guillotine line break already evaluated.

TITLE: PM 86-052 (Cont'd)

SAFETY SUMMARY (Cont'd)

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 are bounded by the double ended guillotine line break already evaluated.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-005, Addition of Automatic Containment Atmosphere Control (CAC) Purge and Vent Valve Isolation on Off-Gas Stack High Radiation on Unit 1

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 PM 86-006 will be operable or partially operable before the Unit 1 PM 86-005 is completed. Unit 2 PM 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 normally open 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 RB exhaust radiation monitor, K609A, auxiliary trip unit, C51A-Z2A, relay K83A, and 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 is connected to the B loop radiation monitor the same as Contacts 1 and 2 are connected to the A loop.

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 loop; Unit 1 - 3-55, K83A and B, K82 in A and B loop. 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 loop; Unit 1 - 3-55, K82 in A and B loop. This will give both units an Off-Gas 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 loop. 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 loop.

TITLE: PM 86-005 (Cont'd)

SAFETY SUMMARY: This modification adds a primary containment Group 6 isolation trip from main stack high radiation. Presently installed safety-related equipment function will remain unchanged. Radioactive release to environment from open purge/vent valves will be reduced due to automatic isolation instead of waiting for operator manual isolation. Devices in this modification which interface with safety-related equipment will be procured, installed, and tested as Q-List equipment. Separation criteria of existing isolation logic circuits is maintained. This is an addition trip input to a safety-related system during normal equipment operation or in case of equipment malfunction. This additional trip input does not affect the probability of an accident or create a possibility for safety-related equipment malfunction. The failure of the new circuit would not inhibit normal operation of the safety-related system. Change to "Process and Effluent Radiological Monitoring System" and "Primary Containment Isolation and Nuclear Steam Supply Shutoff System" by adding primary containment isolation from main stack high radiation. Section 11.5.2.3; Tables 11.5.2-1; 7.3.1.1.6; 7.3.1.1.7, Table 7.3.1-3; Table 7.3.1-6; Figure 11.5.2-3. Adds control function to main stack radiation monitor, initiating containment isolation on main stack high radiation. Section 11.5.2.3; Tables 11.5.2-1; 7.3.1.1.6; 7.3.1.1.7; Table 7.3.1-3; Table 7.3.1-6; 6.2.4. The new circuit would not inhibit operation of any safety-related system, therefore, it would not decrease the margin of safety.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-006, Addition of the Automatic Containment Atmosphere Control (CAC) Purge and Vent Valve Isolation on Off-Gas Main Stack High Radiation on Unit 2

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 PM 86-006 will be operable or partially operable before the Unit 1 PM 86-005 is completed. Unit 2 PM 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 normally open 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 2 relays (3-55 and 3-56) will be mounted in Unit 2 Control Room cabinet XU-53 and will be powered from a separately fused circuit off of the "CAC isolation trip override" circuit. Unit 2 trip relay 3-56 is controlled by Unit 2 relay 3-55. Unit 2 Relay 3-56 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 RB vent radiation monitor K609A and auxiliary trip unit C51A-Z2A, relay K83A, 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 3-56 is connected to the B loop radiation monitor the same as contacts 1 and 2 are connected to the A loop.

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 loop; Unit 1 - 3-55, K83A and B, K82 in A and B loop. 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 loop; Unit 1 - 3-55, K82 in A and B loop. This will give both units an Off-Gas vent pipe HI-HI alarm and a Group 6 primary containment 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 loop. 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 loop.

TITLE: PM 86-006 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

Change to "Process and Effluent Radiological Monitoring System" and "Primary Containment Isolation and Nuclear Steam Supply Shutoff System" by adding primary containment isolation from main stack high radiation. Section 11.5.2.3; Tables 11.5.2-1; 7.3.1.1.6; 7.3.1.1.7, Table 7.3.1-3; Table 7.3.1-6; Figure 11.5.2-3.

Adds control function to main stack radiation monitor, initiating containment isolation on main stack high radiation. Section 11.5.2.3; Tables 11.5.2-1; 7.3.1.1.6; 7.3.1.1.7; Table 7.3.1-3; Table 7.3.1-6; 6.2.4.

SAFETY SUMMARY: This modification adds a primary containment Group 6 isolation trip from main stack high radiation. Presently installed safety-related equipment function will remain unchanged. Radioactive release to environment from open purge/vent valves will be reduced due to automatic isolation instead of waiting for operator manual isolation. Devices in this modification which interface with safety-related equipment will be procured, installed, and tested as Q-List equipment. Separation criteria of existing isolation logic circuits is maintained. This is an addition trip input to a safety-related system. It does not change the operation or function of the safety-related system during normal equipment operation or in case of equipment malfunction. This additional trip input does not affect the probability of an accident or create a possibility for safety-related equipment malfunction. The failure of the new circuit would not inhibit normal operation of the safety-related system. The new circuit would not inhibit operation of any safety-related system, therefore, it would not decrease the margin of safety.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-131, Installation of Narrow Range Pressure Transmitter and Indicator for the Unit 1 Drywell

FUNCTIONAL SUMMARY: This modification adds a new drywell pressure instrumentation loop to measure the range -5 to +5 psig.

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

This new instrument loop is required to meet RG 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.

This modification adds instrumentation to supplement existing instrumentation described in the FSAR. This change involves FSAR Table 7.5.1-1, safety-related display instrumentation.

SAFETY SUMMARY: The new instrument loop has no control functions. Since this modification adds new instrumentation qualified as 1E and since this instrumentation does not affect any installed equipment, it is concluded that no safety hazard is introduced by this plant modification. The new instrument loop is not Technical specification related nor is it electrically connected to any Technical Specification related equipment.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-112, Rod Worth Minimizer Low Power Setpoint Change on Unit 2

FUNCTIONAL Summary: The rod worth minimizer's low power setpoint (LPSP) is being changed to increase conservatism toward the requirements of Technical Specification 3/4.1.4.1. The low power alarm point (LPAP) remains at the setting previously implemented by PM 75-402. Both of the setpoints are decreasing as the alarm units in reactor feedwater flow analyzer instrument 2-C32-FA-K608 are connected for low operation. References to a feedwater flow input to the RWM are being deleted from the FSAR and several other documents to reflect the plant's actual configuration and design drawings.

SAFETY SUMMARY: This setpoint change will insure that the rod worth minimizer is operable as required at less than 20% of rated thermal power (Section 15.4.1). No equipment is affected which could increase the severity of an event within any of the analyzed accident sequences. This change does not reduce maintenance or surveillance requirements nor reduce the rating, design margin, or redundancy of safety-related components. This change does not affect any redundant configurations nor violate any active failure criteria. This setpoint change involves only the RWM. No other safety-related equipment is affected, therefore, common mode failure probability is not increased and single active failure criteria is not violated. The setpoint change increases the conservatism of the RWM LPSP setting.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-040, Replacement of Extraction Steam Piping From the Unit 1 Feedwater Deaeration to the No. 3 Feedwater Heaters

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 through wall 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: This modification does not affect previous evaluations relative to the occurrence or consequence of accidents evaluated in the FSAR. The involved piping is not safety-related and has no affect on Nuclear Safety Systems.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-130, Installation of a Line of Closely Spaced Fire Suppression Sprinkler Heads Adjacent to Draft Stops in the Unit 2 Reactor Building 50-ft Elevation

FUNCTIONAL SUMMARY: 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 50-ft elevation, separation zones of twenty 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 bound 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 six 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.

SAFETY SUMMARY: This change enhances the protection of redundant safety features by adding protective water curtains in the separation zone to prevent propagation of combustion products across the zone. FSAR Tables 9.5.1.1 and 9.5.1.4. Margin of safety as defined in the basis to Technical Specifications will 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 84-105, Installation of a Quick Responding Fire Suppression Sprinkler Head in the Unit 1 Emergency Core Cooling System Room

FUNCTIONAL SUMMARY: This modification installs a single quick-responding sprinkler head to provide coverage for both the reactor core isolation cooling (RCIC) steam isolation valve (1-E51-F008) and the residual heat removal (RHR) common suction valve (1-E11-F008). The combination of the passive cable protection 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.

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 performed 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 steam leak detection instruments in the area will maintain their function as described in the FSAR until a substantial

TITLE: PM 84-105 (Cont'd)

SAFETY SUMMARY (Cont'd)

(> 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:

1. 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.
2. 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 affects 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 affects 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. FSAR changes are required to include this installation/system and revise the descriptive/location information in Section 9.5. This modification does not affect any Radwaste Systems.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-071, Installation of a Water Fountain in the Radiological Waste Control Room Supplied by the Potable Water System From the Unit 2 Battery Room Safety Shower

FUNCTIONAL SUMMARY: This modification details the installation of a water fountain in the Radwaste Control Room. The water fountain will be supplied with potable water from the battery room 2A safety shower in the Unit 2 cable spread room, tying in above the ceiling of the battery room. The water will pass through an existing penetration in the Control Building fire barrier wall above the Radwaste Control Room ceiling, and down to the center of the Control Room where it will connect to a freestanding water cooler near an existing floor drain. The water fountain will be supplied with electricity utilizing an empty circuit from an electrical distribution panel in the Radwaste Building.

All installation will be performed by plant Instrumentation and Control Mechanical Maintenance personnel.

SAFETY SUMMARY: Installation of the water fountain will not affect the probability of the occurrence or consequences of any accident previously identified in the FSAR.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-037, Repairs to the Flow Spargers of the Unit 1 Reactor Feed Pumps Minimum Flow Lines and Replacement of Existing Pipe Supports and the Deletion of Unit 1 Moisture Separator Reheater 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 man-rem 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.

SAFETY SUMMARY: The moisture separator reheaters (MSRs) are not safety-related components, thus modifications (removal of the blanket steam valves) do not change the FSAR evaluation of accident 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 MSRs are not addressed in any Technical Specifications, thus, the margin of safety is not affected.

