CP&L

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BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 & 2
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LICENSE NOS. DRP-71/DPR-62
ANNUAL REPORT IN ACCORDANCE WITH 10CFR50.59

Gentlemen:

In accordance with 10 CFR 50.59, the following annual report is submitted for 1989. This report contains brief functional summaries of procedures and plant modifications which are changes to the facility as described in the FSAR. The report also contains those tests or experiments conducted in 1989 which are not described in the FSAR.

Very truly yours,

J. L. Harness, General Manager
Brunswick Nuclear Project

TH/

Enclosure

cc: Mr. S. D. Ebneter
Mr. N. B. Le
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Table of Contents

Table of Contents

Table of Contents

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 80-062, Unit 1 Public Address System Volume Bypass

FUNCTIONAL SUMMARY: The purpose of this modification is to install remote volume bypass into the Unit #1 public address system. The volume bypass shall consist of a tone transmitter installed in the Multi-Tone (Alarm) Generator and tone receivers installed in strategically located speaker amplifiers (including wall station speaker amplifiers). This volume bypass will provide maximum volume for evacuation signals and will make the volume control tamper proof for emergency signals, but not for voice signals.

In addition, high energy flashing emergency lights will be located in Unit #1 areas where high-noise levels make the public address system alarm signal inaudible.

SAFETY SUMMARY: This modification adds remote volume bypass to the public address system which will insure that, for evacuation signals, maximum volume is provided; and that volume controls are tamper proof for evacuation signals, but not for voice signals. The system changes are being installed in accordance with manufacturers recommendations, thus assuring the proper operability of the system. Following installation, acceptance tests of the modification and routine tests of the system will insure that the P. A. system remote volume bypass functions properly.

Since the system does not interfere with other systems, it cannot impact the previous safety analysis performed for other equipment nor can it create an additional accident path. These changes do not affect the Technical Specifications and enhance the ability to ensure that evacuation alarms can be heard.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-198, Radwaste Storm Drain Basis Discharge Valve Replacement

FUNCTIONAL SUMMARY: The storm drain basin was equipped with a locked butterfly valve and a blank flange to prevent free flow from the basis and to ensure there is not an inadvertent release during periods of heavy rain. To enhance the operators ability to dewater the basin, a second butterfly valve was installed in place of the existing blank flange. This will expedite the process of overflow release and eliminate the physical labor necessary to remove the blank flange.

SAFETY SUMMARY: Replacing the existing blank flange with a second butterfly valve will not increase the possibility of inadvertent releases. The 2 valves still provide a double isolation. Administrative controls will remain in place to maintain both butterfly valves in a locked closed position. No reduction in the margin of safety will be made by use of these controls. Regulatory requirements will still be met for releases. The enhanced ability of the operations personnel to manipulate the valves will reduce the potential flooding risks of certain plant equipment.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 82-221J, SW Cross-Tie Completion

FUNCTIONAL SUMMARY: It is desired to have the capability to isolate the service water discharge piping of each unit to permit piping inspection and/or repair without restricting the operation of the Service Water System. To accomplish this, the discharge piping from each unit must be cross connected with the discharge piping of the other unit. Also, a means of isolation between the units must be established to divert the discharge flow from one unit to the other and provide isolation from the piping to be inspected and/or repaired.

To achieve the desired Unit 1 and Unit 2 service water discharge piping cross tie, three plant modifications will be required. The Unit 1 and Unit 2 work/plant modification implementation division is shown by the attached sketch.

This plant modification will add the following items to complete the Service Water Discharge Piping Cross Tie:

1. To provide double isolation on existing piping (line no 2-SM-296-30-R-1), a 30" diameter manual cast iron butterfly valve 1-SW-V443 will be installed in existing discharge cross tie piping (new line no. 1-SW-296-30-R-1) adjacent to existing valve 2-SW-V443.

2. Drain line containing valve 1-SW-V667 will be added between valves 1-SW-V443 and 2-SW-V443.

3. A manhole will be added to the existing discharge piping (line no. 1-SW 31-36-R-1) to provide access for inspection.

By the implementation of this cross tie piping by the Unit 1 and Unit 2 plant modifications, piping inspections and/or repairs may be accomplished without restricting the operation of the Service Water System.

The major components affected are the existing service Water Piping where the isolation valve and manhole will be added.

Plant Modifications 82-221D and 82-220D require completion prior to implementation of this plant modification. This plant modification must be coordinated with Plant Modification 87-240 (Service Water Piping Inspection and Repair).

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

SAFETY SUMMARY: The implementation of this modification will increase the operational capability of the service water system with no increase in accident probability. The changes in operation of the service water system resulting from this modification will not affect any previously evaluated accident. The similarity of equipment to the original equipment in design, function, and operability will not reduce system safety. The high reliability of the equipment installed by the modification will not reduce the margin of safety.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-108, Diesel Generator Building Halon Suppression

FUNCTIONAL SUMMARY: In accordance with the requirements of 10 CFR 50, 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 Plant Safe Shutdown Capability has been performed and is contained in the Alternate Shutdown Capability Assessment (ASCA) Report. Based on this report, the Brunswick Steam Electric Plant concluded it can provide equivalent protection to the requirements of 10 CFR 50, Appendix R, Section III.G. through a combination of alternative shutdown capability in accordance with Section III.G. requirements, separation of redundant functions in accordance with Section III.G.2, or provide equivalent protection for redundant functions as an alternative to the prescriptive measures of III.G.2.

Title 10, CFR 50, Appendix R, Section III.G. requires that 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 methods of providing equivalent protection for redundant functions as an alternative to Section III.G.2, and as committed to in the BSEP ASCA Report, is fulfilled by installing a halon fire suppression system in Fire Area DG-1, which is the basement (Elevation 2'-0") level of the Diesel Generator Building. A cross-zoned, supervised Halon 1301 suppression system is being installed with this modification in order to provide rapid fire suppression in the Diesel Generator Building basement.

Utilizing NFPA-STD-12A as a guide, this total flooding system will have single shot capability and be designed to achieve and maintain a concentration of six percent by volume for a minimum 10-minute soak period. Required concentrations of halon gas will be maintained in part by interlocking the basement ventilation fan with the fire detection system so that when a fire condition is detected in the basement, the exhaust fan will shut down and its associated dampers will close.

To reduce the likelihood of accidental discharge, a cross-zoned detection arrangement will be employed to automatically actuate the suppression system. All of the detectors installed for the system will be of the ionization type and are arranged such that physically adjacent detectors are associated with opposite initiating zones.

Automatic operation of the system is accomplished through the use of six ionization smoke detector zones in the ceiling area that are divided into Cross Zones A and B. Detector Zones 1, 3, and 5 are in Cross Zone A and Detector Zones 2, 4, and 6 are in Cross Zone B. Automatic operation of the system requires at least one detector from Cross Zone A and one from Cross Zone B to be activated prior to a halon discharge. The sequence of automatic operation is as follows: (1) the first detector zone in Cross zones A or B actuates, activating a halon control panel alarm, a flashing warning sign at each of two doors to the + 2' elevation, a bell sounding near the control panel and two bells sounding in the hazard area, two flashing lights in the hazard area, Air Dampers A-D-DG and S-D-DG to close an Fan N-EF-DG to stop, and an annunciator in the Plant Control Room; (2) the second detector zone actuates that is cross zones with the first detector zone that actuates followed by simultaneous activation of two alarm horns and two strobe lights in the +2' elevation, two flashing warning signs at the doors to the +2' elevation, alarm horn and strobe light at the control panel, control panel alarms, annunciation in the control room, and the nitrogen pilot cylinders at each halon station will discharge causing the 12 halon cylinders to immediately start to discharge. The halon system is designed to discharge halon from the nozzles within a nominal 10-second period.

The discharge of halon can be stopped if one of the two abort stations is depressed prior to activation of two cross zones of detection.

Manual operation with two nitrogen cylinder initiation stations at Elevation 23' and two electrical switch stations at Elevation 23' bypasses the detector circuits for immediate exhaust fan shutdown, closing of dampers, actuation of discharge audible/visual indicators, and halon discharge. The control panel circuits are electrically supervised to annunciate at the panel and in the control room should a fault occur.

The major components that will be installed in this system are as follows:

a. Local control cabinet which acts as the central terminating point for all systems cables and contains all logic modules which control the system functions.

b. Detection instrumentation which consists of 112 ionization type detectors arranged in six zones. Zones 1, 3, and 5 are in Cross Zone A and Zones 2, 4, and 6 are in Cross Zone B.

c. Halon storage is provided by 12 identically charged cylinders, each housing 500 lbs. of liquid halon which is pressurized to 360 psig at 70 degrees F by dry nitrogen.

d. Piping networks that distribute the halon throughout the entire basement level and discharge it through a combination of fifteen 360 degree and nine 180 degree nozzles.

e. Nitrogen Pilot System to discharge the halon cylinders.

f. Miscellaneous visual and audible alarms and switch devices which are integrated with the system.

The installation work contained in this modification package is considered as Fire Protection-Related and Q-list work. Conduit and cable supports, piping, tubing, and equipment supports are installed as non-safety-related seismic, or non-seismic, as applicable. Equipment and material as listed on the bill of material will be procured as Fire Protection-Q, Q-list and Non-Q material, as applicable. The power supply for the control panel is provided from a Q-list power distribution panel.

<u>SAFETY SUMMARY:</u> The installation of a Halon Fire Suppression System in the Diesel Generator Building basement will not introduce any new equipment/components or operational logic controls that would result in new operational occurrences or new postulated accidents to existing Plant Systems or Structures that would adversely impact any accident analyses previously evaluated in the FSAR (Chapter 15) and, therefore, will not increase the probability of occurrence of any accident previously evaluated in the FSAR (Chapter 15).

The installation of a Halon Fire Suppression System in the Diesel Generator Building basement will not introduce any new equipment/components or operational logic controls that would cause a malfunction of existing Plant equipment important to safety previously evaluated in the FSAR and, therefore, the probability of occurrence of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

The installation of a Halon Fire Suppression System in the Diesel Generator Building basement will not introduce any new equipment/components or operational logic controls that would result in the creation of an accident or possibility for malfunction of equipment important to safety or a different type than already evaluated in the FSAR and, therefore, 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.

The installation of a Halon Fire Suppression System in the Diesel Generator Building basement will not introduce any new equipment/components or operational logic controls that would adversely impact the margin of safety of existing Plant Systems or Structures as defined in the basis to any technical specification of the proposed technical specification for this halon system and, therefore, the margin of safety as defined in the basis to any technical specification or proposed technical specification for this halon system will not be reduced.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 84-146, Unit 2, Appendix R Communications Upgrade

FUNCTIONAL SUMMARY: The purpose of this modification was to install three sound-powered phone circuits to provide reliable communication between Unit 2 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 alternate safe shutdown activities.

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

SAFETY SUMMARY: Addition of a dedicated phone system for remote shutdown in the event of a fire improves plant reliability and does not increase the probability of a previously evaluated accident. Addition of a more reliable communication system to ensure remote shutdown capability does not increase the consequences of a previously evaluated accident, nor does it affect any safety-related equipment to the point of changing the probability of malfunction. Addition of the independent sound-powered phone system does not alter the design basis of any safety related equipment. Improved reliability to provide remote shutdown increases the margin of safety as defined in the basis of the technical specifications.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 86-033, Unit 1 Anticipated Transient Without Scram-Recirculation Pump Trip/Alternate Rod Injection

FUNCTIONAL SUMMARY: The purpose of this Plant Modification is to make changes, as described below, to the RPT System and CRD Pilot Air Header to satisfy ATWS requirements of 10CFR50.62.

The basic change to the plant is to incorporate a Control Rod insertion (SCRAM) function from the revised RPT Logic. The existing RPT Pressure Switches 1-B21-PS-N045 A,B,C & D. This change will enhance system reliability, and provide ease of testing while at power.

The RPT Trip logic is being revised so that one logic train, which consists of one high pressure signal or one low water level signal from each of two opposite safety divisions, will provide the trip initiation to trip both Recirculation pumps A & B. The initiation signals associated with each logic train are as follows:

Logic Train A:

1-B21-PTM-N045A & C trip set at 1115 psig*

1-B21-LTM-N024A-2 & 25A-2 trip set at 118 in. water*

Logic Train B:

1-B21-PTM-N045B & D trip set at 1115 psig*

1-B21-LTM-N024B & 25B-2 trip set at 118 in. water*

This revised logic is shown detailed on attached logic detail sheets.

The Recirculation Pumps A & B trip equipment does not change from the implementation of this Plant Mod except as noted above.

To SCRAM the control rods, pilot solenoid valves, (ARI valves), have been added to the CRD pilot air header as shown on attached ARI System P&ID detail. The ARI valves are energized to vent when the proper RPT Logic signal combinations, (see previous), are received. There are three vent paths with two ARI valves in series on each path to eliminate the possibility of a single leaking valve from causing a full or partial SCRAM. Venting the CRD Pilot Air Header, with ARI valves, will open all CRD SCRAM Valves and also close the CRD Discharge Volume Vent & Drain Isolation Valves, initiating a SCRAM. The required vent time, (time from vent initiation to control rod movement), is 15 seconds.

* Pressure signals are "increasing" and level signals are "decreasing".