TITLE: PM 86-037 (Cont'd)

SAFETY SUMMARY (Cont'd)

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 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 do 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.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-076, Removal of Master Controller Speed Demand Limiter, Error Limiting Network and Speed Controller of Unit 1 Reactor Recirculation Motor Generator Set Control Loops

FUNCTIONAL SUMMARY: This modification will perform the following within the Reactor Recirculation Pump/MG Sets Control System for Unit 1:

1. Removes the master controller (1-B32-R620) and speed demand limiter (1-B32-K615).
2. Removes the error limiting networks (1-B32-K620A and 1-B32-K620B) and speed controller (1-B32-R622A and 1-B32-R622B).
3. Replaces the M/A transfer stations with manual control stations (1-B32-R621A and 1-B32-R621B).
4. Replaces the feedwater interlock timers (1-B32-K23A and 1-B32-K32B).
5. Installs manual resets for the speed limiter No. 1 bypass relays (1-B32-K2A and 1-B32-K2B).
6. Corrects 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 was PM 86-012.

SAFETY SUMMARY: Section 4.4.3.2, delete references to master controller operation.

- Figure 4.4.3-1, delete "automatic flow control range".
- Figure 4.4.1-1, delete "automatic flow control region".
- Section 4.4.3.4, delete entire section.
- Figure 5.4.1-3, revise note 13, delete note 14.
- Figure 5.4.1-4, delete master controller, revise for manual reset of Limiter No. 1.
- Section 7.7.1.2, delete reference to automatic recirculation flow, master controller, speed controller, function generator, and load following.
- Section 7.7.1.4.2.1, delete reference to automatic load following.
- Figure 7.7.1-1, delete load/speed error signal and master controller.
- Section 7.7.2.2, change speed controller to manual controller.

TITLE: PM 86-076 (Cont'd)

SAFETY SUMMARY (Cont'd)

- Section 4.4A, "Nuclear System stability analysis" General Electric (GE) is reviewing the stability analysis to determine if this PM will affect the analysis. Preliminary indications are that this PM will have no affect on the analysis. GE shall confirm this in writing prior to installation.
- Section 15.3.2, delete reference to master controller and speed demand limiter.
- Section 15.4.4, delete reference to master controller and automatic load following.

The analysis in Section 15.3.2 addresses a recirculation failure (speed controller failure) resulting in decreased recirculation flow. The Section 15.3.2 analysis states that the transient resulting from the single MG set failure (zero speed failure) is similar but less sever than the trip of one recirculation pump. With removal of the speed controller (this PM), the failure would result from the 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 removes 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 will improve reliability of the Control System reducing probability of failure. It will have no affect on the operating parameters of the fluid coupler. This modification affects no Q-List equipment. It will have no affect on required preventative maintenance or inspections. It will not result in violation of the single active failure or separation criteria. Common mode failure is not increased. This modification does not necessitate a change in Technical Specifications. It does not affect the requirements or change the assumptions of any Technical Specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 83-003, Installation of High Density Fuel Storage System (HDFSS) Modules in Unit 1 Spent Fuel Pool

FUNCTIONAL SUMMARY: 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 HDFSS modules. The first three HDFSS modules were installed under PM 82-205. The remaining modules to be installed consist of rectangular fuel storage cells combined in a 13 x 15 and 13 x 19 matrix assembly. In order to install the modules, it is necessary to remove the existing control rod storage hangers and the two existing shipping-can support tubes. New shipping-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. The 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. 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. 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 (Section 5.6, Fuel Storage). The two new shipping-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. The shipping-can support tubes are fabricated from 10-inch (Sch 40) 304 stainless steel pipe. Racks for trash cask and transuranic (in-core detector) liners will be installed north of the spent fuel shipping cask storage area. 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. The shipping-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. The effectiveness of the Boral poison in the modules was tested by source/detection/instruments prior to loading fuel assemblies into the modules

TITLE: PM 83-003 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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

1. The level verification tool
2. The wet lifting fixture
3. The support base lifting fixture
4. The uprighting fixture
5. The tolerance verification fixture (dummy fuel assembly)
6. The wet lifting fixture attach/release tool
7. The module wall positioner
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. 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. There will be no spare parts for the HDFSS modules, however, there will be stock level additions for the parts of the control rod hanger assemblies. 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. Periodic testing of the Boral poison will be in accordance with PT-90.11. This modification will cause changes to the updated BSEP FSAR (in addition to the change implemented as part of PM 82-204, HDFSS Installation - Phase I) due to the following: Addition of two HDFSS modules to the Unit 1 spent fuel storage pool (SFP), removal of 34 control rod storage brackets from the Unit 1 SFP, and removal of the aluminum framework from the spent fuel shipping cask storage area in the Unit 1 SFP. The addition of two HDFSS modules to the Unit 1 SFP will change the description of the spent fuel storage arrangement in Sections 9.1.2.2.2 and 9.1.2.2.2 of the FSAR. In addition, a new Figure 9.1.2-1a must be added to show a spent fuel storage pool configuration with five HDFSS modules for Unit 1. FP-9820 will become the new Figure 9.1.2-1a after the changes shown on SK-M-83-003-001 have been incorporated. The removal of 34 control rod storage brackets from the Unit 1 SFP will reduce the number of storage positions from 96 to 22. An additional 130 control rods can be accommodated from temporary control rod hangers supported from brackets mounted to the curb of the SFP. F-18004 will become a new Figure 9.1.2-5a after the changes shown on SK-M-83-003-009 have been incorporated. The description of control rod storage in Section 9.1.2.2.1 will require changing to reflect the new control rod storage arrangement.

TITLE: PM 83-003 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

PMs 77-75 and 77-56 were written to add restraining pads to the aluminum framework of the spent fuel shipping cask storage area at elevation 93.3 ft. The restraining pads were thought necessary to prevent toppling of the spent fuel cask in the unlikely event of an earthquake. However, an analysis by United Engineers & Construction (UE&C) proved that the spent fuel shipping cask would not tip over during an earthquake with the redundant yoke attached and PMs 77-55 and 77-56 were never installed. Therefore, the description of the spent fuel shipping cask storage area given in Section 9.1.4.2.4 requires correction. The remaining aluminum framework is unnecessary and will be removed from Unit 1 by this modification to provide space for a new sipping-can station and the curb-mounted control rod hangers. The safety evaluation section of the FSAR for the spent fuel storage racks (Section 9.1.2.3) was changed by PM 83-204 and assumed a "worse case" pool configuration of five HDFSS modules. Therefore, Section 9.1.2.3 requires no technical changes in the analyses. However, there is a description that needs correction. Section 9.1.2.3.1 refers to an 18-inch diameter sump which does not exist, and the FSAR will be change to delete this reference.

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 systems 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 fuel 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 I, 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 spent 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

TITLE: PM 83-003 (Cont'd)

SAFETY SUMMARY (Cont'd)

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 increase 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 on Unit 1 (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;

TITLE: PM 83-003 (Cont'd)

SAFETY SUMMARY (Cont'd)

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:

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

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

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-007, Installation of Alternate 480 Vac Feed for Unit 1 Battery Charger 1B-2 From Motor Control Center (MCC) 1XB and the 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. The FSAR change will indicate the alternate source and associated transfer switches to battery chargers 1B-1, 1B-2, 2B-1, and 2B-2 to meet the requirements of 10CFR50, Appendix R, FSAR, Section 8.3.2.1.2.

SAFETY SUMMARY: Addition of an alternate 480 Vac source and associated transfer switch to the battery charger for 10CFR50, Appendix R, requirements will not increase the probability of an accident previously evaluated. Addition of an alternate circuit and transfer switch does not affect equipment evaluated in Chapter 15, thus consequences of an accident will not be increased. 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 previously evaluated. Enhancement of the battery charger ac supply circuit in no way adversely effect safety-related equipment, therefore, will not increase the consequence of an evaluated malfunction. Enhancement of the battery charger ac supply circuit for 10CFR50, Appendix R, requirements will not increase the probability or possibility of an accident or malfunction of a different type. Addition of an alternate ac supply to the battery charger will not reduce the margin of safety as defined in any basis to any Technical Specification. Additions of an alternate feed and associated transfer switch to battery charger 1B-2 does not affect liquid or solid Radwaste Systems as described in the FSAR.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-005, Installation of Alternate 480 Vac Feed for Unit 1 Battery Charger 1B-1 From Motor Control Center (MCC) 1XB and the Associated Transfer Switch

FUNCTIONAL SUMMARY: This modification shall accomplish the following items: It shall provide an 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 sunt 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." The FSAR change will indicate the alternate source and associated transfer switches to battery chargers 1B-1, 1B-2, 2B-1, and 2B-2 to meet the requirements of 10CFR50, Appendix R, FSAR, Section 8.3.2.1.2.

SAFETY SUMMARY: Addition of an alternate ac source and associated transfer switch to the battery charger for 10CFR50, Appendix R, requirements will not increase the probability of an accident previously evaluated. Addition of alternate circuits and transfer switch does not affect equipment evaluated in Chapter 15, thus the consequences of an accident will not be increased. 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 previously evaluated. Enhancement of the battery charger ac supply circuits in no way adversely effect safety-related equipment, thus will not increase the consequence of an evaluated malfunction. Enhancement of the battery charger ac supply circuit for 10CFR50, Appendix R, requirements will not increase the probability or possibility of an accident or malfunction of a different type. Enhancement of the battery charger ac supply circuits will not reduce the margin of safety as defined in the basis to any Technical Specification. Adding an alternate 480V feed to battery charger, 1B-1, will not affect liquid, solid, and/or gaseous radwaste systems.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-020, Installation of Instrumentation for Unit 2 Reactor Coolant Recirculation Pumps C002A and C002B in Order to Monitor the Pumps' Shaft Seals Staging Water Outflow and the Seals Leakage Outflow

FUNCTIONAL SUMMARY: This plant modification has been developed to provide a seal flow monitoring instrumentation system for both RCR Pumps C002A and C002B in Unit 2. At the present time, no instrumentation exists, although some of the components from the previous instrumentation system have been spared and still remain.