New control switches 1-C11-CS-5560, 1-C11-CS-5561, and 1-C11-CS-5562 are being added at Control Room RTGB Panel 1-H12-P603. The following is a description of the function of these control switches:

a. Control Switch 1-C11-CS-5560 is a three position switch (INOP, AUTO, and TRIP) equipped with two indicating lights (White and Green).

   The normal position for this switch is in the "AUTO" position with the green indicating light on, which allows automatic trip initiation for both Recirculation Pumps A and B.

In the "INOP" position with the white indicating light on, the ARI System is prevented from operating either manually or automatically.

Placing Control Switch 1-C11-CS-5560 in the "TRIP" position is used in conjunction with Control Switch 1-C11-CS-5561 to manually activate the ARI system.

b. Control Switch 1-C11-CS-5560 is a two position, key operated switch (NORMAL and TRIP) equipped with one red indicating light.

The normal position for this switch is in the "NORMAL" position with the red indicating light off.

Placing this switch in the "TRIP" Position, along with Control Switch 1-C11-CS-5560, will cause the following actions to occur:

1. All ARI solenoid valves energize.
2. Reactor Scrams
3. Annunciator A5-6-4 "ARI INITIATED" alarms.
4. Red indicating light on 1-C11-CS-5561 illuminates.

As long as both switches remain in the "TRIP" position the system can not be reset.

c. Control Switch 1-C11-CS-5562 is a two position switch (NORMAL and RESET) with no indicating lights. This switch is used to reset the system after a 25 second delay with either 1-C11-CS-5560 or 1-C11-CS-5561 returned to their normal position.

When all three of the above control switches in their normal operating position, an automatic initiation signal would cause the following actions to occur:

1. All ARI solenoid valves energize.
2. Reactor Scrams
3. Annunciator A5-6-4 "ARI INITIATED" alarms.

<u>SAFETY SUMMARY:</u> This modification eliminates the possibility of recirculation pump trips due to a single instrument signal and implements a control rod scram in parallel with the pump trips. The probability of occurrence of a spurious recirculation pump trip and that of a trip without the presence of control rods in the core are reduced, thus fuel thermal margins are less likely to be violated.

The consequences of accidents resulting from low water and vessel high pressure are reduced since both control rod scram and recirculation pump trip (pumps A and B) are initiated where only single pump trips were previously initiated.

The malfunction of a single level or pressure instrument could previously have initiated a single pump trip. The new RPT logic eliminated a spurious trip due to a single instrument failure. The failure of any single pressure and/or level instrument (in the new RPT logic) to send its trip signal will not impact the RPT function, since four channels (2 level and 2 pressure) will be installed. Based on the above, the probability of occurrence and the consequences of malfunction of the RPT system is reduced.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

The new RPT logic affects how the recirculation pumps are tripped. The pumps have been previously evaluated for probability of an accident and possibility of malfunction in the FSAR. The ARI valves initiated by the RPT logic to scram the reactor are not safety-related and have no impact on existing RPS valves or other safety-related equipment.

The ATWS RPT actuation logic has been revised to eliminate spurious pump trips due to single pressure or level instrument trips. Based on the above, the margin of safety has remained unchanged as defined in the technical specification bases.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 87-006, Feedwater Flow and Pressure Instrument Relocation

FUNCTIONAL SUMMARY: This modification is necessary to resolve HED discrepancies identified in HED Resolution Package 001. Work scope involves relocating the existing reactor water level indicators B21-R604A and B21-R604B from Panel H12-P603 to Panel H12-P601. Presently, reactor water level indication is not supplied on Panel H12-601 for use while operating ECCS systems. In the space vacated by the 604A and 604B indicators in Panel H12-603, a new Esterline Angus Tigraph Recorder will be installed. The recorder will replace the existing GEMAC narrow range reactor pressure recorder which is located too high in Panel H12-P603 to be read accurately. The new recorder will record narrow range reactor pressure (800-1100 psig) from C32-PT-N008 on Channel B. Channel A will record reactor water level (0-210") from B21-LT-N026B loop. Turbine steam flow, which was recorded on C32-PR-R609, will now only be monitored by an ERFIS Point and not recorded. The space vacated by the old GEMAC recorder in P603 will be enlarged and four new indicators will be installed for Reactor Feed Pump A and B suction flow and discharge pressure. This will group additional feedwater flow and pressure indication with the flow controllers. Completion of this modification will meet CP&L's commitments to the control room design review summary report.

SAFETY SUMMARY: FSAR Figure 7.7.1-5 and Table 7.5.1-1 will be revise to show the new level and pressure parameters. Relocation of indicators and recorders will place the instruments in a more desired visual envelope. No automatic protective systems or plant environmental conditions are affected. The requirements of Regulatory Guide 1.97 Category I will be maintained for reactor water level indicators 604A and 604B. The improved visual envelope will enhance operator's ability to respond to the necessary displayed parameters, therefore more effectively mitigating potential accidents and transients. Technical specification requirements will be complied with for this modification.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 87-031 and 87-032, Remove Recorder and Indicator D12, E41 System

FUNCTIONAL SUMMARY: Plant Operations is not currently using the linear channel radiation monitor recorder in the condenser off-gas radiation monitoring system. For compliance with NUREG-0737, this Human Engineering Deficiency (HED) was identified and assigned a project number of HED-5034.

This recorder (D12-RR-R602) will be removed from the BOP RTG Board. Associated wiring will be removed. The plant cable will be spared. The panel opening will be closed with a blank plate.

Plant Operations is not using the HPCI Turbine Vibration Meter and it's associated recorder. For compliance with NUREG-0737, this Human Engineering Deficiency (HED) was identified and assigned a project number of HED-1133.

This HPCI Turbine Vibration Meter (E41-C002-002) and Turbine Vibration Recorder (E41-C002-003) will be removed from Panel H12-P601. Associated wiring will be removed. Plant cables will be spared. The panel openings will be closed with blank plates.

SAFETY SUMMARY: The devices serve no safety function and removal of them within this plant modification will have no affect on safety equipment. The defined devices have no affect on Chapter 15 accidents. The devices are not required for technical specification application.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 87-164 and 87-165, Group 1 Isolation Level Setpoint Change from LL-2 to LL-3

FUNCTIONAL SUMMARY: The modification to be performed will change the Reactor Vessel Water Level isolation setpoint for the Group 1 Isolation Valves from its existing value of Low Level 2 to a new value of Low Level 3. The reason for change, components affected, and Logic description are found below.

The lowering of the MSIV level setpoint (from LL2 to LL3) is an NRC staff recommendation to meet NUREG-0737, Item II.K.3.16 requirements relating to SRV challenges. It has also been recommended by the GE BWR Owner's Group. Additionally, most BWRs to which this change is applicable have already implemented it.

The following benefits will be realized as a result of the MSIV level setpoint change:

1. Reduction of the probability of MSIV closure (reactor isolation) after plant scrams, maintaining use of the Feedwater system for level control and the main condenser as a heat sink. This would also reduce required time for HPCI and RCIC operation.

2. Reduction of SRV challenges which is beneficial with respect to containment duty, SRV maintenance, and the possibility of relief valves being stuck open.

3. Prevention of unnecessary use of the suppression pool as a heat sink which is particularly important for an ATWS event which might cause excessive suppression pool heatup.

4. Possible increase in the life expectancy of the feedwater sparger as isolation causes a loss of feedwater flow which may require maintaining HPCI operation (i.e, without mixing with feedwater), then the cold HPCI water could produce thermal cycles at the feedwater sparger contributing to fatigue.

The level setpoint for the Main Steam Line Drain Valves (B21-F016 and B21-F019) and the Reactor Water Sample Line Valves (B32-F019 and B32-F020) and is also being lowered from LL2 to LL3. These valves together with the MSIVs form the current Group 1 Primary Containment Isolation Valves. The lowering of the water level setpoint for these valves will maintain the present Group 1 structure and functions. This aids in the elimination of possible Operator confusion and training problems. The level setpoints for these additional valves were also lowered by all of the BWRs implementing the MSIV setpoint change.

Components Affected

Four existing Rosemount 510DU master trip Units (B21-LTM-N024A-1, B21-LTM-N025A-1, B21-LTM-N024B-1, and B21-LTM-N025B-1 in cabinets XU-65, XU-66, XU-67, and XU-68 respectively will be replaced by four new Rosemount 710DU master trip units (new designations B21-LTM-N024A-1-1, B21-LTM,N025A-1-1, B21-LTM,N024B-1-1, B21-LTM-N025B-1-1, which will be relocated from existing slots in XU-65, XU-66, XU-67, and XU-68 to spare slots in card file 2 for the same cabinets. These master trip units will drive existing relays A71B-K1A, A71B-K1B, A71B-K1C, and A71B-K1D (relays are located in cabinets H12-P609 and H12-P611). No new cable or conduit is required for these connections.

A new Rosemount 710DU slave trip unit will be connected to each of the added 710DU master trip units (4 total) and located adjacent to its master trip unit in card file 2 of cabinets XU-65, XU-66, XU-67, and XU-68. The slave trip units shall be designated as follows: B21-LTS-N024A-1-2, B21-LTS-N025A-1-2, B21-LTS-N024B-1-2, B21-LTS-N025B-1-2. The new slave trip units will drive four new GE HFA relays to be located in cabinets H12-P609 and H12-P611 (new designations A71B-K1E, A71B-K1F, A71B-K1G, and A71B-K1H). Four new cables will be pulled in existing conduits to make these connections.

The Reactor Vessel Water Low Level 2 logic from the master trip units shall initiate the following actions:

1. Isolation of the Group 3 isolation valves (G31-F001-RWCU Inboard Isolation Valve and G31-F004-RWCU Outboard Isolation Valve).

2. Isolation of Secondary Containment (Reactor Building HVAC).

3. Initiation of Standby Gas Treatment System.

4. Annunciation of Reactor Vessel, LO LO Water Level.

The Reactor Vessel Water Low Level 3 logic from the slave trip units shall initiate the following actions:

1. Isolation of the Main Steam Line Isolation Valves (Inboard-B21-F022A, B, C & D; Outboard-B21-F028A, B, C & D) and the Main Steam Line Drain Isolations Valves (Inboard-B21-F016; Outboard-B21-F019).

2. Isolation of the Recirculation Sample Isolation Valves (Inboard-B32-F019; Outboard-B32-F020).

The logic for the Low Level 2 master trip units shall be as follows:

1. B21-LTM-N024A-1-1 & B21-LTM-N024B-1-1 (both together)

   or

   B21-LTM-N025A-1-1 & B21-LTM-N025B-1-1 (both together)

   shall initiate Standby Gas Treatment and isolation Reactor Building ventilation.

2. B21-LTM-N024A-1-1 & B21-LTM-N024B-1-1 (both together) shall close the RWCU Inboard Isolation Valve (G31-F001).

3. B21-LTM-N025A-1-1 & B21-LTM-N025B-1-1 (both together) shall close the RWCU Outboard Isolation Valve (G31-F004).

4. B21-LTM-N024A-1-1 & B21-LTM-N025A-1-1 shall actuate the existing "REACTOR VESS LO LO WATER LEVEL SYS A" annunciator (A-06-1-6)

5. B21-LTM-N024B-1-1 & B21-LTM-N025B-1-1 shall actuate the

existing "REACTOR VESS LOW LOW WATER LEVEL SYS B" annunciator (A-06 2-6).

The logic for the Low Level 3 slave trip units shall be as follows:

1. B21-LTS-N024A-1-2 or B21-LTS-N025A-1-2

   AND

   B21-LTS-N024B-1-2 or B21-LTS-N025B-1-2

   shall close the Main Steam Line Isolation Valves (Inboard-B21-F022A, B, C, & D; Outboard-B21-F028A, B, C, & D).

2. B21-LTS-N024A-1-2 & B21-LTS-N024B-1-2 (both together shall close the Main Steam Line Drain Inboard Isolation Valve (B21-F016) and the Recirculation Sample Inboard Isolation Valve (B32-F020).

3. B21-LTS-N025A-1-2 & B21-LTS-N025B-1-2 (both together) shall close the Main Steam Line Drain Outboard Isolation Valve (B21-F019) and the Recirculation Sample Outboard Isolation Valve (B32-F020).

<u>SAFETY SUMMARY:</u> The probability of occurrence of any of the Chapter 15 accidents is not increased by lowering the Group 1 Primary Containment Isolation (PCIS) valve's reactor water level setpoint. The probability of inadvertent//spurious initiation of closure of the MSIVs as analyzed in the FSAR (Main Steam Isolation Valve Closure) will not be affected because the operator actions for initiation will remain the same and the new LL2 automatic trip circuitry will be qualified to the same standards as the existing LL2 automatic trip circuitry which it replaces. The probability of initiation of closure of the MSIVs on a low reactor water level signal will be decreased as the new setpoint of LL3 is below the existing setpoint of LL2 and will therefore not be reached as often due to the number of automatic actions at LL2 which occur to prevent further lowering of reactor water level (e.g., HPCI and RCIC initiation). The extended availability (i.e., below LL2) of the feedwater system to maintain level will also help to prevent further lowering of reactor water level to LL3 and therefore reduces the probability of MSIV closure.

The consequences of any Chapter 15 accidents will not be increased by lowering the Group 1 PCIS valve's reactor water level setpoint. Sections 4 and 5 of NEDC-30601-P (Safety Review of Water Level Setpoint Change for BSEP Units 1 and 2) present this position and required supporting analyses. Although NEDC-30601-P was based on the original FSAR prior to updating in accordance with the guidelines of Reg. Guide 1.70, Rev.3, all current (Updated) FSAR Chapter 15 accidents were considered as part of Chapter 14 of the FSAR prior to updating.

The probability of occurrence or a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. One of the major reasons for lowering the Group 1 PCIS reactor water level setpoint (particularly the MSIV level setpoint) is to reduce challenges of the SRVs and therefore reduce the probability of malfunctions of the SRVs. The reliability and redundancy of the new LL2 Group 1 valve automatic trip circuitry will be equal to the replaced LL2 Group 1 valve automatic trip circuitry as specified in Chapter 7 of the FSAR. The Group 1 PCIS valves, in addition to the SRVs, should be cycled less as a result of the lower water level setpoint, thereby decelerating component failure rates. Radiological releases and equipment damage are evaluated as acceptable by Sections 4 and 5 of NEDC-30601-P.

The consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The single active failure criteria and separation criteria of FSAR Chapter 7 will be maintained for the new LL3 Group 1 automatic trip circuitry.

The possibility of an accident of malfunction of a different type than already evaluated will not be created by the lowering of the Group 1 PCIS valve reactor water level setpoint. Added LL3 Group 1 automatic trip circuitry will meet the current Chapter 7 single active failure criteria and separation. Although the scenarios for some existing Chapter 15 accidents/transients may be changed, the new scenarios have been evaluated as acceptable in NEDC-30601-P.

The margin of safety will not be reduced. The LL3 reactor water level setpoint still ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents, as justified by NEDC-30601-P. No other technical specification bases are affected.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 87-169 and 87-170, Nitrogen Usage Determination

FUNCTIONAL SUMMARY: The Non-Interruptible Instrument Air System is a subsystem of the Instrument and Service Air System which is described in SD-46. It provides instrument-quality air to vital instrumentation and pneumatic controls within the Reactor Containment vessel. Non-Interruptible instrument air is provided to the following control Systems:

a. Main Steam Isolation Valves
b. Safety Relief Valves
c. Reactor Building Closed Cooling Water Flow Control Valves
d. Reactor Recirculation
e. Suppression Chamber Vacuum Breaker Valves.
f. Reactor Head Vents

The purpose of the modification is to upgrade the existing Unit 1 Non-Interruptible Instrument Air (RNA) System as a result of concerns regarding potential sources of oxygen in the inerted drywell atmosphere.

This modification will provide a source of gaseous nitrogen which is dry, oil-free and free of foreign materials to the vital instrumentation and pneumatic controls presently supplied by the Non-Interruptible instrument air subsystem. The gaseous nitrogen for this Pneumatic Nitrogen System will be provided from cryogenic storage tanks located in the yard area Southeast of the Unit 2 Reactor Building. Liquid nitrogen is converted to gaseous nitrogen by two ambient vaporizers, one of which is valved off as a spare. Each vaporizer is sized to supply the total requirement of both Unit 1 and Unit 2 simultaneously. Two parallel pressure-regulator valves are provided downstream of the ambient vaporizers for improved reliability.

A single pipe run is provided from the pressure-regulator valves (interface flange) to the Unit 1 Reactor Building where the line tees at the southeast corner of the building to feed the Division I and Division II headers within the Reactor Building. A second interface flange connection is provided for a future (single) pipe run to the Unit 2 Reactor Building. A remotely-operated crosstie valve, which is normally closed, permits the control room operator to use liquid nitrogen from the Unit 1 or Unit 2 tanks. During normal plant operation one tank is active and the other tank is the backup.

The tie-in points to the Div.I and Div. II Non-Interruptible Instrument Air headers will be tee connections in the piping immediately upstream of the manual isolation valves 1-RNA-V255 and 1-RNA-V256, respectively.

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

a. The existing Non-Interruptible Instrument Air piping (RNA System) will remain in place, but will be valved off by isolation valves upstream of the respective Div. I and Div. II tee connections to the Pneumatic Nitrogen System. During a system outage, these valves will be opened and the Pneumatic Nitrogen System isolate to permit use of the Non-Interruptible Instrument Air while plant personnel are working and entering the drywell.

b. As used during power operation, the Nitrogen Pneumatic System will eliminate the introduction of oxygen into the drywell due to operation of pneumatic controls.

c. An alarm will be added in the control room to signal low cryogenic tank pressure. Local and control room indication will be provided for tank level and pressure.

d. Control room and local indication of vaporizer outlet temperature will be provided. Vaporizer outlet temperature is alarmed for a low setpoint.

e. The vaporizer will be automatically isolated on low vaporizer outlet temperature. Two temperature shutoff devices mounted in series will provide additional reliability.

f. Pressure regulating valves operating downstream of the ambient vaporizers are self regulating based on downstream pressure.

g. A unit crosstie line will be provided between the Unit 1 and Unit 2 tanks immediately downstream of the ambient vaporizers but upstream of the pressure-reducing valves. This will provide a nitrogen source backup for the two systems. A unit crosstie valve, provided in this line, is operator controlled by a switch on each RTGB. Either the Unit 1 or Unit 2 operator can open the crosstie valve. Both manual switches must be in the closed position in order for the crosstie valve to be closed.

h. Existing indicators and annunciator alarm windows located on the control room RTG Board will be relocated to other panels in order to achieve functional grouping of the new PNS System devices. This is being accomplished in compliance with human factors engineering criteria.

<u>SAFETY SUMMARY:</u> No new anticipated operational occurrences or postulated accidents will be introduced as a result of the work contained in this modification. The pneumatic nitrogen system, added to the non-interruptible instrument air system, requires no external source of power except for instrumentation and control devices. The pneumatic nitrogen system adds no new function to the existing system, nor does it reduce the present function of that system.

Since the pneumatic nitrogen system adds no new function to the non-interruptible instrument air system, no additional consequences of an accident than those previously evaluated in the FSAR could be introduced. The PNS system will remove the source of oxygen in the drywell and remove the chance of a combustible atmosphere in the drywell during post accident conditions.

The pneumatic system, while providing the pneumatic requirements of the instruments and controls in the drywell, will be isolated in the case of a core spray LOCA or loss of power. In such an event, the nitrogen backup system aligns with the ADS headers and the CAC valves for the reactor building to torus vacuum breakers. Since both the non-interruptible instrument air system and the PNS are isolated by the same valve, the possibility of occurrence of a malfunction of the nitrogen backup system will not be increased. The signal interlock between the non-safety related PNS system and the safety related nitrogen backup system uses a Q-list, environmentally qualified pressure switch. This is a direct acting switch with no external power source.

Because of the isolation of the pneumatic nitrogen system during a LOCA or loss of power, the consequences of a malfunction in the nitrogen backup system will not be increased due to the installation of the PNS system.

Replacing that portion of the RNA system which provides the pneumatic requirements, for the instruments and controls within the drywell, with a nitrogen system tied in upstream of the containment isolation valves and adding containment isolation check valves to the RNA system to meet the requirements of GDC 56 will create no new conditions which could lead to an accident or a malfunction of equipment important to safety of a different type than already evaluated in the FSAR.

Because the basic system design and operational capabilities remain unchanged, the margin of safety defined in the bases for the Technical Specifications is not reduced. The nitrogen backup system is designed with enough capacity to serve the essential loads for twenty four hours which is evaluated by calculation No. 84-196-233 in plant modification 84-196 and verified that adequate margins exist.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 87-196, Background Reduction Structure

FUNCTIONAL SUMMARY: New "walk-in" radiation monitors will be installed at the three power egress areas of the Reactor Building to replace the current temporary monitoring stations and to update the Radiation Detection System used in monitoring personnel as they exit the areas. A new permanent enclosure at each of the three existing stations to be removed will be built to provide shielding from background radiation for the new, highly sensitive radiation monitors.

Lean-to enclosures will be built outside both egress areas of the breezeway, on the north side of Unit 1 and the south side of Unit 2 Reactor buildings to house new monitors. Each of these enclosures will have an 8" concrete slab on grade, 75% solid concrete masonry block walls, and concrete on metal pan roof deck supported by steel beams. Doors and frames will be hollow metal. Interior block walls and concrete roof will provide additional shielding for the walk-in monitors. The enclosures will have fluorescent lighting and split system heating and cooling. Enclosure power will be supplied from 208V/120V, 3 phase, 4 wire Lighting Panel BRS1 coming off existing Transformer F10.

These new permanent enclosures with upgraded monitoring units will improve the current Radiation Monitoring and Contamination Program and also addresses the December 1987 INPO Evaluation RP.9-1, Radiation Control.

The enclosures will require core bores through Turbine Building walls for power feed conduit and saw kerfs for enclosure roof counter flashing, and anchor holes for roof beam wedge anchors in Reactor Building walls and masonry wall reinforcing embedment to existing Radwaste loading dock slab which have been evaluated and show no impact on the safety of the structures.

An additional radiation monitor unit will be added in existing location (to be verified later by E&RC) which will include a gas manifold at the exterior of location for personnel decon.

SAFETY SUMMARY: The three background radiation structures with monitors will upgrade and replace current temporary frisking stations and does not increase probability of occurrence of any accident previously evaluated in the FSAR. The modification does not reduce the capability of any plant system or shutdown operations.

This modification does not reduce the capability of any system associated with accidents evaluated in the FSAR, therefore it does not increase any consequences.

This modification does not affect any equipment important to safety evaluated in the FSAR, therefore there is no increase in probability of malfunction.

This modification does not affect any equipment important to safety evaluated in the FSAR, therefore there is no increase in probability of malfunction.

This modification does not affect plant equipment of systems associated with safety, therefore there is no probability of an accident or possibility for malfunction of equipment important to safety of a different type.

This modification does not affect any Technical Specification.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 87-203 and 87-204, Upgrade Turbine Building Sample System

FUNCT ONAL SUMMARY: The Turbine Building Sample System Upgrade shall provide the following:

1. Add Series Isolation Valves to Sample TS-1 and TS-2 and replace the existing sample point isolation valves, which shall allow a positive means of sample isolation during component calibration or maintenance.

2. Provide flow indicators in the cooling water headers of the secondary sample coolers and chiller condenser.

3. Upgrade the existing chiller units capacity control temperature controller, to provide better temperature control and to permit ease of temperature controller calibration.

4. Provide pressure indicators downstream of the secondary coolers of sample points TS-1, TS-2, TS-3 and TS-6.

5. Add flow indicators in the sample stream bypass lines of sample points TS-1, TS-2, TS-3 and TS-6.

6. Modify the TS-1, Sample stream to incorporate a dissolved hydrogen analyzer for hydrogen water chemistry analysis. This sample stream shall use a spare secondary cooler previously designated as TS-7.

7. Upgrade the existing conductivity samples to provide conductivity analyzers which automatically temperature compensate the analyzed sample.

8. Reroute (4) four grab sample lines from the inlet side to the outlet of the secondary coolers. This will allow Chemistry personnel to obtain a 25 degree C grab sample.

9. Add series grab sample isolation valves downstream of the existing valves. This will allow E&RC personnel to set the grab sample flow rate with the existing valves and provide isolation using the new valves.

10. Modify the dry and wet section panels as required to support installation and E&RC Chemistry requirements, as well as Maintenance.

11. Remove and replace existing recorders TS-CR-863 and TS-AR-868 with a combined recorder/data logger.

12. Remove and replace existing temperature indicators TI-1328, -1329, -1330, -1331 and -1420 with RTD temperature elements, the output of which shall interface with the new recorder /data logger. This shall provide a means to trend the sample stream temperatures and determine if temperature transients within a sample stream caused a false alarm to be generated.

13. Reroute the GE GAM RAD sampler, B21-2001. Drain from the open sink in the sample station to the hood section of the sample panel. This will satisfy an ALARA concern.

14. Modify the wet section panel to provide calibration connection points, cooling water connections and drain/fill points for calibration of the dissolved gas calibrator.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

15. Replace existing sample system components on a one for one basis, ie., sample coolers pressure control devices, etc., to provide sample station life extension.

SAFETY SUMMARY: Upgrading existing equipment and adding components to the Turbine Building Sample Station will no increase the probability of occurrence of any accident previously evaluated. The consequences of any accident previously evaluated will not be increased due to the addition, relocation, or upgrade of equipment within the TBSS. The modification is an upgrade of a non-safety related system, which will enhance the system's ability to monitor plant parameters. The ability to adequately monitor plant chemistry parameters will further increase the reliability of plant by ensuring proper actions will be more accurately assessed.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 87-259 and 87-260, Unit 1 and Unit 2 E41-F006 Relocation

FUNCTIONAL SUMMARY: The High Pressure Coolant Injection (HPCI) system is an Engineered Safeguards System and is part of the Emergency Core Cooling System (ECCS). The HPCI system shall start and deliver design flow of 4250 gpm within 20 seconds upon receipt of an initiation signal.