SAFETY SUMMARY: The scope of this plant modification is to install seal flow monitoring instrumentation for the RCR pumps C002A and C002B. At the present time there is no seal flow monitoring instrumentation. The instrumentation shall be installed in the following:

1. Outer seal leakage discharge lines downstream of piping supplied by the pump vendor.
2. Seal staging flow and inner seal leakage discharge lines downstream of piping supplied by the pump vendor.

Items 1 and 2 are not Q-List and no nuclear safety concerns will exist or be created by this activity.

No Technical Specifications for surveillance or operability requirements exist for the RCR pumps seal flow monitoring instrumentation.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 85-056, Elimination of the Unit 2 Residual Heat Removal System Steam Condensing Mode

FUNCTIONAL SUMMARY: The purpose of the modification is to remove a potential high energy source from the Unit 2 Reactor Building. This plant modification resulted as part of the 79-01B commitment to include the effects on high energy line breaks (HELB) inside and outside of containment. United engineers conducted a study in which all high energy pipe routes were identified, break location evaluated, and cases with the worst impact to the surrounding equipment, analyzed. The results and conclusions are compiled in Report No. 9527-058-S-MS-001, Reactor Building Environmental Report.

As a result of the study, the steam condensing line to the RHR heat exchangers from the HPCI steam line was determined to be a severe high energy source. The ten-inch line in question, penetrates through the roof of the pipe tunnel (RCIC Room), splits and runs to each RHR Room on elevation 20 ft. The isolation valves for these lines are located above the tunnel in the supply lines to each RHR heat exchanger. The portion of the line up to the isolation valves contains steam at main steam pressure and temperature. The analysis determined that if there was a break in this line, outside the pipe tunnel before the isolation valves, main steam would flow into the Reactor Building rendering safety-related equipment out-of-service.

This modification installs a cap in line 2-E41-21-10-606 inside the pipe tunnel. Since the steam condensing mode is seldom used, the new cap will render this mode of operation, inoperable. This will move the high energy boundary inside the pipe tunnel, alleviating environmental concerns associated with the line.

This modification will also install caps on line 2-E11-47-12-300 and line 2-E11-49-12-300. These lines will be abandoned in place.

This modification will also relocate the instrument taps to 2-E11-PTN026A and 2-E11-N026B to a point near the top of the respective RHR exchangers.

Refer to memorandum BESU-85/026 on page C62 of this PM.

SAFETY SUMMARY: Deleting steam condensing mode does not create the possibility of an accident or malfunction of a different type than previously evaluated in FSAR. No margin of safety for any Technical Specification has been reduced. Minor changes to process drawings are required to show installation of pipe caps. The installation of pipe caps will help prevent loss of safety-related equipment in case of a line break. No Radwaste Systems are affected by this modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-370, Automation of the Unit 2 Automatic Depressurization System (ADS) to Eliminate Need to Manually Depressurize the Reactor for Transient and Accident Events Which Do Not Directly Pressurize the Drywell.

FUNCTIONAL SUMMARY: The Automatic Depressurization System (ADS), through dedicated safety/relief valves, functions as a backup to the operation of the High Pressure Injection Systems [Feedwater, High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC)] for protection against excessive fuel cladding heatup upon loss of coolant over a wide range of steam or liquid line breaks inside the drywell. The ADS depressurizes the reactor vessel, permitting operation of the low pressure injection systems [Condensate, Low Pressure Coolant Injection (LPCI), Core Spray (CS)]. The ADS is currently activated upon coincident signals of low water level in the RPV, high drywell pressure, and a low pressure ECCS pump running. A time delay of approximately two minutes after receipt of the signals allows the operator to reset the logic and prevent an automatic blowdown if RPV water level is being restored or if the signals are erroneous. The ADS can be manually initiated as well. Manual ADS is accomplished using open/close manual switches for each individual ADS valve.

For transient and accident events which do not directly pressurize the drywell (produce a high drywell pressure) and are further deteriorated by a loss of all High Pressure Injection Systems (HPIS), adequate core cooling is assured by manual depressurization of the RPV followed by injection from the Low Pressure Injection Systems (LPIS).

This modification further automates the ADS to eliminate the need to manually depressurize the RPV for transient and accident events which do not directly pressurize the drywell. This is accomplished by removal of the high drywell pressure permissives for automatic depressurization. For events which do not pressurize the drywell, automatic depressurization will occur on low RPV water level after approximately a two-minute time delay if a low pressure ECCS pump is running. the same automatic depressurization sequence will occur for events which do pressurize the drywell. The ADS will now function as a backup to the operation of the HPCIS for loss of coolant both inside and outside the drywell. An automatic depressurization inhibit switch is being added to provide the operator with a means of preventing automatic depressurization of the RPV if the coolant inventory can be brought under control or restored through other means (i.e., feedwater, HPCI, RCIC). Also the ADS would be inhibited during an anticipated transient without scram (ATWS) event once boron injection has been initiated.

These modifications of the ADS logic satisfy the requirements of NUREG-0737, Item II.K.3.18 to provide additional assurance of adequate core cooling for events which do not directly produce a high drywell pressure signal.

TITLE: PM 84-370 (Cont'd)

SAFETY SUMMARY: The Automatic Depressurization System (ADS), through selected safety/relief valves, functions as a backup to the High Pressure Injection Systems [Feedwater, High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC)] for protection against excessive fuel cladding heatup upon loss of coolant over a wide range of steam or liquid line breaks inside the drywell. The ADS depressurizes the RPV, permitting the operation of the low pressure injection systems [Condensate, Low Pressure Coolant Injection (LPCI), Core Spray (CS)]. The ADS is currently activated automatically upon the coincident signals of low RPV water level, high drywell pressure, and a low pressure ECCS pump running. A time delay of approximately two minutes after receipt of the signals allows the operator to reset the logic and prevent an automatic blowdown if the RPV level is being restored or if the signals are erroneous. The ADS can be manually initiated as well.

For transients and accidents which do not directly produce a high drywell pressure signal and are degraded by the loss of all high pressure injection systems, adequate core cooling is assured by manual depressurization of the RPV followed by injection from the low pressure systems. Events which may require manual depressurization can be grouped into two classes:

1. RPV isolation (including breaks outside the drywell) with loss of High Pressure Makeup Systems, and
2. RPV isolations with loss of High Pressure Makeup Systems further degraded by a stuck open relief valve (SORV).

For transients that do not cause an isolation, the main condenser is available for depressurization and ADS operation is not required.

An isolation event that is further degraded by an SORV (this case is beyond current system design for single failure) causes additional coolant inventory loss and reduces the time available for the operator to manually depressurize the RPV if that is necessary. However, even for this highly degraded event, the operator has sufficient time to manually depressurize the RPV in order to permit the operation of the Low Pressure Injection Systems. It was shown in NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", Section 3.5.2.1, that the operator has at least 30 to 40 minutes to initiate the ADS and prevent excessive fuel cladding heatup for both of the previously described classes of events. This minimum time represents a "worst case" situation starting from full power with equilibrium core exposure and complete failure of all High Pressure Makeup Systems. Lower initial core power, low fuel exposure, control rod drive leakage flow, or partial operation of the High Pressure Systems would significantly increase the time available for operator action.

TITLE: PM 84-370 (Cont'd)

SAFETY SUMMARY (Cont'd)

The intent of NUREG-0737, Item II.K.3.18 is to provide additional assurance of adequate core cooling for events which do not directly produce a high drywell pressure signal. This intent was satisfied through the development and implementation of the symptom-oriented BWR emergency procedure guidelines (EPGs) and emergency operating procedures (EOPs). Events in each of the previously described classes are slow, well behaved, well understood transients which allow the operator sufficient time to actuate ADS if it is needed. The symptom-oriented procedures lead the operator through conditions of increasing levels of degradation (system failures) and provide specific guidance on when initiation of ADS is required. Thus, with the implementation of the new EOPs, the operator has increased information on the use of ADS and can more reliably perform the actions necessary to assure adequate core cooling for a wide range of transient and accident conditions, including events in the previously described classes. Additional assurance of adequate core cooling can be further enhanced through modification of the ADS logic.

Anticipated transient without scram (ATWS) events are also a consideration in the proposed modifications to the ADS initiation logic. For these events, it is important to prevent ADS initiation once boron injection has been initiated.

The BWR owners group (BWROG) performed a study of alternatives to the present ADS actuation logic and identified modifications that would eliminate the need for manual actuation to ensure core coverage. Seven alternatives were identified in the BWROG study. NRC evaluation of the study conclude that two of the seven alternatives are acceptable as indicated by letter to Mr. E. E. Utley (CP&L) from Mr. D. B. Vassallo (NRC), NUREG-0737, Item II.K.3.18, "ADS Logic Modification," dated June 3, 1983. CP&L committed to modify Brunswick-1 and Brunswick-2 to eliminate the high drywell pressure permissive and add manual inhibit switches to the ADS logic by letter to Mr. D. B. Vassallo (NRC) from Mr. P. W. Howe (CP&L) TMI Action Item II.K.3.18 ADS logic modification dated March 9, 1984. The modification described above is the second option outlined in the BWROG study and one of two alternatives deemed as acceptable by the NRC.

This modification eliminates the high drywell pressure trip from the current logic sequence and adds a manual switch which allows the operator to prevent (inhibit) automatic ADS initiation. The ADS sequence would then be activated on low RPV water level only. The remainder of the logic sequence remains unchanged from current design. The effect of high drywell pressure on other systems is unaffected. This modification is depicted in Figure I.

The following sections analyzes the modification of the ADS logic: 1) For its effect on assurance of adequate core cooling for the two classes of isolation events previously identified in Section 1, 2) for its effect on LOCA analyses contained in the FSAR, and 3) for its effect on ATWS mitigation (assuming alternate 3A plant modification).

TITLE: PM 84-370 (Cont'd)

SAFETY SUMMARY (Cont'd)

For these analyses, it is assumed that isolation has occurred and scram is successful but all High Pressure Injection Systems fail to operate. In order to insure adequate core cooling for these events the RPV must be depressurized to allow the low pressure makeup systems to inject. The modeling used in these analyses is the same as that which was previously used in NEDO-24708A.