The HPCI system is designed to:

a. Provide adequate core cooling to prevent overheating of the reactor fuel (maximum 2200 degrees F fuel cladding temperature) in the event of a Loss of Coolant Accident (LOCA) which does not result in rapid depressurization of the pressure vessel (line breaks up to 5 inches in diameter).

b. Provide plant shutdown support by maintaining sufficient reactor water inventory until the reactor is depressurized to allow operation of the Low Pressure Coolant Injection (LPCI) or Core Spray (CS) systems.

c. Provide redundancy to the Reactor Core Isolation Cooling (RCIC) system by being capable of fulfilling the RCIC objectives in the event RCIC becomes inoperable.

d. Provide automatic or start-up operation independent of auxiliary AC power, service air, or external cooling water systems.

e. Provide automatic HPCI system isolation in the event of a High Energy Line Break (HELB) through Group 4 isolation or equivalent (check valves).

BASIC FUNCTION OF PM 87-259

Modifications will consist of minor piping changes and will not affect the HPCI system design basis as described in this section. Examples of the type of changes are as follows:

Change	Reason
Relocate E41-F006	Correct valve orientation. Place in less severe ambient and accident environment. Place in lower general radiation area (man-rem eduction for maintenance). Allow for manual operation of valve.
Install 14" Check Valve	Immediate isolation of HPCI return line should an HELB occur between the MSIV pit and E41-F006.
Install Vent and Drain Lines/Valves	Vent/Drain/fill HPCI return line. Conduct required LLRTs.
Cut and Cap E41-F006 Gland Leak-off and	Mechanical Maintenance request. BSEP is installing live-loaded

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

Eliminate E41-V65	packing where possible and eliminating the need for gland leak-off lines.

Note: E41-F006 is not part of Group 4 isolation. Final isolation of the HPCI return line in the event of an HELB will require shutting B21-F032A and placing the unit in Operating Condition Mode 4.

The HPCI return line is classified as Q-List, Seismic Category I, Class 1-E equipment.

SAFETY SUMMARY: The modification will not increase the probability of occurrence of any accident previously evaluated in the FSAR (Chapter 15). HPCI return line failures are not related in any way to the initiation of any accident in Chapter 15 of the FSAR. The system is one of the initiators of an inadvertent pump start transient; however, this change will not increase the probability of this event occurring.

The HPCI system is necessary to mitigate the consequences of accidents evaluated in Chapter 15 of the FSAR. This modification has been shown to have a negligible effect on the hydraulic capabilities of the HPCI system, and will not prevent the HPCI pump from delivering rated flow against design pressure.

No increase will result from this change in the probability of occurrence of malfunction of equipment important to safety previously evaluated in the FSAR, as it has been shown that the probability of HPCI return line failure has not been increased due to the proposed modifications.

The modification will not increase the consequences of malfunction of equipment important to safety previously evaluated in the FSAR. The modification does not affect backup systems to the HPCI (ADS/LPCI or Core Spray), thereby not affecting their ability to mitigate an accident.

The modification will not create the possibility of an accident or malfunction of equipment previously evaluated in the FSAR. No new material, equipment or mode of failure not previously evaluated in the FSAR will be introduced by this modification.

The margin of safety as defined in the basis of any technical specification will not be reduced. The HPCI pump will not be prevented from delivering rated flow against design pressure. Valve E41-F006 remains the containment isolation boundary. No containment isolation valves required to meet the Technical Specification surveillance requirements will be added or deleted.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 88-008 and 88-036, Unit 1 and Unit 2, Replace Reactor Water Cleanup Class-2 Piping

FUNCTIONAL SUMMARY: The purpose of the plant modification is to replace the Reactor Water Cleanup System Group IIA piping, either replace or refurbish the associated system components (valves, supports, etc.) as required, and perform the acceptance testing.

The following list outlines major components affected:

1. The portion of the RWCU system to be affected is from the Regenerative Heat Exchanger 1C (Line 1/2-G31-50-4-907) to Valve 1/2-G31-F042.

2. Relocation of Flow Orifice FE-N040. This will result in a reroute of instrument tubing and relocation of Valves 1/2-G31-V15 and 1/2-G31-V16.

3. Replacement of Valves 1/2-G31-V15 and 1/2-G31-V16.

4. Temporary removal/installation of Temperature Element TE-N015.

5. Replacement of link seal (sleeve #638) on Line 1/2-G31-50-4-907.

6. Temporary removal/reinstallation of supports and restraints (as required) on Line 1/2-G31-50-4-907.

7. Modification of Anchor PS-6192. PS-6192 is the only Seismic Category I component in this plant modification to be modified.

The RWCU Group IIA piping is being replaced due to the original piping being damaged by Intergranular Stress Corrosion Cracking. The replacement piping material will be upgraded to Type 316L stainless steel with special chemistry requirements which is less susceptible to Intergranular Stress Corrosion Cracking.

The function or operation is not affected for the RWCU system and the components covered by this plant modification.

SAFETY SUMMARY: The modification upgrades the Class II system piping and removes a pipe support justified by an engineering analysis. The Class I portion of the system necessary to isolate for an accident will not be affected by this change. The new piping material is less susceptible to intergranular stress corrosion and cracking, and is both structurally and metallurgically compatible. No reduction in the margin of safety for either the transient analyses or system components is made by this change.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 88-014 and 88-015, Unit 1 and Unit 2 Direct Current Motor Surge Suppression

FUNCTIONAL SUMMARY: Due to the valve problems that have occurred and the history of motor (DC) failures at BSEP, valve, actuator, and motor vendors were contacted concerning these problems. These discussions revealed that changes are required in the BSEP safety-related motor-operated valve logic. These changes will improve valve and motor life and reliability. Also, a motor-operated valve task group was formed to review/evaluate issues associated with safety-related motor-operated valves. The changes listed below are a result of vendor recommendations as well as valve task group evaluations on specific motor-operated valves.

1. DC safety-related valve motors currently have shunt fields that are continuously energized. The logic requires revision such the shunt fields are deenergized when the motor is not operating.

2. DC safety-related valve motors currently have heaters inside the motor casing that operate when the motor is deenergized. The purpose of the heater was to minimize condensation damage for motors stored for long periods, installed outside, installed in underground pits, or installed in nonventilated enclosed areas. Therefore, the heaters should be disconnected and spared.

3. DC safety-related valve motors currently have heaters in the limit switch compartment that are continuously energized. The environmental qualification of the actuator is not dependent on these heaters, but can actually be degraded by wiring resting against the hot heaters. Therefore, the heaters should be disconnected and spared.

4. DC safety-related valve motors currently have reduced voltage starting resistors installed. These resistors limit the current available to the motor and can result in motor stall under differential pressure conditions. Therefore, the starting resistors should be removed from the circuit.

5. DC safety-related valve motors currently have no device to suppress the voltage surge when the shunt field is deenergized. This could damage the motor windings, thus affecting motor operation. Therefore, a surge suppression device (resistor) should be installed in the circuit.

6. DC Safety-related valve motors currently have no mechanism that would deenergize the control circuit during a motor overload condition. This condition could damage the motor, valve, or actuator. Therefore, the logic requires modification that would preclude this from taking place.

7. In addition, the MOV task group recommended changing the overload heater sizes, replacing a breaker, and replacing one power feed cable.

Based on the above, PCN 06152A has been initiated/revised to modify the DC safety-related valve motors to help increase:

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

1. The life of the motors by reducing the internal temperature the motor.

2. The life of the motor by reducing the potential for stall currents in the motor.

3. The reliability of the valve by increasing the reliability of the motor.

This Plant Modification will modify the MCC compartments listed below as follows:

1. Determinate an spare motor heater power cables.

2. Determinate and remove starting resistor(s).

3. Determinate and remove 1A and 2A relays/resistors as applicable.

4. Add jumper(s) in place of starting resistor(s).

5. Add resistor in parallel with shunt field.

6. Rewire spare contacts of 1F and 1R (convert to NO as required) in power leg of shunt field to allow shunt to be energized only when the motor is in operation. Compartment B26 (Node L1F) does not have any spare contacts; therefore, two relays will be added. Compartments B18 and B48 only have one spare contact; the 1R contact was used as an anti-pumping function; but on further investigation, due to the gear ratio, this feature is not necessary, so the 1R contact will be used in the shunt field and the No. 8 limit switch contact will be jumpered (see supporting documentation in Section C).

7. Rewire the Spare 74 contact to deenergize the control circuit while maintaining position indication on an overload condition and replace the overload heaters.

8. Replace the breaker on Compartment B15, and replace the power cable for Compartment B20.

9. As a result of modifying the shunt field to be energized only when the motor is operating, a wire change in transfer contactor cabinet (Node L6C) is required to energize the normal or alternate contactors when the respective breaker is closed.

<u>MCC/COMPARTMENT</u>

1XDA/B14	E41-F004
1XDA/B15	E41-F059
1XDA/B16*	E41-F007
1XDA/B17	E41-F006
1XDA/B18	E41-F008
1XDA/B19	E41-F011
1XDA/B20**	E41-F003
1XDA/B21	E41-F001
1XDA/B22	E41-F042
1XDA/B23	E41-F041

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

1XDA/B24	E41-F012
1XDA/B26***	E11-F008 (alternate feed)
1XDB/B50	E11-F008 (normal feed)
1XDB/B48	C31-F[illegible]
1XDB/B51	[illegible]-F040
1XDB/B52	[illegible]-F019
1XDB/B37	[illegible]51-V8
1XDB/B38	E51-F010
1XDB/B39	E51-F046
1XDB/B40	E51-F012
1XDB/B41	E51-F013
1XDB/B42	E51-F022
1XDB/B43	E51-F008
1XDB/B44	E51-F045
1XDB/B45	E51-F031
1XDB/B46	E51-F029
1XDB/B47	E51-F019

* Replacing breaker.

** Replacing power cable.

*** Relays (2) installed to energize shunt field only when motor is operating.

SAFETY SUMMARY: Although the changes made by this modification are very detailed, the basis function and operation of the individual valves, as well as the associated system logic, have not been changed. Therefore, the probability of occurrence and the consequences of an accident previously evaluated in the FSAR will not increase. The life and reliability of the motor operated valves will be increased, therefore the probability of occurrence and consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not increase. The changes made by this modification ensure that affected systems required to operate under accident conditions will perform as required. The margin of safety will be maintained by the valve design.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 89-003 and 89-004, Reactor Building Ventilation Isolation Damper Upgrade

FUNCTIONAL SUMMARY:

History

The original specification for the BSEP safety-related secondary containment ventilation isolation dampers was based on fail-closed damper actuator solenoids. The actual damper actuator solenoids installed were fail as-is. Another original design discrepancy was that, even though there were two divisions of uninterruptible air supplies, only one of these divisionalized supplies was used to supply both divisions of secondary containment isolation dampers. This meant that a single active failure of that division (such as the operator closing a single valve) would render all four fail as-is secondary containment isolation dampers inoperable. The original source of long term post-accident instrument air credited for the secondary containment isolation dampers was a manual hookup to the Containment Atmosphere Dilution System (CAD).

The two original design discrepancies were not rectified when the safety-related standby air compressors were added to the Reactor Building Non-interruptible Instrument Air (RNA) system, nor were they detected when the CAD hookup was deleted from the plant design. The impact of the original design was not realized during a subsequent review of plant safety-related equipment classifications, nor in the subsequent down rate of the stand-by instrument air compressors to non-safety. This down rate of the stand-by air compressors removed the only remaining credible source of long term instrument air to the secondary containment isolation dampers.

As a result of events leading to LER 88-034, the consequences of the non-divisionalized, fail as-is design with no long term reliable instrument air supply were realized and determined to be unreviewed. The air to close the dampers could still be credited but there would be no credible air supply to hold them closed for the duration of the accident.

PRESENT SITUATION

The four secondary containment ventilation isolation dampers are 72" butterfly valves. They have two requirement in performing their safety function. 1) is that they close in four seconds and 2) is that they remain closed for the duration of the accident, which is 30 days in the LOCA analysis. The system must be designed against single active failure.

At present the valves fail as-is on loss of air. Both Divisions of valve trains are supplied from a single division of non-safety instrument air. This means they could not be credited with remaining closed for the duration of an accident which resulted in loss of instrument air.

OPERATION

This design change will add a latching mechanism to each valve to keep the valves closed for 30 days post-accident. It will also place the Division II valves on Div II RNA instead of the Division I RNA which exists in the present design. There will be no change to either the actuators which close the valves, the electric/air solenoid valves which control the position of the actuators, or the control circuit logic which operates the electric/air solenoid valves. The accumulator air circuit will be modified by the addition of the latch actuator and a pair of manual vent valves will be added for periodic testing purposes. Viewing ports will be added to the spool piece between each divisionalized pair of dampers to be used to view the mechanical latches during operations and Periodic Tests. After this modification, the existing rad monitor, damper control circuitry, electric/air solenoid valve, and actuator design will still not be capable of closing the isolation valves without electric power. The final design will not result in automatic secondary containment isolation on loss of power or loss of air pressure, i.e., it will not be fail-safe. There will continue to be automatic secondary containment isolation for those LOCA and high radiation signals which result in the secondary containment isolation signal (SCIS) in the existing design.

The latch added to each valve by this design change will be released by its latch actuator (a small spring-closed/air-to-open valve operator) when sufficient air pressure is present in the accumulator portion of the air control circuit. This release function is not a safety function. The safety function of the latch actuator is performed by the return spring of the actuator. The spring will move the latch to the safe position (latched) upon loss of air pressure in the accumulator portion of the air control circuit. If the latch precedes the valve to its safe (latched) position as the valve moves to its closed position, the design is such that the valve can force the latch out of the way to finish its stroke and the latch will drop back to the latched position.

The latch is analyzed to maintain its latched position during and after experiencing the OBE or SSE, that is, it will hold the valve in its closed position after either of these events should the valve be required to close and remain closed concurrent with loss of RNA.

IMPACT ON OPERATIONS

There will be no Technical Specification changes as a result of this modification.

There will be a change to the annunciator procedure for low RNA header pressure. The fact that the safety-related accumulators are capable of closing the valves in four seconds is based on the assumption that they will be supplied with non-safety RNA up until the time that the accident or initiating event resulting in the SCIS occurs. If RNA is lost for a period of time prior to the SCIS, credit for the four second closing time could only be taken if the SCIS were generated prior to the depletion of the air pressure in the accumulators. The change to the annunciator procedure will require the operator to manually initiate the SCIS as an action on annunciation for low RNA header pressure within the minimum time period credited for accumulator leakdown and maintain secondary containment until the low pressure indication is dispositioned.

There will be a recommended change to the operating procedure for the reactor building ventilation system such that after any occurrence where air pressure to the latch actuator(s) has been lost while the isolation dampers were closed, the latches for the dampers must be physically verified to be in the unlatched position after air pressure is restored before opening the dampers.

There will be a recommended change to the Periodic Test for the isolation dampers to test the latch function and to verify the time period which can be credited for operator action between loss of RNA header pressure and the point where the accumulator system pressure drops below the minimum allowable. The PT change will also recommend testing each division of isolation dampers separately to verify the single failure criteria.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

SAFETY SUMMARY: The analyzed function of the isolation dampers is to close and remain closed, thereby mitigating the consequences of an accident, specifically the reduction of dose at the site boundary and control room. Their failure to close or to open for any reason during normal operations would not contribute to the probability of occurrence of any analyzed accident. No RPS or safe shutdown systems are affected by these dampers. Spurious closing during operation would not be unsafe, although this condition would soon lead to a plant trip (high MSIV pit temperature) which could challenge safety systems. Failure of more than one of either pair of dampers to close, and the effect of their failure to close during an accident requiring their closing is not analyzed and would not change the probability of occurrence of any analyzed accident.

The ventilation dampers are part of the ESF for the mitigation of the consequences of certain accidents, in that they limit inleakage into the reactor building during operation of the SBGT system, thus allowing the SBGT system to maintain the reactor building at a sub-atmospheric pressure of -0.25 inches WC. The mitigation is complete when they perform their safety functions. The consequences of these certain accidents would be increased if the dampers did not perform their safety functions; however, the modification does not reduce the effectiveness of the dampers in mitigating these accidents.

No changes being made to the dampers by this modification change the failure probability, either open or closed. The probability that the dampers will lose station instrument air following an accident increases due to downrating of the instrument air system. The damper latch, however, more than offsets this increased probability by providing a means for performing the same safety function without requiring instrument air. The latch performs the same safety function formerly ensured by the uninterruptible instrument air. The new latch conforms to approved design basis, and is built and maintained to the appropriate quality classification, tested initially to demonstrate satisfactory operation, and exposed to a surveillance program which will periodically verify its satisfactory operation. Downrating the instrument air system does not affect the requirement for closure in four seconds, because the safety-related accumulator on each damper air circuit will be fully charged pre-accident, and would be capable of providing enough air to accomplish the four second closure requirement.

The latch design and testing requirements ensure no new malfunctions or accidents will be created by this change. The change from Division I air supply to Division II supply will prevent a single inadvertent or adverse operator action from threatening the safety function of four second closure.

A reduction in margin to be considered would be an increased offsite dose if more than one division of dampers fail to close when required. One division is analyzed to fail to close as a part of the single active failure analysis. Therefore, no reduction in the margin of safety is created by this mod.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 89-005 and 89-006, Standby Gas Filter Inlet and Outlet Valve Positive Light

FUNCTIONAL SUMMARY: This Plant Modification will modify the existing standby gas filters inlet and outlet valve position lights to give positive valve open and closed indication. The existing valve indication is such that if one valve is open more than 96%, and the other valve is open more than 96%, and the other valve is open more than 4%, the indication is that both valves are open. This plant modification will modify the valves open and closed indication such that, if both valves are greater than 96% open, you will have a valve open indication (red light); if both valves are less 4% open you will have a closed indication (green light). This revised wiring configuration will have no intermediate valve position indication, but will ensure positive indication of both valves opening and closing.

SAFETY SUMMARY: The modifications to the SBGT inlet and outlet valve position lights will further ensure the valves are in their proper position to pe form their intended functions. This changes enhances the reliability of the SBGT system by providing increased margin of safety relative to the valve position. The SBGT system is not an initiator of any accidents, but provides a mitigating function for potential releases. By increasing the reliability of the system, the probability of equipment malfunction is decreased, and the mitigating function of the SBGT system is further ensured; therefore, no increase in consequences will result from this change.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PM 89-023, 1-E51-F007 Alternating Direct Current Power Supply

FUNCTIONAL SUMMARY: Presently, 1-E51-F007, the inboard steam supply isolation valve for the RCIC turbine driven pumps, is electrically powered from Division I MCC 1XC, Comp. DS4 and Division II MCC 1XD, Comp. DY1 through an automatic transfer switch. Normally, the valve is powered from MCC 1XC but upon loss of this supply the transfer switch will enable the valve to be supplied from the alternate source once an operator closes the breaker in MCC 1XD, Comp. DY1 and operated the key lock valve control switch on the from of Comp. DY1. The alternate feed from MCC 1XD to Valve 1-E51-F007 was implemented per P.M. 84-055 to comply with CP&L's commitments to the NRC concerning the requirements of 10CFR, Part 50, Appendix R, Section III G.

Recently, it was determined that a control room fire would not only cause loss of automatic control of the diesel generators but could also cause valve 1-E51-F007 to close spuriously prior to an operator switching control of it from the control room to local control at MCC 1XC. Once Valve 1-E51-F007 closes spuriously, there is no way to reopen it immediately since offsite power is assumed to be lost (see ASCA Report, Book I, Page 3-9) and the diesel generators can not be manually started in the specified time line. During that time period, since steam has been cutoff to the RCIC Turbine Driven Pumps, the reactor water level will decrease due to the loss of makeup water from the RCIC system.

Per enclosure 1 of NRC Generic Letter 81-12, either HPCI or RCIC is recommended for providing the reactor coolant makeup function so as to achieve hot shutdown in a BWR nuclear plant. At BSEP HPCI is connected to Train 'A' while RCIC is connected to Train 'B' (see ASCA Report, Book I, Page 3-20). RCIC or HPCI can be manually be controlled from either the Control Room or a local station.

For a Control Room Fire, HPCI is assumed to be lost. Only RCIC is assumed to be available to provide the Reactor Coolant makeup function while the reactor is brought to either hot shutdown or cold shutdown. For a Control Room fire, RCIC must be available to prevent uncovering the reactor core. MOV 1-E51-F007 must be open so steam is supplied to RCIC turbine, which in turn drives the RCIC Pumps.

To prevent the loss of RCIC due to the postulated spurious closure of Valve 1-E51-F007 during an Appendix R event, an AC independent feed will be used as the most efficient means of reestablishing RCIC flow to the RPV.

To provide an AC independent alternate feed to valve 1-E51-F007, this modification will install a DC feed from the Unit 1 station batteries through an invertor to power the AC valve. The DC feed will be obtained by installing a new breaker into compartment B49 of MCC 1XDB.

The changes introduced per this modification are required to maintain CP&L's commitments to the NRC concerning 10CFR50, Appendix R, Section IIIG, as previously stated.

To obtain the power supply for the F007 valve, an existing starter for a disabled valve (1-E11-F023) will be removed, permanently disabling the MOV for this valve. Unit 1 head spray has been disabled in accordance with EER 86-0275. MOV 1-E11-F022 will remain under clearance, with MOV 1-E11-F023 disabled electrically. Both of these MOVs were under clearance prior to this modification in accordance with EER 86-0275. This modification further disables a system that was deemed not required for the safe operation of Unit 1 per EER 86-0275.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

SAFETY SUMMARY: EER 86-0275 addresses the disabling of the RHR Head Spray system. The safety summary of EER 86-0275 (RHR head spray) is not impacted by this change. The safety summary of the new ASSD feed to the 1-E51-F007 is further addressed below.

The alternate feed to the F007 valve is required only to operate during an Appendix R fire, and no credit is taken for the valve in either preventing or mitigating the consequences of a Chapter 15 accident. The normal feed to the F007 valve is isolated from the alternate feed at the qualified automatic transfer switch. The normal feed will not be affected by the new alternate feed in its ability to perform so as to prevent and mitigate the accidents addressed in chapter 15 of the FSAR.

The installation of the new breaker into MCC 1XDB will not affect the safe operation of the MCC. Calculation BNP-E-6.002, Rev. 0, demonstrates that buses upstream will not be affected because of the coordination between protective devices. Re-supplying the alternate feed from a battery backed source improves its reliability, but does not change its function. The alternate feed is a non-Q circuit required only to operate for an Appendix R fire, and has no affect on any safety system other than the Unit 1 Battery 1B-1 and the specific DC buses. Any fault in the alternate feed will be isolated from the MCC by the new DC breaker. The Appendix R BSEP DC Load Study was reviewed and it was determined that adequate margin exists in the Unit 1 battery for operation of the alternate feed.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 89-034 and 89-045, Appendix R Safety Relief Valve Dedicated Air Supply

FUNCTIONAL SUMMARY: As a result of recent outage constraints on the ability to complete installation of plant modification No. 86-007 (Inadequate Core Cooling Detectors), a study was conducted to determine if an alternate method of safe shut down was available. A study group composed of Operations, Tech Support, and BESU studied alternate methods of shut-down including scenarios involving the Automatic Depressurization System. The study group determined that the Automatic Depressurization System safety relief valve(s) (ADS SRV) could provide Alternate Safe Shutdown; however, frequent or continuous actuation of the ADS SRVs would be required.

The existing ADS SRV accumulators were analyzed to determine if they were large enough to accommodate the frequent or continuous actuation as outlined above. It was determined, by the study group, that the accumulators were not large enough to insure Alternate Safe Shut Down (as outlined above) without a reliable pneumatic supply. The ADS SRV accumulators were originally designed to provide sufficient capacity to cycle the valves open three to five times at design pressure (Ref. NUREG 0737, II.K.3.28). These accumulators were also large enough to meet the original Appendix "R" requirements to insure Alternate Safe Shutdown (ASSD).

The capacity of the accumulators raises a new Appendix "R" concern with regard to the BSEP Alternate Shutdown Capability Assessment (ASCA) which this plant modification is written to resolve. This modification will take credit for portions of the existing RNA system including Nitrogen Back-up and Pneumatic Nitrogen to provide a reliable pneumatic supply to the ADS SRVs. This reliable pneumatic supply will be used as the basis for the alternate safe shutdown capability is lieu of the accumulators use in the original Appendix "R" calculations.

A review of the Reactor Building pneumatic supply systems determined that the Div. I Nitrogen Back-up System and the Div. II Pneumatic Nitrogen System could be protected from an Appendix "R" standpoint to provide reliable operation of the SRV's during various fire scenarios.

Because the systems referenced above were not originally analyzed from an Appendix "R" standpoint, the use of these systems to provide reliable ADS SRV operation during the following fire scenarios will require some modification of equipment.

The purpose of this plant modification is to resolve 10CFR50 Appendix "R" (Alternate Safe Shutdown) concerns. This modification addresses three specific fire scenarios. These scenarios are based on the fact that the systems chosen above which are required for the operation of the ADS SRV's, are located in three different fire zones. Each of the fire scenarios for a particular zone is outlined below.

1. The first scenario addresses a postulated fire in the "south' zone of the Reactor Building. A fire in this zone could disable the Div. II N2 Back-up System because of components which are located in the "south" fire zone but would not affect the Div. I N2 Back-up System which does not have any components located in the "south" fire zone. In this scenario the Div. I N2 Back-up System would be available to provide a reliable pneumatic supply to the ADS SRVs. The components of the Div. I N2 Back-up System are not presently classified as Appendix "R" essential safe-shutdown components. The Div. I N2 Back-up System

will be evaluated and added to the ASSD procedure per the requirements of the Appendix "R" program.

2. The second scenario addresses a postulated fire in the "north" zone of the Reactor Building. A fire in this zone could inop both divisions of Nitrogen back-up. From an Appendix "R" standpoint the use of Pneumatic Nitrogen (Div. II RNA/PNS) as a back-up system would be acceptable in this scenario if the following equipment and procedural modifications are made. The cable providing the power feed for valve 1-RNA-SV-5261 (Div. II RNA/PNS) is presently routed through a portion of the "North" fire zone. This cable will be rerouted out of the "north" fire zone. In addition the components of the Div. II RNA/PNS will be added to procedure ASSD-0 as Appendix "R" essential safe-shutdown components. The circuit for valve No. 1-RNA-SV-5261 (CKT 25) is not presently listed as a vital ASSD load and will be added to ASSD-01 as part of this modification.