Elimination of the high drywell pressure trip results in a reactor system response to the transients considered in this safety analyses similar to that for a small break LOCA event where the break flow discharges into the primary containment.

For a break inside the drywell, the high drywell pressure trip occurs before the low RPV water level trip. since the current ADS logic requires both signals, elimination of the high drywell pressure trip has no effect on ADS performance for events which pressurize the drywell.

The RPV water level response for isolation events is bounded by small break LOCA analyses where the majority of the inventory is lost, not through the break but through the cycling of the relief valves. The RPV water level response for a stuck open relief valve is essentially the same as that for a small recirculation line break. Thus the break spectrum analyses provided in existing FSAR demonstrates that adequate core cooling will be assured.

The ADS timer allows the High Pressure Systems approximately two minutes to start and restore RPV water level to above the trip setpoint. If the RPV water level is not restored before the ADS timer times out, the RPV is depressurized. The RCIC System is sized to prevent core uncover for isolations, but does not have sufficient capacity.

To restore RPV water level to above the trip setpoint and reset the ADS initiation logic within the allotted two minutes. Thus with an isolation and loss of high capacity, high pressure makeup, the RCIC could bring RPV water level under control and ADS actuation would not be needed. However, with the high drywell pressure trip eliminated, ADS actuation would occur unless manually defeated.

The addition of a manual inhibit switch to the ADS initiation logic has no effect on the automatic ADS response to isolation events.

This section discusses the impact of this modification on design calculations for proposed modifications for ATWS.

TITLE: PM 84-370 (Cont'd)

SAFETY SUMMARY (Cont'd)

Currently, a large range of modifications are being considered in response to the ATWS issue. These extend from implementation of the BWROG EPGs and no equipment modifications to requiring alternate 3A of NUREG-0460, Volume 4 (draft). In this section the modification of the ADS logic is considered in conjunction with alternate 3A design.

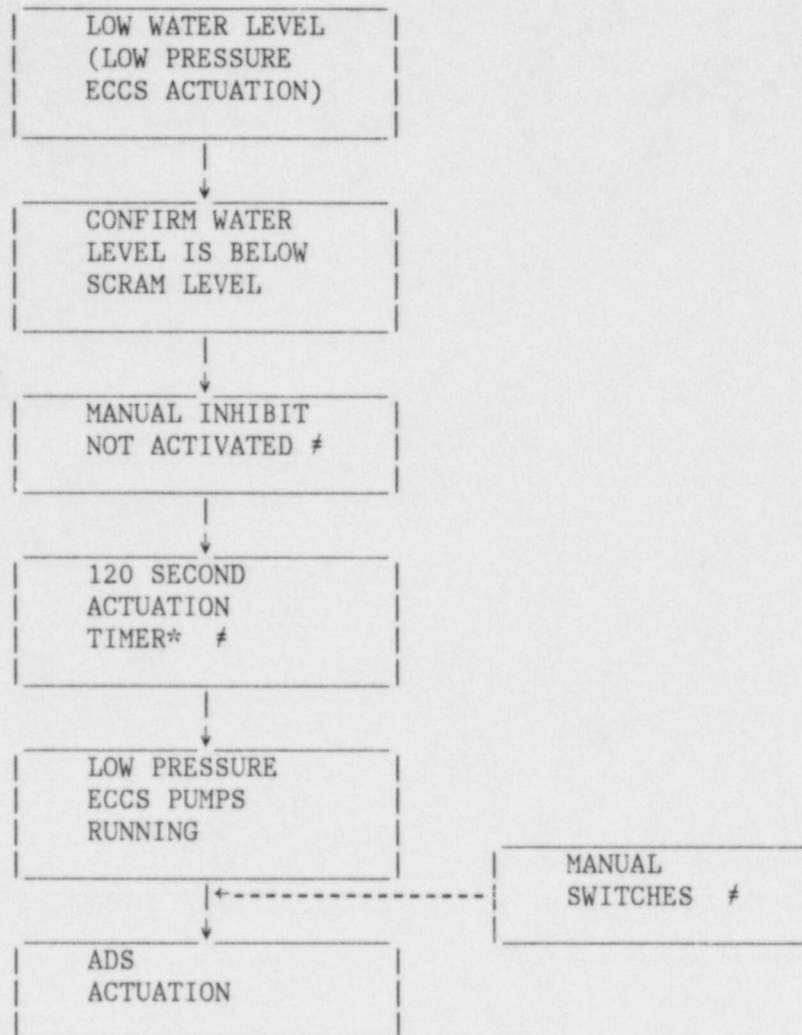
All alternate 3A calculations referenced in this section are documented in NEDE-24222, Volume 2.

Elimination of the high drywell pressure trip from the ADS initiation logic requires RPV water level to recover to above the low RPV water level trip setpoint (Level 3) before the ADS timer times out if an automatic ADS actuation is to be prevented during an ATWS event. ATWS alternate 3A design calculations performed for BWR/4-6 plant show that RPV water level is restored to above Level 3 before the timer times out for BWR/4 and BWR/5 plants. Brunswick-1 and Brunswick-2 are of the BWR/4 type design. Thus, these modifications have no impact upon ATWS alternate 3A Design.

These modifications meet all of the applicable design and licensing requirements, provide additional assurance of adequate core cooling by further automating RPV depressurization for isolations and SORV events, and satisfy ATWS modification design concerns.

TITLE: PM 84-370 (Cont'd)

SAFETY SUMMARY (Cont'd)



#Activation of manual inhibit will reset 120-second timer. ADS actuation after manual inhibit, if needed, will be performed manually.

*120-second actuation timer will reset if reactor water level recovers above trip elevation before it times out. The timer will restart if the low reactor water level signal occurs again.

Figure I, Eliminate high drywell pressure trip and add manual inhibit switch (Option 2).

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-138, Installation and/or Relocation of Emergency Plant Lighting in the Unit 2 Reactor Building to Provide Additional Lighting for Access/Egress Routes to Allow Compliance With 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 six 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.

TITLE: PM 84-138 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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 2R3, circuit 11, shall be located as shown on sketch.
 8. One new emergency lighting unit EL-2R31, fed from panel 2R2, circuit 5, shall be located as shown on sketch.

TITLE: PM 84-138 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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 safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the probability of occurrence of any accident previously evaluated.

Emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the consequences of any accident previously evaluated in the FSAR.

Emergency lighting units will be seismically supported such that failure during a postulated seismic event would not impact equipment important to safety previously evaluated in the FSAR.

Operation or failure of the emergency lighting units would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The lighting units and conduit will be supported to preclude structural failure or impact upon other equipment during a seismic event.

The emergency lighting units do not directly or indirectly affect the operation of equipment important to safety and would not create the probability of an accident or possibility for malfunctions of equipment different than already evaluated in the FSAR.

The emergency lighting units will not reduce the margin of safety since there is no interface with any system required to mitigate the consequences of an accident. The emergency lighting units serve no nuclear safety function.

The FSAR change required only impacts the listing of emergency lighting installed in the various plant areas. No Radioactive Waste Treatment System is impacted by this modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-139, Installation and/or Relocation of Emergency Plant Lighting in the Units 1 and 2 Common Service Water System Intake Structure to Provide Additional Lighting for Access/Egress Routes to Allow Compliance With 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 emergency lighting units installed by this modification will provide lighting for access/egress in the service water intake structure 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 six 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. Service Water Intake Structure Elevation 4' 0", See Sketch SK-E-84-139-01

1. One new emergency lighting unit EL-SW7, fed from panel 2SW1, circuit 11, shall be located as shown on sketch.
2. One new emergency lighting unit EL-SW8, fed from panel 2SW1, circuit 7, shall be located as shown on sketch.
3. One new emergency lighting unit EL-SW9, fed from panel 2SW1, circuit 9, shall be located as shown on sketch.
4. One new emergency lighting unit EL-SW10, fed from panel 2SW1, circuit 11, shall be located as shown on sketch.
5. One new emergency lighting unit EL-SW11, fed from panel 2SW1, circuit 9, shall be located as shown on sketch.
6. One new emergency lighting unit EL-SW12, fed from panel 2SW1, circuit 11, shall be located as shown on sketch.

TITLE: PM 84-139 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

B. Service Water Intake Structure Elevation 20' 0", See Sketch SK-E-84-139-01

1. One new emergency lighting unit EL-SW3, fed from panel 2SWE1, circuit 1, shall be located as shown on sketch.
2. One new emergency lighting unit EL-SW4, fed from panel 2SW1, circuit 1, shall be located as shown on sketch.
3. One new emergency lighting unit EL-SW5, fed from panel 2SW1, circuit 5, shall be located as shown on sketch.
4. One new emergency lighting unit EL-SW6, fed from panel 2SWE1, circuit 5, shall be located as shown on sketch.

All conduits shall be field run. Emergency lighting unit and conduit supports will be designed in accordance with drawing 9527-L-3576, Appendix A, "Non-Seismic Conduit and Electrical Supports of Seismic, Self-Contained dc emergency Lighting System Supports," Revision 1 and later.

SAFETY SUMMARY: The emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the probability of occurrence of any accident previously evaluated.

Emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the consequences of any accident previously evaluated in the FSAR.

Emergency lighting units will be seismically supported such that failure during a postulated seismic event would not impact equipment important to safety previously evaluated in the FSAR.

Operation or failure of the emergency lighting units would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The lighting units and conduit will be supported to preclude structural failure or impact upon other equipment during a seismic event.

The emergency lighting units do not directly or indirectly affect the operation of equipment important to safety and would not create the probability of an accident or possibility for malfunctions of equipment different than already evaluated in the FSAR.

TITLE: PM 84-139 (Cont'd)

SAFETY SUMMARY (Cont'd)

The emergency lighting units will not reduce the margin of safety since there is no interface with any system required to mitigate the consequences of an accident. The emergency lighting units serve no nuclear safety function.