3. The third scenario addresses a postulated fire in the Control Building. A fire in the Control Building could damage the cables providing power to both divisions of N2 Back-up as well as both divisions of RNA/PNS. In this scenario the possibility of a "hot short" (power cables being energized by other circuitry as a result of the control room fire) exists. Valve No. 1-RNA-SV-5482 (Div. I N2 Backup Sys.) is a normally closed - energize to close - fail open valve. A "hot short" could possibly prevent this valve from failing to its open position. Valve No. 1-RNA-SV-5253 (Div. I N2 Backup Sys.) is a normally open - energize to close - fail open valve. A "hot short" or spurious power could cause this valve to close. This would prevent N2 Back-up from supplying N2 to the SRV's. This modification will provide for the local isolation (de-energizing) of these valves via manual key-lock isolation switches mounted in a convenient location in the vicinity of the valves. The use of these manual key-lock isolation switches will be added to the ASSD procedures per the requirement of the Appendix R program.

The modifications outlined above will insure that the ADS SRVs will function to provide safe shut-down of the reactor within the guidelines of the BSEP ASCA. The existing system design criteria for the systems affected will be utilized in the changes implemented by this plant modification.

These modifications will make use of the existing emergency lighting equipment installed in the Reactor building. The new isolation switches will be mounted in a box directly to one side of an existing pressure gauge showing N2 system pressure which is specifically illuminated by an emergency light. The proposed location of the switches should receive adequate illumination levels and will eliminate the need to install additional emergency lighting equipment.

SAFETY SUMMARY: The analyzed function of the RNA/PNS system is to provide air/nitrogen to all vital reactor instrumentation and air operated pilot valves during normal operations. This system is divisionalized to provide proof against a single active failure from making the supply for those functions inoperable. As part of the mod, the cable which supplies control power to valve RNA-SV-5261 is rerouted to provide protection against the Appendix R analyzed fire. The cable will be sized to provide the same requirements as the existing cable and will be terminated at the same location. The probability of a cable failure is not increased due to same cable type, termination type and same breaker load. This cable reroute is being performed on only one division and will not increase the probability of failure on that division.

The design of the system is such that, on loss of normal air supply (RNA/PNS), a pressure switch signals the solenoid control valves to open and nitrogen will be fed to the vital drywell pneumatic loads. The N2 system is divisionalized and is proof against single active failure. The mod installs isolation bypass switches in the solenoid valve circuits to provide a means to locally de-energize the valves in the Appendix R fire scenarios. The need for these switches is based on the fact that the cables providing the remote valve control will be damaged, possibly causing a "hot short" in the loop. The switches, once installed in the circuit, would not increase the probability of an accident as failure of these switches would not prevent the N2 backup system from performing its safety function and the N2 backup system would still fail-safe (open to drywell). The switches are also only installed in one Division of N2 back-up (Div. I) and would be designed against single active failure.

The RNA/PNS system is Non-Q and is not required for safe shutdown of the plant. Under the LOCA analysis, the N2 backup system would be relied upon to provide the vital loads in the drywell until a long-term nitrogen supply is brought in. Should the cable providing control to the RNA-5261 valve fail, the valve would fail in one of the following scenarios:

1. The Cable deenergizes and the solenoid valve closes. This would stop the normal air supply from reaching the drywell. Upon loss of header pressure, the N2 Backup system would open its supply valves, allowing nitrogen to supply the pneumatic loads.

2. The valve fails in the open position, which would not interrupt th[illegible] supply of air/nitrogen to the drywell. Therefore, if the cabl[illegible], no consequence will result for which an evaluation was not prov[illegible]

The N2 backup system is designed to fail in the safe condition, whic[illegible]ans RNA-SV-5253 and RNA-SV-5482 failing in the open position. This allows [illegible]rogen to supply the vital drywell pneumatic loads. Should the isolation bypa[illegible] switches for these valves fail to provide circuit continuity, the valves would fail in the open position. Therefore, the consequences of any previously evaluated accident would not be increased.

The rerouting of the control cable to the Division II RNA/PNS does not change the failure probability of the system. The cable is procured and installed to the same criteria as the existing system. The reroute of this cable is performed to move this cable out of the fire zone that this portion of RNA/PNS is providing divisional protection for.

The installation of the isolation bypass switches in the Division I N2 Backup system does not increase the failure probability of the system. The switches are installed to provide a local means of allowing the N2 system to operate as designed. Should a fault occur in the control circuits of valves RNA-SV-5253 or 5482 due to an Appendix R fire or other causes, the switches would be used to bypass those faults. Upon bypassing the hot short or other problem, the valves would move to the safe position. The switches are procured and installed per the design criteria specified for the N2 backup system.

No increase in the consequences of malfunction of equipment important to safety will be created by this change. The cable reroute ensures proper system operation. The system is designed to protect against the direct short failure that could occur. The isolation bypass switches failure design ensures that the valves fail-safe. The N2 Backup system would still be available in the event of switch failure.

Divisionalized air systems are provided to protect against cable failure. As this change does not affect any other portion of this system, no new unanalyzed scenarios are created. The cable reroute provides additional protection for the Division II RNA/PNS against the analyzed "North" reactor building fire.

The installation of the isolation bypass switches in the valve circuitry will not introduce accident scenarios not already analyzed in the FSAR. The Division I N2 Backup system has been able to protect itself from a cable fault preventing the actuation of the valves.

Should the RNA/PNS system fail, the N2 Backup system would provide essential pneumatic loads to the drywell instruments and valves. The N2 backup system is designed with a RNA/PNS system failure as its basis. The modification to the system will not reduce the margin of safety, as it does not inhibit the system to fail-safe.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 89-048 and 89-049, Upgrade of Service Water Valve 103 to a Motor Operated Valve

FUNCTIONAL SUMMARY: Recent NRC Diagnostic Evaluation Team Concerns (see EER 89-0135) have focused attention onto the adequacy of the Nuclear Service Water (SYS) to meet Design Bases flow requirements. A single failure criteria analysis has determined that another valve is needed in addition to V106 (Div. I Feed) to provide redundant automatic isolation of the TBCCW Heat Exchanger. This modification will convert Valve SW-V103 from manual operation to motor operation powered from Division II AC power system. This modification will refurbish and utilize available spare motors and actuators to expedite installation.

Calculation TCNG0050A-M-100, "Service Water System Flows at Maximum RBCCW Hx Flowrate, " has determined that the RBCCW Heat Exchanger must be isolated from the Nuclear Service Water System on a Loss of Coolant Accident (LOCA) signal to ensure that the SWS can meet its flow requirements. Prior to this modification, both LOCA and LOOP (Loss of Offsite Power) signals were required to close V106 automatically. This modification will delete the LOOP signal so that V106 closes automatically on a LOCA signal only. V103 will be controlled to close automatically on a LOCA signal only.

Prior to this modification, V106 could receive control signals from both Division I and Division II sources. To provide totally independent redundant operation, Division II control signals will be deleted from V106. The result of this modification will be V106, powered and controlled from Division I sources, and V103, powered and controlled from Division II sources, providing independent, redundant means of isolating the RBCCW Heat Exchanger from the Nuclear Service Water System on a LOCA signal.

SAFETY SUMMARY: Evaluations of the SW system have determined that the single-failure criteria analysis for safety systems was not adequately met, and that the SW pump performance characteristics could not ensure reliable performance of the pumps during the worst-case conditions of the design-basis accident.

Hydraulic analysis of the SW system has determined in the event of a LOCA with single-failure conditions such that only one NSW pump can be run, concurrent with conditions that demand maximize service water flow requirements, the performance characteristics of the SW pumps would put the operating pump in a run-out condition. At the other extreme, during a LOCA where both NSW pumps are available and running, and in a condition where system demands minimize SW flow, the pumps would be operating near their shut-off head.

From a single failure analysis criteria aspect, if a LOCA/LOOP event occurs, SW-V106 should drive close, eliminating the RBCCW System flow requirements from the NSW header. If at this time emergency bus E-3 fails to energize, the SW-V106 valve would not close. Service water flow would continue through the RBCCW System, and adequate cooling flow from the NSW header may not be provided to the RHR heat exchangers and the DG heat exchangers. PMs 89-048 and 89-049 correct these deficiencies.

Changing the operation of SW-V106 and operating SW-V103 concurrently to close at initiation of a LOCA will ensure that a single SW pump will not operate at a run-out condition by eliminating the RBCCW flow path. Evaluation of the remaining SW loads at minimum flow conditions with single failure components analyzed has shown that operation of both NSW pumps during LOCA and LOCA/LOOP conditions will not cause them to operate outside their design capabilities. The changes allow currently imposed operating restrictions, which ensure these conditions will not occur, to be relaxed. Bringing the system back up to design standards will not result in an unreviewed safety question, as the system has been evaluated to ensure the conditions discussed will be precluded. This ensures the correct emergency response and capabilities of the SW system, with adequate margin maintained.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

<u>TITLE</u>: PM 89-053, Unit 1 RCIC F013 Valve Replacement

<u>FUNCTIONAL SUMMARY</u>: Plant Modification 89-053 serves to replace the BSEP Unit One Reactor Core Isolation Cooling (RCIC) Injection Isolation Valve (4" Dia., 900#), Tag 1-E51-F013. The existing valve leaks by the seat when the actuator positions the valve to the closed position (to ensure containment isolation, the existing valve was closed to a leak-tight position using the actuator handwheel). There is not enough stellite facing remaining on the valve seat rings to re-machine the existing seat rings to a leak-tight condition. To completely replace the valve seat rings, the valve must be cut out of the existing piping, requiring a plant shutdown. Replacement of the whole valve (in the same location) was also considered as it was estimated that a shorter plant shutdown would be needed in comparison to the shutdown required for valve seat ring replacement. However, a third alternative was evaluated and determined to have the greatest benefit in minimizing potential outage time. The existing valve 1-E51-F013 will be tightened to provide a leak-tight isolation boundary, thus isolating the RCIC system from reactor pressure. A new motor-operated valve will be installed upstream of Valve 1-E51-F013 while the plant continues operation (until the expiration of the time limitation of the LCO caused by uncertainty of the 1-E51-F013 valve actuator's ability to operate the valve properly.)

The existing 1-E51-F013 will be manually closed to a leak-tight condition, thus providing isolation from the temperature and pressure of the downstream piping (which includes tie-ins with the RWCU System, Feedwater System, and, further downstream, a tie-in to the reactor).

A section of RCIC piping, upstream of the existing valve will be cut from the system and replaced with a new 1-E51-F013 valve. Additionally, the piping between the existing and new valve will be replaced to ensure this piping satisfies ASME Section III Class 1 requirements. (The existing valve is the equivalent of an ASME Section III, Class 1; the piping upstream of the valve was the equivalent of ASME Section III, Class 2. The weld joining the valve to the upstream piping was the class boundary. By this modification, the new valve-to-upstream pipe weld will be the boundary between Class 1 and Class 2, respectively. The pipe between the old and new valves will be Class 1--see following figure.) This piping spool (between the existing and new valves) will include a drain/test connection which will contain double isolation valves to satisfy Class 1 requirements.

After the new valve and attached piping spool are welded in place and all weld NDE is completed, the Limitorque actuator on the existing valve will be removed and installed upon the new valve. This will be done by first ensuring that the existing valve will remain shut (by securing the valve stem with an appropriate mechanical clamping device) then the existing actuator will be taken off the existing valve, the stem nut from the new valve inserted into the existing actuator (to ensure existing actuator/new valve compatibility), and the actuator will be installed onto the new valve. A new manual operator (or Limitorque actuator which will not be connected to an electrical power source) will then be installed on the existing valve. Before mounting this new operator on the existing valve, the valve stem nut from the actuator which had been on the existing valve will be inserted into the new operator (to ensure new operator/existing valve compatibility).

After all acceptance testing is complete, the existing valve will be opened using its new manual operator (or the manual operator on the electrically disconnected actuator) and locked in the open position.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

To reduce the number of plant procedure revisions required by this modification, the 1-E51-F013 valve number will be transferred to the new motor-operated valve, and a new valve number will be assigned to the original valve.

To ensure a smooth transition between the inside surfaces of the new valve and the attached piping (whose inside diameters do not match), the piping spool discussed above will incorporate a smooth transition area which is tapered internally to accommodate the change in inner diameters of the new valve and adjacent piping to facilitate welding. The outside of this transition area will be built up with weld metal to ensure that the piping minimum wall thickness requirements are met. This transition area will also be incorporated into a short section of new piping to be installed upstream of the new valve (see figure below).

The valve stem leak-off line on the existing valve will be left as is; the valve steam leak-off connection on the new valve will be capped. The existing drain line 1-E51-46 tying in immediately upstream of the existing valve 1-E51-F013 will cut back (to a location such that it does not interfere with the new installation), capped, and abandoned in place. Valve 1-E51-V35 will be included in this abandoned-in-place piping.

The new 1-E51-F013 (transferred to BNP) was evaluated by EER 89-0178. The EER (89-0178) currently contains an action item to track the replacement of the modified 1-E51-F013 and 1-E51-V100 valves with a new double disc valve as part of the MOV upgrade. The 1-E51-F013 and 1-E51-V100 valves will be replaced during the next Unit 1 refuel outage or EER 89-0178 will be revised to address corrosion allowance for an extended period of time. This option will be tracked via the EER action item process for EER 89-0178.