The FSAR change required only impacts the listing of emergency lighting installed in the various plant areas. No Radioactive Waste Treatment System is impacted by this modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-136, Installation and/or Relocation of Emergency Plant Lighting in the Unit 2 Turbine Building to Provide Additional Lighting for Access/Egress Routes to Allow Compliance With 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 Turbine Building breezeway and controlled access corridor 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 six 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 2 Turbine Building Breezeway Elevation 20' 0", See Sketch SK-E-84-136-01
 - 1. One existing lighting unit EL-B2 (lamp only), fed from panel 2T2, circuit 11, shall be relocated as shown on sketch.
- B. Unit 2 Turbine Building Access Corridor Elevation 45' 0", See Sketch SK-E-84-136-01
 - 1. One new emergency lighting unit EL-TAC6, fed from panel 2T6, 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-TAC5, fed from panel 2T6, circuit 17, shall be located as shown on sketch.
 - 3. One new emergency lighting unit EL-TAC4, fed from panel 2TE2, circuit 1, shall be located as shown on sketch.

TITLE: PM 84-136 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

All conduits shall be field run. Emergency lighting and conduit supports will be designed and installed in accordance with drawing 9527-L-3576, Appendix A, "Non-Seismic Conduit and Electrical Box Supports of Seismic, Self-Contained dc Emergency Lighting System Supports," Revision 1 and later.

SAFETY SUMMARY: Increases the number of self-contained emergency lighting units described in Table 9.5.1-3 of the FSAR. Does not alter the function or operating mode of existing units.

The emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the probability of occurrence of any accident previously evaluated.

Emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the consequences of any accident previously evaluated in the FSAR.

Emergency lighting units will be seismically supported such that failure during a postulated seismic event would not impact equipment important to safety previously evaluated in the FSAR.

Operation or failure of the emergency lighting units would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The lighting units and conduit will be supported to preclude structural failure or impact upon other equipment during a seismic event.

The emergency lighting units do not directly or indirectly affect the operation of equipment important to safety and would not create the probability of an accident or possibility for malfunctions of equipment different than already evaluated in the FSAR.

The emergency lighting units will not reduce the margin of safety since there is no interface with any system required to mitigate the consequences of an accident. The emergency lighting units serve no nuclear safety function.

The FSAR change required only impacts the listing of emergency lighting installed in the various plant areas. No Radioactive Waste Treatment System is impacted by this modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-2184, Replacement of Unit 2 Residual Heat Removal System Pump Room 2B Cooler Coils and Coils Isolation Valves 2-SW-V120 and V124 and Deletion of Coolers Minimum Flow Line and Service Water Radiation Monitor RT-58-5

FUNCTIONAL SUMMARY: This modification replaces the cooling water coils in the Residual Heat Removal (RHR) pump 2B room cooler. The new coils will be constructed of 70/30 copper-nickel water boxes and tubes. This modification will also modify the cooler inlet and outlet piping as necessary to match the nozzles of the new coils and to remove the piping to SW radiation monitor RT-58-5. This radiation monitor is also being removed. Valves 2SW-V120 and 2SW-V124 will be replaced with new aluminum-bronze valves. The new spring to open actuator procured with 2SW-V124 will eliminate the need for the air volume tank and trip valve now installed to provide valve fail open capability, and these will be deleted. All Cu-Ni to Cu-Ni flange bolts on the cooler inlet and outlet piping will be changed to monel bolts, eliminating the need for dielectric insulators. In order to provide access to the cooler, it will be necessary to temporarily remove the grating above the coolers, the cooler fan unit, and room cooler conduit, piping, and supports. The removal and reinstallation of these interferences, are detailed in the installation procedure.

Also, flow orifice 2SW-F0-1194 has been resized and replaced to return cooling unit to its original design capacity.

Minimum flow bypass lines will be removed to eliminate the degradation of piping from erosion currently being experienced. This modification will remove the minimum flow lines around RHR pump room cooler 2B to prevent degradation of the Cu-Ni pipe and tubes from erosion.

SAFETY SUMMARY: This is a replacement modification and will increase the system reliability. The replacements performed by this modification are of similar components with the same functional and operating characteristics of the original components. The replacement components have the same functional and operating characteristics of the original components. This modification will increase system reliability. Replacement components are equivalent in response to anticipated malfunctions to the original components. The replacement components are similar to the original components in design, functional, and operating characteristics. The similarity of the replacement components to the original components precludes any detrimental affect to any Technical Specification margin of safety. This modification is a component replacement on the Service Water System and does not affect any of the Radwaste Systems.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-2184, Replacement of Unit 2 Core Spray Pump Room Cooler Coils

DESCRIPTION: This modification replaces the cooling water coils in the core spray pump room cooler 2C. The new coils will be constructed of 70/30 Cu-Ni. Water boxes inlet/outlet nozzles and tubes. This modification will also modify supply and return piping to match minor changes in the coil nozzle locations.

SAFETY SUMMARY: Coil material/construction changes and cooler support have increased the reliability of the system. The probability of an accident is decreased. No new failure modes are added and the system function has not been affected.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 85-046, Installation of a Permanent Service Air Supply Line for Pressurization of the Unit 2 Primary Containment for Integrated Leak Rate Testing.

FUNCTIONAL Summary: The initial design to perform the integrated leak rate testing (ILRT) periodic test (PT) was the use of a two-inch line that connects reactor noninterruptible air (RNA) to RHR containment spray in-between valves, E11-F016B and E11-F021B. This design proved inadequate in that it did not have the required capacity to pressurize the containment in a reasonable amount of time. For this reason, temporary piping was installed each outage to perform the ILRT PT.

This plant modification installs a permanent four-inch pipe to allow pressurization of primary containment and adds a spectacle flange to the existing two-inch pipe connecting RNA and RHR containment spray. Portable air compressors will provide air to the ILRT Service Air System (SA) pipe. The ILRT SA pipe will tie-in to the RHR containment spray line between valves, E11-F016A and E11-F021A. This tie-in is made with a flanged spool piece so that the SA pipe can be disconnected from the RHR containment spray line during normal plant operation. The RHR containment spray test connection on A loop is modified by this plant modification to accommodate the ILRT SA pipe tie-in.

The ILRT SA pipe is designed for water as well as air and is available for other uses such as supplying temporary construction air during outages.

SAFETY SUMMARY: Figure 5.4.7.3, flow diagram of RHR System, shows the RHR air test connection that is modified by this plant modification. The ILRT SA pipe does not tie into any plant systems during normal plant operation and does not affect any systems previously evaluated in the FSAR. The ILRT SA pipe does not affect any plant system during normal operation, therefore, previously evaluated accident consequences are unchanged. No equipment is affected by the ILRT SA pipe during normal plant operation, therefore the probability of equipment malfunction is unchanged. No equipment is affected by the ILRT SA pipe during normal plant operation, therefore, the consequence of equipment malfunction is unchanged. The ILRT pipe does not introduce any new variables to the operating plant because it does not tie into any systems during plant operation, therefore, no new accident situation is introduced. The ILRT SA pipe does not tie into any plant system during normal plant operation and, therefore, does not affect the margin of safety as defined in the basis of the Technical Specifications.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-218Z, Replace of Unit 2 Core Spray Pump Room Cooler Coils

DESCRIPTION: This modification replaces the cooling water coils in the core spray pump room cooler 2D. The new coils will be constructed of 70/30 Cu-Ni. Water boxes, inlet/outlet nozzles, and tubes. This modification will also modify, supply, and return piping to match minor changes in the coil nozzle locations.

SAFETY SUMMARY: Coil material/construction changes and cooler support have increased the reliability of the system. The probability of an accident is decreased. No new failure modes are added and the system function has not been affected.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-288M*, Replacement of Reactor Instrument Penetration (RIP) Isolation Valve at Unit 2 Primary Containment Penetration, 2-X-69, With Excess Flow Check Valves

FUNCTIONAL SUMMARY: The purpose of this modification is to replace existing air-operated isolation valves and flow switches on the instrument lines exiting from drywell penetration, 2-X-69, with excess flow check valves and manual isolation valves. 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 in the valve operators and frozen solenoid valves caused by moisture in the Instrument Air System.

The new design will install excess flow check valves that close on high flow across the valve seat and manual isolation valves which will be used for long term isolation. The excess flow check valve will remain closed after isolating until the differential pressure across the valve seat equalizes by energizing the valve reset solenoid by means of a remote control switch in the Control Room. Position switches will be utilized to provide remote annunciation and indication of valve position in the Control Room. The design function of providing containment isolation in the event of an instrument line break downstream of the valves, will remain the same.

A test connection valve will be installed between the EFCV and the instrument rack to allow the performance of required periodic testing and maintenance.

Air-operated valves, 2-B21-F042B, 2-B21-F044B, 2-B21-F046B, 2-B21-F048B, and 2-IA-PB-1217E along with the associated RNA System equipment will be removed. Flow switches 2-B21-FS-F043B, 2-B21-FS-F045B, 2-B21-FS-F047B, and 2-B21-FS-F049B will also be removed. At penetration 2-X-69 (Ports A through F), the new installation is comprised of valcor excess flow check valves in series with manual isolation globe valves and test connection valves. See the matrix below for unique identification of equipment. NOTE: Excess flow check valve, 2-B21-F049D, and manual isolation valve, 2-B21-F048D, are existing equipment at F port for penetration, 2-X-69; they will be replaced with new equipment as listed below.

TITLE: PM 82-288M* (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

Penetration Number	Line Number	EFCV Number	Manual Isolation Valve Number	Test Connection Valve Number
2-X-69A	2-B21-707	2-B21-F042B	2-B21-V48	2-B21-V118
2-X-69B	2-B21-708	2-B21-F044B	2-B21-V49	2-B21-V119
2-X-69C	2-B21-709	2-B21-F046B	2-B21-V50	2-B21-V120
2-X-69D	2-B21-710	2-B21-F048B	2-B21-V51	2-B21-V121
2-X-69E	2-B21-776	2-B21-IV-2149	2-B21-V53	2-B21-V155 To Be Installed Per PM 83-112
2-X-69F	2-B21-774	2-B21-F049D	2-B21-V52	2-B21-123

NOTE: PM 83-112 to complete installation of piping on each side of penetration 2-X-69E. The scope of work for this modification will be to install EFCV and manual isolation valve only. (Reference FSI-7080, sheets 15-14.)