SAFETY SUMMARY: Since this modification will not alter the system function (although it will alter the system's piping configuration adjacent to the F013 valve) or normal or abnormal operational modes (although it will add three valves to the RCIC system lineup), then the probability of occurrence of a previously evaluated accident will not change.

As a containment isolation valve, the F013 mitigates the spread of radioactivity into the reactor building. The new valve has been procured to the design equivalent of the existing valve, and will be tested to show that its stroke time is within the time limitations for the old valve. Other performance tests will be conducted to ensure that the valve can be operated properly from the RTGB. Furthermore, the new valve and piping to be installed will be leak rate tested in accordance with acceptance criteria given by plant procedures. With these constraints, the new valve will ensure containment integrity should an accident occur within the primary containment boundary.

The existing valve will serve as a system pressure boundary during installation of the new valve. Acceptance testing will ensure the valve performs its design functions. The new valve and operator are design equivalents of the existing components, and will be tested to appropriate ASME requirements. Therefore, no increase in the probability or consequences of an equipment malfunction would result from this change.

No new accidents or malfunctions are introduced by this change. Appropriate measures will be taken during installation to ensure no reactor safety concerns exist. In addition, by meeting the design requirements of the previous valve, no new malfunctions will be introduced by this change.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

No margin of safety will be reduced as a result of the change. The new motor has been appropriately evaluated for this use. The old valve, although in place and inactive, will not affect system operability.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PMs 89-074 and 89-075, Units 1 and 2, Replacement of SW-F01187 and 1188 Valves

FUNCTIONAL SUMMARY: EER 89-0220 showed that the Unit 1 Core Spray Pu... ...om Cooler 1A, the Unit 2 Core Spray Pump Room Cooler 2D and the RHR Room cool...s do not meet the required flow rate for design basis heat removal at a Service Water temperature of 90 degrees F after the ten minute phase following a DBA. To correct the deficiency in the Core Spray Pump Room Coolers, flow orifices 2-SW-F01187 and 1-SW-F01188 will be modified to increase the flow through these heat exchangers to the rate stated in the FSAR of 55 gpm at minimum SW system flow conditions. Analysis demonstrates that the heat exchangers will deliver the design base heat removal capability at a flow rate substantially less than stated in the FSAR (300 gpm). The conservative figure of 210 gpm per cooler will be used as the required flow rate at the design base temperature of 90 degrees F., based on manufacturer's required rate of 186 gpm (93 gpm per heat exchanger) plus 10% margin per cooling unit.

SAFETY SUMMARY: The modification of the flow orifices does not constitute a change to the SW system, but a restoration of flow to original design conditions at the CS pump room coolers in question. The amount of flow increase through the CS pump room coolers represents decrease of flow at the RHR room cooler of 1.1%. This small change is not considered to have a significant impact on the heat exchanger performance. The flow requirements established in the RHR room cooler manufacturer's analysis for the design basis temperature further ensures that the flow rate is adequate in the RHR room coolers. This modification is restoring the system to meet the original design requirements, hence the original safety analysis remains the same, and new safety concerns (accident occurrence probabilities and consequences) are not created by this change.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: PC 87-001 and 87-002, Unit 1 and Unit 2, Process Computer Software Upgrades

FUNCTIONAL SUMMARY: The Unit 1 and Unit 2 Process Computer Nodal P1 Software was upgraded from GEXL+15 to NODAL-P1, to accurately monitor the new GE Advanced Fuel Designs. These new fuel designs are more complex than the previously loaded fuel in several ways. The foremost difference is the fact that the advanced fuel has an axially varying design, which is opposed to the axially constant design of the current fuel. This results in a great increase in the complexity of the monitoring model. The NODAL-P1 Process Computer Software Upgrade ensures accurate monitoring of the more complex fuel. The specifics of the upgrade are considered proprietary information, and are not included in this report. More information may be obtained on this new software product in the GE White Paper CORE MONITORING WITH ADVANCED FUEL DESIGNS, REV. 1.

SAFETY SUMMARY: The effect of the new software model on the plant process computer is limited to the NSS software and data banks. No changes in the hardware are required to implement the new software model. The accuracy of the new software's power distribution model was shown to be consistent with that of the previous software. This was accomplished during methods qualification prior to the J-factor method's implementation for use in core performance monitoring. Differences between the old and new methods were found to be small and statistically random relative to the differences between the old method and gamma scan results. Differences in computed thermal margins exceeding 2% were investigated and resolved prior to implementation of the new software for use in core performance monitoring. Since no hardware changes are needed and since the new software has been favorable compared with present software and will be properly installed and tested prior to use, the probability of occurrence of an accident or malfunction will not be increased.

Since the only changes are software changes which have been shown to produce power distributions and thermal margins comparable to that of the present software, no new accident or malfunctions of a different type will be involved.

The changes will not change any Technical Specification limits or safety margins. The GETAB thermal limits analysis is not affected by the use of new method in monitoring core performance. The accuracy of the new model has been compared with the old model, and is fully consistent with the accuracy of the previous model.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

<u>TITLE</u>: EERs 88-206 and 89-0062, Unit 1 and Unit 2, Temporary Deactivation of the Reactor Head Vent Line

<u>FUNCTIONAL SUMMARY</u>: In order to eliminate Appendix R concerns with the current configuration of the Reactor Head Vent valves, B21-F003 and F004, these valves are being temporarily secured in the closed position. The B21-F003 and F004 valves are considered high-low pressure interface valves in the Appendix R design. The current configuration of the head vent valves' control wiring is such that it is not adequately protected from a spurious operation due to a fire in the reactor building or control room. As such, they could spuriously open, creating an unwanted depressurization of the RPV, and a loss of water inventory from the reactor vessel. This condition is not analyzed in the Appendix R design and therefore is undesirable. Plant modifications are required to correct this design deficiency. In the interim, the F003 and F004 valves will be secured and manually closed. These EERs provide the justification for removing the valves from service, and also provide instructions for deactivation of the valves.

<u>SAFETY SUMMARY</u>: Procedure and operating practices reviews have identified alternate vent paths for the reactor. These vent paths will ensure that the RPV is adequately vented during startup and shutdown of the Unit. No credit is taken for the head vent valves in ASSD procedures. The valves are utilized in other Operations procedures, but not in methods differing from those used during normal startup and shutdown. These valves closed whenever the reactor temperature is above 212 degrees F.

The F003 and F004 valves were originally installed to permit remote venting of radioactive gases which may accumulate in the reactor head space during reactor cooldown after the steam lines have been flooded or the MSIVs have been closed. This venting is performed prior to head detensioning, and can be accomplished via the B21-F001 and F002 valves. The GE System Engineer has stated that drawing a vacuum is not a concern.

Based on the above, it can be concluded that disabling the F003 and F004 valves will not increase the probability of occurrence nor the consequences of any accident previously evaluated in Chapter 15 of the FSAR. Nor will this change increase the probability of occurrence or the consequences of any malfunction of equipment important to safety. This change will not create any new accident or malfunctions, nor will it reduce the margin of safety as defined in the basis of any technical specification.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EERs 89-001 and 89-021, Unit 1 and Unit 2 Secondary Containment Damper Isolation Evaluation

FUNCTIONAL SUMMARY: Evaluations determined that if reactor non-interruptible air (RNA) pressure is lost to the secondary containment isolation dampers (both Division I and Division II dampers are supplied by Division I air), the damper will fail as-is. Since the non-interruptible air system in the reactor building is not environmentally qualified and no longer classified as Q-list, the air system may not by available at all times.

The isolation valves are provided with ASME Section VIII seismic accumulators to ensure the valves close on an initiation signal. However, the air system may not be available to maintain pressure in the accumulators, which may mean that secondary containment would not be maintained for an extended time. As a temporary measure (action to permanently correct deficiency has been completed), a diesel driven air compressor will be provided to be aligned to the instrument air header during an emergency situation, in order to maintain an adequate air supply to the damper to keep it closed and preserve secondary containment. The compressor will be tested on a daily basis to ensure its proper operation when needed. Also, since closure of a single valve could isolate air to both divisions, caution tags have been placed on appropriate Division I RNA valves which could isolate the air supply to both divisions of the secondary containment isolation dampers. It is highly unlikely that an operator would isolate a RNA valve during power operation; however, it was considered prudent to take mitigating actions to prevent this occurrence.

SAFETY SUMMARY: The design of the secondary containment and isolation valves is to mitigate the consequences of an accident. A malfunction of the system will not initiate an accident. The temporary air compressor, as currently installed, does not directly interface with any systems important to safety, and therefore cannot have an affect on the probability of an accident.

The accident scenarios evaluated in the FSAR that require secondary containment are a line break inside primary containment and a refueling accident. Since the reactor building Non-interruptible air system is a seismic Class I system, the system will remain intact and operable in a mild environment. Following a line break, the environment inside the reactor building may cause failure of the standby air compressors. In this case, the use of the temporary air compressor will ensure an air supply to the secondary containment isolation valves over the long term. This will ensure that secondary containment is maintained. The temporary air compressor will provide backup to the existing air compressors in order to ensure the consequences of an accident will not increase.

The secondary containment isolation valves will fail as-is on a complete loss of instrument air. Seismic category I accumulator tanks are provided to ensure that the valves will have an adequate air supply to close on an isolation signal. Procedural controls have been implemented to ensure that if instrument air is lost, the valves will be closed manually so that their leakage requirements are met. In the event that all instrument air is lost as a result of an accident, the valves will be manually closed if access to the reactor building is possible. If access to the reactor building is not possible, the corrective actions outlined in this EER will ensure that an air supply is available for long term secondary containment integrity. The temporary air compressor will not supply air to any components except in the event of an accident when all other air sources are lost. If the compressor is needed, the components it will supply (if supply pressure is maintained below 95 psig) will not need to be operated, but will be maintained in their post-accident position. Therefore, the quality of the air will not affect their ability to function.

The consequences of failure of equipment classified as important to safety will remain unchanged. No additional requirements have been placed on any equipment important to safety that have not been evaluated in the FSAR.

The nature of secondary containment and the associated isolation valves is such that no failure of this equipment can initiate an accident, whether or not analyzed in the FSAR. The possibility of failure of the containment isolation valves and subsequent loss of secondary containment two hours after an accident due to a loss of instrument air has not been analyzed. Due to the amount of time that is available to the plant staff to restore air to the secondary containment isolation valves and the number of options given in order to restore the air, adequate assurance is available that this can be accomplished within the required time frame.

The basis for the technical specifications indicates that testing of the SBGT system and the secondary containment isolation valves is adequate to ensure that there are no violations of secondary containment integrity. This does not account for the long term maintenance of secondary containment following a loss of instrument air. The use of operator action in order to maintain the integrity of secondary containment also is not taken into consideration.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 89-011, Temporary Deviation of Unit 2 Safety Related Throttle Valves from ANSI B16.5

FUNCTIONAL SUMMARY: FSAR Section 3.1.2.4.1 requires valves to be fabricated and designed to ANSI B16.5. Erosion of the valve walls has decreased the wall thickness to below the requirements of ANSI B16.5. This EER justifies the acceptability of the wall thickness of the valves versus ANSI B16.34.

SAFETY SUMMARY: The existing condition of the 2-E11-F017 A & B and 2-E11-F024 A & B valves meets code minimum wall requirements using the actual design pressures for the system piping. The valves will be monitored to ensure these minimum wall requirements are not exceeded and sufficient wall thickness remains for them to fulfill their intended function. The EER shows that the RHR system can be continuously operated in the LPCI, Suppression Pool Cooling, and the Shutdown cooling modes of operation for an extended period of time before erosion reduces wall thickness of the Valves to a point that code minimum wall thickness standards would not be met. The valves and the RHR system are not generally used during operation, and can be monitored or operated during a LOCA in such a manner that significant erosion which could affect the ability to accomplish a required safety function can be detected or prevented.

Measures have been taken to ensure the valves do not degrade to unacceptable conditions such that required safety functions could not be accomplished. If conditions did degrade to a point where the valve(s) failed, an analysis for line breaks outside primary containment has been evaluated in the FSAR which bounds this event. The conservatisms delineated in the safety analysis performed under EER No. 89-011 and compensatory measures implemented through this EER ensure the RHR systems will be available for both the short and long term cooling requirements following a design basis accident.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 89-0091, Evaluation of Temporary Flange Installation on 2-G16-V69

FUNCTIONAL SUMMARY: A quarter inch flange was temporarily installed on the bonnet of the 2-G16-V69 valve. This is an isolation plug valve in the dewater line from the Chem. Nuclear dewatering facility to the Radwaste building. The valve is leaking and is to be repaired by the plant Trouble ticket process. The length of repairs to the valve would not permit continued dewatering operations, and therefore the flange installed as a pressure boundary is necessary to continue dewatering activities while the valve is being repaired.