SAFETY SUMMARY: Valve replacement is a one-for-one changeout of existing air-operated valves for new self-actuating valves of an improved design. All valves are subject to a complete functional test prior to operability. The design function of providing containment isolation in the event of a break in the system pressure boundary outside of the primary containment remains unchanged. A decrease in system failure is expected due to the increased reliability of new equipment. This modification involves a one-for-one changeout, installing new self-actuated valves of increased reliability. The new valves serve an identical function to those removed. This modification does not introduce any equipment or design which poses any new failure mode to the system. This modification reduces potential failure modes by eliminating an active system (instrument air) previously required for valve actuation. The existing 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 self-actuated, reducing the dependence on any support systems. The existing isolation valves are maintenance concerns, generally resulting from moisture in the Instrument Air System and faulty diaphragms in the operators. Replacement valves, while new in design, should be less of a concern to Maintenance.

The modification does not alter any safety margin as defined by the Technical Specifications for BSEP Unit 2.

This modification does not affect any Radioactive Waste Treatment System.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-135, Installation and/or Relocation of Emergency Lighting on Units 1 and 2 to Provide Additional Lighting for Access/Egress Routes in Accordance With 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 Assessments (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 six 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. The 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-37, fed from panel 1T2, circuit 9, shall be located as shown on sketch.
 2. One existing lighting unit EL-85 (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 weather-proof fixtures.
 2. One new emergency lighting unit EL-TAC2, fed from panel 1T6, circuit 13, 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.

TITLE: PM-84-135 (Cont'd)

FUNCTIONAL-SUMMARY (Cont'd)

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, "Non-Seismic 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 directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the consequences of any accident previously evaluated in the FSAR.

Emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the probability of occurrence of any accident previously evaluated.

Emergency lighting units will be seismically supported such that failure during a postulated seismic event would not impact equipment important to safety previously evaluated in the FSAR.

Operation or failure of the emergency lighting units would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The lighting units and conduit will be supported to preclude structural failure or impact upon other equipment during a seismic event.

The emergency lighting units do not directly or indirectly affect the operation of equipment important to safety and would not create the probability of an accident or possibility for malfunctions of equipment different than already evaluated in the FSAR.

The emergency lighting units will not reduce the margin of safety since there is no interface with any system required to mitigate the consequences of an accident. The emergency lighting units serve no nuclear safety function.

The FSAR change required only impacts the listing of emergency lighting installed in the various plant areas. No Radioactive Waste Treatment System is impacted by this modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-112, Installation of Fire Stop and Cable Coating at Unit 2 Reactor Building 50-foot Elevation

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, 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 ft and 80 ft elevations of the Reactor Building were inspected and found to be adequately sealed:

<u>Tray No.</u>	<u>Location</u>
44Q/CA	Column S-20R
44P/BA	Column S-20R
44N/DA	Column S-20R
43P/DB	Column L-22R, 23R
43P/CB	Column L-22R, 23R
43P/BB	Column L-22R, 23R

Redundant safe shutdown systems are located on the 50-ft elevation of the Unit 2 Reactor Building these systems are represented by train A and B reactor instrument racks located on the northeast and southeast sides of the drywell wall. With the 20-ft separation provided on the east side of the 50-ft elevation, the only feasible propagation path for a fire which could threaten both trains, is across the west side and along the south end of elevation 50 ft. To reduce the possibility of a fire involving both redundant trains, a fire stop will be installed for the exposed cables in the west center zone of elevation 50 ft. This modification package will provide for the installation

TITLE: PM 84-112 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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 in the tray within 5 ft of either side of the fire stop will be coated, regardless of whether the cable penetrates the fire stop, 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 represents a passive fire protection feature that does not affect any systems or components important to safety. The installation of a passive fire stop will not create the possibility for an accident or malfunction of any plant systems. The margin of safety as defined in the Technical Specifications is unaffected by the work in this modification package. There are no new Technical Specifications or changes to existing Technical Specifications required as a result of the work in this modification package. FSAR, Section 8.3.1.4.4 will be updated to include reference to the fire stop installed by this modification package.

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 & Betts flame-safe fire-retardant coating.

The installation of a fire stop at Unit 2 Reactor Building, elevation 50' 0", does not introduce any adverse environmental concerns.

The installation of a cable tray fire stop does not affect any Radwaste Systems. The FSAR will be updated to include reference to the fire stop.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-111, Installation of Fire Stop and Cable Coating at Unit 1 Reactor Building, 50-foot elevation

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, 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 ft and 80 ft elevations of the Reactor Building were inspected and found to be adequately sealed:

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

Redundant safe shutdown systems are located on the 50-ft elevation of the Unit 1 Reactor Building. These systems are represented by train A and B reactor instrument racks located on the northeast and southeast sides of the drywell wall. With the 20-ft separation provided on the east side of the 50-ft elevation, the only feasible propagation path for a fire which could threaten both trains, is across the south side of elevation 50 ft. 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 ft.

TITLE: PM 84-111 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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 five ft 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 represents a passive fire protection feature that does not affect any systems or components important to safety. The installation of a passive fire stop will not create the possibility for an accident or malfunction of any plant systems. The margin of safety as defined in the Technical Specifications is unaffected by the work in this modification package. There are no new Technical Specifications or changes to existing Technical Specifications required as a result of the work in this modification package. FSAR, Section 8.3.1.4.4 will be updated to include reference to the fire stop installed by this modification package.

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 & Betts flame-safe fire-retardant coating.

The installation of a fire stop at Unit 1 Reactor Building, elevation 50' 0", does not introduce any adverse environmental concerns.

The installation of a cable tray fire stop does not affect any Radwaste Systems. The FSAR will be updated to include reference to the fire stop.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-137, Installation of Emergency Lighting Units and Hand-Held Units on Unit 1 to Provide for Compliance With 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 six 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.

TITLE: PM 84-137 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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.

TITLE: PM 84-137 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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: Increases the number of self-contained emergency lighting units described in Table 9.5.1-3 of the FSAR. Does not alter the function or operation of existing units. This modification does not directly or indirectly impact systems important to safety as described in the FSAR analysis or basis for Technical Specifications.

The emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the probability of occurrence of any accident previously evaluated.

Emergency lighting units do not directly or indirectly affect operation of safety-related systems credited in the FSAR (Chapter 15). Therefore, operation or failure of an emergency lighting unit would not increase the consequences of any accident previously evaluated in the FSAR.

Emergency lighting units will be seismically supported such that failure during a postulated seismic event would not impact equipment important to safety previously evaluated in the FSAR.

Operation or failure of the emergency lighting units would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The lighting units and conduit will be supported to preclude structural failure or impact upon other equipment during a seismic event.

The emergency lighting units do not directly or indirectly affect the operation of equipment important to safety and would not create the probability of an accident or possibility for malfunctions of equipment different than already evaluated in the FSAR.

TITLE: PM 84-137 (Cont'd)

SAFETY SUMMARY (Cont'd)

The emergency lighting units will not reduce the margin of safety since there is no interface with any system required to mitigate the consequences of an accident. The emergency lighting units serve no nuclear safety function.

The FSAR change required only impacts the listing of emergency lights installed in the various plant areas. No Radioactive Waste Treatment System is impacted by this modification.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-012, Removal of Unit 2 Reactor Recirculation Pump Motor-Generator Sets A and B Speed Control Loops Master Controller, Speed Demand Limiter, Error limiting Networks, and Speed Controllers to Replace M/A Transfer Station With Manual Controllers and Install a Manual Reset for Speed Limiter No. 1 Runback

FUNCTIONAL SUMMARY: This modification will perform the following within the Reactor Recirculation Pump/MG Sets Control System:

1. Removes the master controller (2-B32-R620) and speed demand limiter (2-B32-K615).
2. Removes the error limiting networks (2-B32-K620A and 2-B32-K620B) and speed controllers (2-B32-R622A and 2-B32-R622B).
3. Replaces the M/A transfer stations with manual control stations (2-B32-R621A and 2-B32-R621B).
4. Replaces the feedwater interlock timers (2-B32-K23A and 2-B32-K23B).
5. Installs a manual reset for speed limiter No. 1 bypass relay (2-B32-K2A and 2-B32-K2B).
6. Makes wiring corrections in 2-B32-P003A and 2-B32-P003B (reactor recirculation MG set cubicles).
7. Corrects drawing errors on 2-FP-5572, sheet 3.

SAFETY SUMMARY: 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 will improve reliability of the control system reducing probability of failure. It will have no affect on the operating parameters of the fluid coupler, therefore, the probability of occurrence of an accident will be reduced and consequences of such an accident will not be increased.

This modification affects no Q-List equipment. It will have no affect on required preventative maintenance or inspections. Therefore, the probability of occurrence or consequences of malfunction of equipment important to safety previously evaluated in the FSAR are not increased.

This modification will only affect the Reactor Recirculation System, it will have no affect on any Q-List equipment, therefore, the probability of an accident or possibility for malfunction of equipment important to safety of a different type than already evaluated in FSAR will not be created.

TITLE: PM 86-012 (Cont'd)

SAFETY SUMMARY (Cont'd)

This modification does not conflict with or require a change to Technical Specifications. It does not affect the requirements or change the assumptions of any Technical Specifications, therefore, the margin of safety is not reduced.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-104, Installation of Two Lines of Closed Head Sprinklers Adjacent to Draft Stops at the Ceiling of the Unit 2 Reactor Building, 20-foot Elevation

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, 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 (SSD) 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 20 foot elevation. Separation zones of 20-ft 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 inside the separation zone are to be installed. A draft stop is provided in conjunction with each set of sprinkler lines consisting of existing structures or the addition of new draft stops. Although this is not a floor opening, the objective of limiting fire spread past a given vertical plane is the same. One sprinkler line/draft curtain combination will be oriented to prevent fire spread from north to south. The other sprinkler line/draft curtain will prevent fire spread from south to north. Sprinkler piping on the east side will use existing concrete

TITLE: PM 84-104 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

beams as draft stops. On the west side, a sheet metal draft stop will be installed between the sprinkler lines. Baffles will be installed to prevent one sprinkler from spraying or "cold-soldering" the adjacent sprinklers. Baffles will be affixed to the pipe in all cases.

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 six 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-3971. The water curtain is hydraulically calculated and designed. All pipe components are FP-Q. The piping, draft stops, baffles, smoke detectors, and conduit will be supported by seismically designed (non-safety) 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.

To provide for adequate fire protection in Zone 7 where the sheet metal draft curtain is being installed, the existing smoke detector will be removed from one side of the draft stop and two new additional smoke detectors will be installed, one on either side of the draft stop. Thus, smoke on either side of the draft stop will be detected and the appropriate alarm system actuated.

An evaluation was made on the potential for damage to safety-related equipment by water spray. Spray shields are being installed on motor control center vents on MCC 1XC and MCC 1XM to protect these components from water spray to assure proper operation.

SAFETY SUMMARY: This change enhances the protection of redundant safety features by adding protective water curtains in the separation zones to prevent propagation of fire across the zone. FSAR, paragraph 9-5.1.1 through 9-5.1.4. Modification reduces the potential for fire to propagate from one safety train to the other through special separation zones and water curtains. The additional water suppression system has no affect on accidents evaluated in FSAR (Chapter 15). There is no direct or indirect interface. Activation of sprinklers will reduce probability of malfunction by limiting fire damage to one train and preclude the potential spread of fire. This modification does not interface with safety systems, and the consequences of malfunction of safety equipment remain unchanged. Activation of sprinkler system does not threaten any equipment identified as important to safety. A flooding analysis has been performed and verifies that safety-related equipment is not affected. The need for spray shields has been evaluated and these are being provided where required in this modification. Smoke detector

TITLE: PM 84-104 (Cont'd)

SAFETY SUMMARY (Cont'd)

alarm systems were reviewed. One smoke detector is being removed and two additional smoke detectors are being added to assure proper alarm functioning. Margin of safety will be increased due to the enhancement of the separation zone between safety trains with water suppression. FSAR changes are required to include this installation/system and revise the descriptive/location information in Section 9-5.1.1 and 9-5.1.4 this modification does not affect any Radwaste System.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-103, Installation of Two Lines of Closed Head Sprinklers Adjacent to Draft Stops at the Ceiling of the Unit 1 Reactor Building 20-ft Elevation

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 alternative shutdown capability assessment (ASCA) report. Based on this, 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 (SSD) 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 20-foot elevation, separation zones of 20-ft width and 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 located inside the separation zone are to be installed. A draft stop is provided in conjunction with each set of sprinkler lines consisting of existing structures or the addition of new draft stops. Although this is not a floor opening, the objective of limiting fire spread past a given vertical plane is the same. One sprinkler line/draft curtain combination will be oriented to prevent fire spread from north to south. The other sprinkler line/draft curtain will prevent fire spread from south to north. Sprinkler piping on the east side will use existing concrete

TITLE: PM 84-103 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

beams as draft stops. On the west side, a sheet metal draft stop will be installed between the sprinkler lines. Baffles will be installed to prevent one sprinkler from spraying or "cold-soldering" the adjacent sprinklers. Baffles will be affixed to the pipe in all cases.

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 six 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-3971. The water curtain is hydraulically calculated and designed. All pipe components are FP-Q. The piping, draft stops, baffles, smoke detectors, and conduit 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.

To provide for adequate fire protection in Zone 7 where the sheet metal draft curtain is being installed, the existing smoke detector will be removed from one side of the draft stop and two new additional smoke detectors will be installed, one on either side of the draft stop. Thus, smoke on either side of the draft stop will be detected and the appropriate alarm system actuated.

An evaluation was made on the potential for damage to safety-related equipment by water spray. Spray shields are being installed on motor control center vents on MCC 1XC and MCC 1XM to protect these components from water spray to assure proper operation.

SAFETY SUMMARY: This change enhances the protection of redundant safety features by adding protective water curtains in the separation zones to prevent propagation of fire across the zone. FSAR, paragraph 9-5.1.1 through 9-5.1.4. Modification reduces the potential for fire to propagate from one safety train to the other through special separation zones and water curtains. The additional water suppression system has no affect on accidents evaluated in FSAR (Chapter 15). There is no direct or indirect interface. Activation of sprinklers will reduce probability of malfunction by limiting fire damage to one train and preclude the potential spread of fire. This modification does not interface with safety systems, and the consequences of malfunction of safety equipment remain unchanged. Activation of sprinkler system does not threaten any equipment identified as important to safety. A flooding analysis has been performed and verifies that safety-related equipment is not affected. The need for spray shields has been evaluated and these are being provided where required in this modification. Smoke detector

TITLE: PM 84-103 (Cont'd)

SAFETY SUMMARY (Cont'd)

alarm systems were reviewed. One smoke detector is being removed and two additional smoke detectors are being added to assure proper alarm functioning. Margin of safety will be increased due to the enhancement of the separation zone between safety trains with water suppression. FSAR changes are required to include this installation/system and revise the descriptive/locational information in Section 9-5.1.1 and 9-5.1.4 this modification does not affect any Radwaste System.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-102, Installation of a Line of Closely Spaced Closed Head Sprinklers Adjacent to Draft Stops in the Unit 2 Reactor Building, 17-foot Elevation

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, 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 (SSD) 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 five-foot elevation, separation zones of 20-ft width and 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 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 from south to north. Existing concrete beams are utilized as a draft stop on the south side of the separation zone, and the existing 20-inch RHR piping in conjunction with the existing concrete beam on the north side of the separation zone acts as the draft stop. Baffles will be installed to prevent

TITLE: PM 84-102 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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 six ft (18 gpm), and no sprinkler will discharge less than 15 gpm.

The Piping system is an extension of the existing 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: This change enhances the protection of redundant safety features by adding protective water curtains at the periphery of the separation zone to prevent fire propagation across the zone. FSAR, Table 9-5.1.1 through 9-5.1.4. Modification reduces the potential for fire to propagate from one safety train to the other through spatial separation zones and water curtains. The additional water suppression system has no affect on accidents evaluated in FSAR (Chapter 15). There is no direct or indirect interface. Activation of sprinklers will reduce probability of malfunction by limiting fire damage to one train and preclude the potential spread of fire. This modification does not interface with safety systems, and the consequences of malfunction of safety equipment remain unchanged. Activation of sprinkler system does not threaten any equipment identified as important to safety. A flooding analysis was performed to verify that sprinkler actuation does not affect safety equipment. The effect of water spray on safety systems has been evaluated, with no additional spray protection being required. The margin of safety will be increased due to the enhancement of the separation zone between safety trains with water suppression. Locational and descriptive changes are required for FSAR Tables 9.5.1-1 and 9.5.1-4. This modification does not affect any Radwaste Systems.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-101, Installation of a Line of Closely Spaced Closed Head Sprinklers Adjacent to Draft Stops in the Unit 1 Reactor Building, 50-foot Elevation

FUNCTIONAL SUMMARY: In accordance with the requirements of 10CFR50, Appendix R, "...Licensees should reexamine those previously approved configurations of fire protection that does 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, 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-ft width and 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 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 from south to north. Existing concrete beams are utilized as a draft stop on the south side of the separation zone, and the existing 20-inch RHR piping in conjunction with the existing concrete beam on the north side of the separation zone acts as the draft stop. Baffles will be installed to prevent

TITLE: PM 84-101 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

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 ft. Sprinkler spacing is at a maximum of six ft (18 gpm), and no sprinkler will discharge less than 15 gpm.

The Piping system is an extension of the existing 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: This change enhances the protection of redundant safety features by adding protective water curtains at the periphery of the separation zone to prevent fire propagation across the zone. FSAR, Table 9-5.1.1 and 9-5.1.4. Modification reduces the potential for fire to propagate from one safety train to the other through spatial separation zones and water curtains. The additional water suppression system has no affect on accidents evaluated in FSAR (Chapter 15). There is no direct or indirect interface. Activation of sprinklers will reduce probability of malfunction by limiting fire damage to one train and preclude the spread of fire. This modification does not interface with safety systems, and the consequences of malfunction of safety equipment remain unchanged. Activation of sprinkler system does not threaten any equipment identified as important to safety. A flooding analysis was performed to verify that sprinkler actuation does not affect safety equipment. The effect of water spray on safety systems has been evaluated, with no additional spray protection being required. The margin of safety will be increased, due to the enhancement of the separation zone between safety trains with water suppression. Locational and descriptive changes are required for FSAR Tables 9.5.1-1 and 9.5.1-4. The modification does not affect any Radwaste Systems.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-107, Installation of a Line of Closely Spaced Closed Head Sprinklers Between Redundant Trains in the Unit 1 Reactor Building, 20-ft Elevation, Southwest

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, 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 a single line of close spaced, closed head sprinklers between the redundant trains, 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 (SSD) 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.

The southwest area of the Unit 1 Reactor Building 20-ft elevation, will be provided with sprinklers utilizing the guidance of NFPA-13. The sprinkler heads are temperature actuated closed heads, (165°F) and will supply a minimum of 3 gpm/linear ft. Sprinkler spacing is at a maximum of six ft (18 gpm), and no sprinkler will discharge less than 15 gpm. Baffles will be installed to prevent one sprinkler from spraying or "cold soldering" the adjacent sprinklers. In other cases, existing structural members will function as baffles. 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-3972. 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 or walls to which they are attached.

TITLE: PM 84-107 (Cont'd)

FUNCTIONAL SUMMARY (Cont'd)

An evaluation was made on the potential for damage to safety-related equipment by water spray. Spray shields are being installed on motor control center vents MCC 1XK to protect this component from water spray to assure proper operation.

SAFETY SUMMARY: This change enhances the protection of redundant safety features by adding a protective water curtain between redundant trains to prevent fire propagation across the zone. FSAR, Table 9-5.1.1 and 9-5.1.4. Modification reduces the potential for fire to propagate from one safety train to the other by installation of this water curtain between the trains. The additional water suppression system has no affect on accidents evaluated in FSAR (Chapter 15). There is no direct or indirect interface. Activation of sprinklers will reduce probability of malfunction by limiting fire damage to one train and preclude the spread of fire. This modification does not interface with safety systems, and the consequences of malfunction of safety equipment remain unchanged. Activation of the sprinkler system does not threaten any equipment identified as important to safety. A flooding analysis was performed and verifies that safety-related equipment is not affected. The need for spray shields has been evaluated and these are being provided where required in this modification. The margin of safety will be increased, due to the enhancement of the separation zone between safety trains with water suppression. Locational and descriptive changes are required for FSAR Tables 9.5.1-1 and 9.5.1-4. The modification does not affect any Radwaste Systems.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 80-134, Installation of Dedicated Containment Atmospheric Dilution (CAD) Nitrogen Injection on Unit 2 for Containment Hydrogen Control During a Postaccident Mode as Per Requirements of NUREG-0737, Item II.E.4.1

FUNCTIONAL SUMMARY: The purpose of this modification is 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 II.E.4.1.

The upgraded system will be designed to meet the above requirement by having 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 will include the following:

The CAD nitrogen injection piping will be 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 will be provided.

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

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

The nitrogen supply from the CAD tank to be Noninterruptible Instrument Air System will be removed and the line capped.

In addition to the above work this modification will also provide the following:

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

All new control valves will be solenoid operated.

High drywell pressure (40 psig) isolation interlock to inerting, makeup, and CAD valves will be removed. Annunciation will still exist.

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

TITLE: PM 80-134 (Cont'd)

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. Also, installation 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 nor does it 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 will be increased by designing the system against single failure of active components. The margin of safety for containment isolation will not be reduced.

CHANGE TO FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-196, Installation of a Safety Grade, Environmentally Qualified, Pneumatic (Nitrogen) Source to Function as a Backup to the Instrument Air System on Unit 2

FUNCTIONAL SUMMARY: This modification will install a safety grade, environmentally qualified, pneumatic (nitrogen) source to function as a backup to the Instrument Air System. The Nitrogen Backup System will provide pneumatic requirements of the Automatic Depressurization System (ADS) valves and the torus to Reactor Building vacuum breaker butterfly valves, 2-CAC-V16 and V17.

The backup system is designed to provide short and long term supply to the ADS valves and a redundant supply to the vacuum breaker butterfly valves. Short term supply will be provided from two redundant bottle racks to be located on the 50-ft elevation of the Reactor Building. Piping from this supply penetrates primary containment and connects to the ADS pneumatic headers. Additional piping branches off to feed the pneumatic headers to the vacuum breaker butterfly valves.

Long-term supply will be provided from two redundant supply lines which will be connected to a compressed nitrogen source outside the Reactor Building. The supply lines tie into the main backup headers from the bottle racks.

The backup system is normally operational with direct feed to the instrument air headers separated by check valves. When instrument air pressure falls below ninety-five pounds per square inch gauge, the backup system which maintains pressure at ninety-five pounds per square inch gauge automatically feeds the instrument air headers. The nitrogen supply pressure will have both local and remote (Control Room) indication as well as Control Room annunciation of low supply pressure from the bottle bank. Supply nitrogen to the ADS headers can also be controlled locally or remotely via remote manual solenoid valves to be installed outside primary containment penetration.

This modification will replace the motor operated instrument air drywell isolation valves, 2-RNA-V101 and V103, with direct acting solenoid valves. The valves will close on a LOCA signal to minimize air inleakage and oxygen introduction to the drywell post LOCA. Valves will be provided with a keylock switch to allow operator override as required.

TITLE: PM 84-196 (Cont'd)

SAETY SUMMARY: This modification adds the Nitrogen Backup System to the Compressed Gas System. The Nitrogen Backup System will provide an environmentally qualified, redundant, safety-related pneumatic source to safety-related, pneumatic valves which are required to be operable post LOCA. These valves are currently supplied by the Noninterruptible Instrument Air System, a redundant safety-related, pneumatic source. The addition of the backup system will increase operational reliability of valves supplied and will not increase the probability or consequences of an accident or malfunction.

The backup system introduces pressurized nitrogen cylinders into the Reactor Building. The cylinders introduce two concerns, missile potential and nitrogen leakage. The conclusion has been reached that neither of these concerns introduce a safety concern or contribute to the increased probability of an existing concern. This conclusion is based on the following:

- Nitrogen cylinders and lines are located in well ventilated, open areas where nitrogen pockets will not accumulate.
- Nitrogen cylinders are supported in seismic racks to preclude failure due to seismic event.
- Nitrogen cylinders are furnished with a rupture disc and each cylinder bank header is provided with a pressure relief valve.
- Nitrogen cylinders shall be hydrostatically tested at regular intervals as required by DOT 178.37-3AA.
- Nitrogen bottles located in low traffic areas to minimize external interfaces.

This modification also replaces motor operated containment isolation valves, 2-RNA-V101 and 2-RNA-V103 with direct acting solenoid valves. The existing valves are remote manual, while the replacement valves will have the added feature of automatic isolation via isolation signal from LOCA logic. The automatic isolation feature allows isolation of air to drywell post LOCA, thereby, reducing an oxygen source to the drywell. Oxygen reduction serves to mitigate post LOCA drywell affects. The automatic isolation feature eliminates pneumatic supply to drywell valves, but a review of valve design bases indicates valves will fail to design safe condition and require no further operation. Exceptions to this are the ADS valves (which will be supplied by backup system and remain operable) and the MSIVs (which under certain conditions are required operable). An override switch and operating procedures allow the MSIVs to be operated. Therefore, no additional safety concerns have been created.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: 1MST-RPS43R, Revision 1, Reactor Protection System Mode Switch In
Startup Bypass Logic Test

APPROVAL DATE: May 15, 1986

DESCRIPTION: This revision deletes this procedure. Requirements of this
procedure are addressed by 1MST-RPS41R.

SAFETY SUMMARY: The deletion of this procedure does not affect nuclear safety.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: EOP-01-ACCP, Revision 3, Containment Control Emergency Operating Procedure

DESCRIPTION: To make procedure consistent with EOP Writer's Guide and reflect as-built of Unit 2 due to implementation of PM 80-134 and PM 84-196.

SAFETY SUMMARY: This procedural change will have no detrimental affect on nuclear safety as the change will reflect the as-built condition of implemented Plant Modifications 80-134 and 84-196.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: EOP-01-CCP, Revision 3, Containment Control Emergency Operating Procedure (EOP)

APPROVAL DATE: May 15, 1986

DESCRIPTION: This revision makes the procedure consistent with the EOP Writer's Guide and reflects the as-built condition of Unit 2 due to implementation of PM 80-134 and PM 80-196.

SAFETY SUMMARY: This procedure revision is administrative and will have no detrimental affect on nuclear safety.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: GP-09, Revision 1, Initial Criticality After Core Alterations

DESCRIPTION: This revision is administrative and involves changes to the procedure to provide for agreement with Periodic Test (PT)-50.2, SRM/IRM/APRM Overlay Determination.

SAFETY SUMMARY: This revision does not affect accident analysis or the margin of safety.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: PT-12.2A, Revision 24, No. 1 Diesel Generator Monthly Load Test

DESCRIPTION: This revision corrects typographical errors and misnomers in the procedure.

SAFETY SUMMARY: The revision is administrative and does not affect nuclear safety.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: MI-10-32, Revision 0, Steam Jet Air Ejector Sample Chamber Drain Procedure

DESCRIPTION: This procedure provides preventative maintenance instructions for draining the steam jet air ejector sample chamber of accumulated moisture.

SAFETY SUMMARY: This procedure will enhance plant safety and not affect previous evaluation concerning the probability of occurrence of any accident previously evaluated in the plant FSAR.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: MI-10-527B, Revision 0, Functional Performance Test of the Central Alarm Station (CAS) Air Conditioning (A/C) Units

DESCRIPTION: This procedure performs a functional performance test on the CAS A/C units to ensure their satisfactory performance.

SAFETY SUMMARY: The CAS A/C units do not affect equipment which is important to safety. In addition, malfunction of the CAS A/C units would not affect any equipment covered under Technical Specifications.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: Abnormal Operating Procedure (AOP)-31.0, Flooding in Turbine Building
Condenser Pit or Pipe Tunnel

DESCRIPTION: The changes made to the procedure involved changes in format and grammar. The scope of the procedure, however, was not changed and remains a method to assist in identifying, controlling, and recovering from or eliminating flooding of the condenser pits or pipe tunnels.

SAFETY SUMMARY: No safety concerns, as addressed in the FSAR, will be affected in a detrimental manner.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: Abnormal Operating Procedure (AOP)-30.0, Reactor Safety-Relief Valve Failures

DESCRIPTION: This procedural change was made to ensure technical accuracy of the procedure. Reference FSAR, Items 5.2.2, 5.4.13, and 15.1.5.

SAFETY SUMMARY: The changes to this procedure were made to ensure that the procedure is technically correct. The scope of the procedure has not changed and remains a method to ensure that temperature limits on the suppression pool are not approached; such that, sufficient thermal capacity exists following a postulated rupture of the Reactor Primary System. In the event a safety-relief valve becomes stuck open or leaks by, therefore, no safety concerns, as addressed in the FSAR, will be affected in a detrimental manner.

CHANGE TO PROCEDURE AS DESCRIBED IN THE FSAR

TITLE: PLP-01, Fire Protection Program Document

DESCRIPTION: This procedure lists the plant's fire protection commitments. Those commitments form a portion of the plant's fire protection program and are a means of complying with General Design Criteria (GDC)-3, Fire Protection, FSAR 95.1, is affected by this change.

SAFETY SUMMARY: The revision is made only to list commitments. There will be no changes to the facility as a result of this revision.