SAFETY SUMMARY: Valve 2-G16-V69 is an isolation valve which allows flow between the Chem Nuclear dewatering facility and the Radwaste building. The valve does not involve any ISI boundaries nor does it perform any ECCS or RPS activities. The flanging of this valve does not create a more severe environmental condition that assumed in the SAR, nor does it reduce the reliability of a safety related component or system or reduce the redundancy of safety related equipment or systems assumed in the SAR. No reduction in the rating or design margin of safety related components in introduced by this change, nor is the frequency of vendor recommended preventive maintenance or inspections reduced. The flanging does not violate single active failure or separation criteria, nor increase the probability of a common mode failure. The flanging of this valve does not impact the validity of assumptions or calculations used for evaluations in the SAR because the design and function of this system remains unchanged.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 89-0124, Evaluation of Service Water Intake Structure Supply Fan Power Supply

FUNCTIONAL SUMMARY: This engineering evaluation involves removing two loads from a Q-list 120/208 Volt AC Emergency Distribution Panel. There are two Service Water Intake Structure (SWIS) supply fans, 1A-SF-SWIS and 2A-SF-SWIS. These fans do not perform any safety-related functions, and are presently inoperable and connected to distribution panel 2A-SW-HQ4. This distribution panel supplies loads that involve the Unit 2 Circulating Water Traveling Screens, the Unit 2 Circulating Water Pumps Bearing Lube Water, and the Unit 2 Service Water Traveling Screen 2A. The loads are not specifically safety-related, but they can have impact safety-related systems. The sparing of the unused and unnecessary cables will minimize the chances that a fault in one of these cables could impact this distribution.

SAFETY SUMMARY: The two SWIS supply fans are presently inoperable. The sparing of the unused and no longer necessary cables will minimize the chances that a fault in one of the cables could impact the distribution panel. The sparing of the cables will have no adverse effects on the other loads supplied by this panel. Therefore, the probability of occurrence or consequences of any accident previously evaluated in the FSAR will not be increased.

The distribution panel supplies loads that are not specifically safety-related, but can have an impact on a safety-related system. The sparing of the unused power cables will minimize the chances that a fault in one of these cables could impact this distribution panel, and thus affect one of the other loads. Therefore, the probability of occurrence and consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased.

The SWIS supply fans are presently not required and the fans are inoperable. The fan power cables connected to distribution panel 2A-SW-HQ4 represent an unnecessary load on the distribution panel, and a path for a potential fault to affect the other loads on the panel. The sparing of these unused power cables will minimize the chances that a fault in one of these cables could impact this distribution panel or any other loads. Therefore, the possibility of an accident or malfunction of equipment important to safety of a different type will not be created.

The sparing of the SWIS supply fan power cables will minimize any affect that these cables could have on any other load. Therefore, the margin of safety as defined in the basis to any technical specification will not be reduced.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

<u>TITLE</u>: EER 89-0166, Verification of Acceptable SW Flow to RHR for Worst-Case Expected LOCA Containment Cooling

<u>FUNCTIONAL SUMMARY</u>: This engineering evaluation was written as a result of DET concerns regarding inconsistent FSAR information for allowable minimum flow rates for RHR and RHR-SW in the containment cooling operating mode and the acceptability of the resulting maximum suppression pool temperatures. This evaluation determined that Safety Guide 1 assumptions regarding determination of ECCS pumps NPSH were not considered fully applicable for all containment cooling modes and were satisfied. In particular, Case D of Table 6.2.1-8 of the FSAR took credit for containment pressurization which is not allowed by the Safety Guide. Alternate acceptable operating requirements are established for current conditions (for LOCA response) and an action item is generated for final resolution of FSAR Table 6.2.1-8.

<u>SAFETY SUMMARY</u>: The containment cooling function of RHR-SW and RHR, used to adequately cool primary containment, is a function needed for accident mitigation. The function is not in service or needed before an accident, so it cannot cause an accident. Therefore, it does not affect the probability of occurrence of an accident.

This evaluation has established minimum operating conditions which are within the capability of the RHR and RHR-SW systems and which allow the systems to fulfill their intended safety function. The resulting peak suppression pool temperature is within the required limits such that the design limits, EQ requirements, and NPSH requirements are satisfied. Since the systems remain capable of performing their intended safety function to mitigate an accident, the is no impact on the consequences of an accident previously analyzed in the FSAR.

This evaluation reduces the allowable range for operating RHR-SW system from 4000-8000 gpm to 4500-8000 gpm. Since this is only a reduction in operating range and is not a change in the manner in which the system is operated, there is no affect on the probability of malfunction of the RHR-SW system. The evaluation also lowers the maximum SW system inlet temperature from 95 degrees F to 90 degrees F for the RHR-SW system. The 90 degrees F limit is consistent with the design limit for the SW system and its normal RHR, RBCCW, and TBCCW coolant loads, as is thus acceptable. Since the RHR-SW system remains able to satisfactorily cool the suppression pool and limit its peak temperatures such that there is no unacceptable impact, there is no increase in the probability of malfunction of equipment important to safety.

While this evaluation reduces the allowable ranges for operation of the RHR-SW and SW systems, the systems are not operated in a different manner. The systems will continue to perform their intended functions and limit suppression pool temperatures to an acceptable level, therefore the consequences of failure are not increased.

The reductions in operating ranges for the RHR-SW and SW systems have been determined such that the RHR-SW system will continue to fulfill its intended safety function and limit peak suppression pool temperature to an acceptable limit which satisfies the applicable criteria. Technical Specification basis 3/4.6.2 specifies a maximum temperature at the end of a blowdown from a LOCA be less than 170 degrees F. Since containment cooling is not presumed to be initiated until after blowdown, containment heat removal capability does not affect this basis.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

The current Technical Specification also specifies that excessive steam condensing loads can be avoided if suppression pool temperature is maintained below 160 degrees F during any SRV operation with sonic conditions at the discharge exit. This basis is applicable to the old "rams head" SRV discharges. Due to newly discovered dynamic loads evaluated under the Mark I torus integrity program, these were replaced with "Tee-quenchers". The applicable temperature limit for the suppression pool per NUREG-0783, Suppression Pool Temperature Limits for BWR Containments, is now 200 degrees F for significant steam fluxes and higher for lower steam fluxes. The peak suppression pool temperature with 4500 gpm RHR-SW flow was found to be less than 194 degrees F, satisfying this limit, which establishes the required margin of safety. The applicable margins of safety as defined in the basis are therefore not reduced.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER No. 89-0195, Compliance with PCIS Table in Technical Specifications

FUNCTIONAL SUMMARY: During an audit of the Inservice Testing Program by the NRC Diagnostic Evaluation Team, it was identified that some valves were not being tested in accordance with technical specifications - specifically, Appendix J testing and stroke time testing. This engineering evaluation explains the basis for why the Appendix J testing is not required by Technical Specifications, adequacy of current stroke time testing, and corrective action for deficiencies.

Listed in Table 1 are those valves which have been questioned as to compliance with technical specifications. The testing may be applicable to these valves as follows:

Description	Valve Number
RHR Sample Valves	E11-F079A E11-F080A E11-F079B E11-F080B
RHR drains to radwaste	E11-F040 E11-F049
HPCI suction from torus	E41-F041

The following sections will examine each of these tests, current status, and any action items necessary to ensure these tests will be performed until the technical specification revision is approved by the NRC.

SAFETY SUMMARY: These valves are not considered primary containment isolation valves in the FSAR; thus, this testing is over and above the requirements of the FSAR.

The FSAR requires ASME Section XI testing. ASME Section XI stroke time tests all these valves closed except E41-F041. This EER adds closed timing on E41-F041 and implements technical specification time limits on the remaining valves.

This is only an interim measure until the latest PCIS technical specification is approved by the NRC. This EER only adds additional testing requirements which is a conservative addition. This EER ensures compliance with existing PCIS technical specification requirements. This should ensure proper function of valves currently in the PCIS tech. spec. table. Stroke time testing should not affect the consequences of malfunction of equipment. Stroke timing of valves has been considered in the evaluation of the FSAR. This EER ensures compliance with existing PCIS technical specification (Table 3.6.3-1).

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER No. 89-0220, JCO for Adequacy of SW System to Meet Design Basis Performance Requirements

FUNCTIONAL SUMMARY: The Nuclear Regulatory Commission (NRC) assembled a Diagnostic Evaluation Team (DET) which performed an independent evaluation of CP&L performance at BNP with emphasis on the Service Water (SW) system. Several key issues pertaining to operability were identified. EER-89-0135 Rev. 1 previously addressed these issues; this EER revises the operating limits for the SW system after corrective actions for the RBCCW valve V103 have been implemented and supersedes EER 89-0135 Rev. 1 in part only. This EER utilizes the same approach in determining the operating limitations, but is based upon a more accurate computer model of the SW system, vendor information on the NSW pumps, and implementation of a modification on the V103/V106 valves [Plant Modifications 89-048 (Unit 1)/89-049 (Unit 2)]. This EER addresses the following items:

1) Maximum RBCCW Flow Rates
2) Minimum RBCCW Flow Rates
3) Maximum temperature limits on SW based on flow rates to components

Maximum flow rate to the RBCCW is a concern due to possible NSW pump runout under certain conditions, including diversion of cooling water to non nuclear safety components. Minimum RBCCW flow rate is a concern due to possible NSW pump operating problems at too low total flow rate to the system, resulting in overload to the motor thrust bearing.

SAFETY SUMMARY: The Service Water system hydraulic analytical model was revised to incorporate implementation of the V103/V106 modification (which installed controls to allow the RBCCW outlet valves V103 and V106 to close to a throttled position upon receipt at a LOCA signal). The revised computer model was utilized in conjunction with hand calculations to determine the operability limits of the RBCCW SW flow, and SW temperature maximum limit. The combined analyses, included primarily as attachments to this engineering evaluation, form a basis which ensures that the SW system can perform its intended design function as described in the FSAR.

The analyses demonstrated that under worst case flow conditions the RBCCW can initially flow 7200 gpm prior to an accident, and not result in NSW pump runout due to excessive total system accident flows. The V103/V106 controls ensure that flow through the RBCCW is limited to prevent NSW pump runout. The analyses also demonstrated that the V103/V106 controls will not cause damage to the NSW pumps due to insufficient total flow rate in the system under minimum flow scenarios. It was shown that throttling of the V103/V106 valves allowed enough total system flow under various valve lineups to prevent excessive thrust on the pump bearings. The RBCCW flow rate may be as low as 2500 gpm before an accident without causing damage to the NSW pumps due to insufficient flows. The analyses also demonstrated that design flows are not provided to the Core Spray and RHR Room Coolers under the worst case accident scenario. Based upon the limited flows, the SW system inlet temperature when in the normal lineup is limited to a maximum of 89.2 degrees F to ensure that accident heat loads will be transferred to the Ultimate Heat Sink.

In conclusion, the SW system analytical model, along with calculations, has been utilized to demonstrate the maximum and minimum RBCCW flow rates needed to ensure no NSW pump runout or thrust bearing damage occurs (7200-2500 gpm) and to demonstrate that the SW system can meet its intended design function if the inlet water temperature is equal to or less than 89.2 degrees F.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER No. 89-0221, EQ Impact Evaluation of Future MSIV Pit Plug Removal at Low Power Operation

FUNCTIONAL SUMMARY: This EER documents that credible high energy line piping crack scenarios, postulated in the MSIV Pit/MS Tunnel while the MSIV Pit plugs have been removed under low power operation, are enveloped by the originally established (Reactor Building general area) generic environmental profile [shown in EER Attachment 2]. This conclusion that no impact to the BSEP EQ Program exists (during occurrences of an MSIV Pit plug removal at approximately 25% power operation) is based upon supplemental HELB/critical crack analyses (particularly analyses that are contained in UE&C Study Report No. 9527.007-S-N-030, dated 12/9/86), and based upon confirmation of which HELB/critical crack scenarios shall be considered to be credible events (through probabilistic determinations, per the BSEP PRA Study and NUREG/CR-4792). Operational considerations/limitations are currently controlled by OG-10.

SAFETY SUMMARY: Since this EER utilizes a probabilistic methodology (rather than deterministic) to confirm which scenarios are in fact considered credible, PNSC concurrence on this safety evaluation with respect to questions related to the "probability of accident" and the "margin of safety" is needed.

In this regard, safety evaluation questions associated with the "probability of accident" and with the "margin of safety" have been answered based upon the fact that all probabilities identified for credible critical crack scenarios are within the NRC's (generally recognized) acceptable risk level, and that DEG break scenarios are highly unlikely. [Reference Section 2.4 of this EER.] Therefore, questions associated with the "probability of accident" and the "margin of safety" have been answered with the assumption that all possible HELB/critical crack scenarios are not probable and that only certain critical crack scenarios need be considered.

As identified within this EER, the effects (within the Reactor Building general areas and within the MSIV Pit/MS Tunnel) of these certain critical crack scenarios, under the limited operating conditions, demonstrate that no increase to the "probability of accident" exists, or that no decrease to the "margin of safety" exists, from that previously assumed in the safety analysis report.

Based upon these conclusions, the following additional bases have been provided for the nuclear safety evaluation questions. These bases serve to confirm previous nuclear safety evaluations associated with the issuance of EER No. 85-0320 and OG-10.

This EER does not increase the probability of occurrence of any accident (or its consequences) previously evaluated in the FSAR.

The effects of ("worst case") HELBs in the MSIV Pit/MS Tunnel were considered in the design of the BSEP Units 1 and 2 (as documented per the correspondence included within EER Attachment 3). This EER further amplifies the unlikely probability of these identified HELBs. For completeness, the remote probability of critical crack scenarios were also identified per this EER, and as noted above, shown to be within an acceptable risk level.

Furthermore, since this EER confirms that mass and energy releases (for any of the identified scenarios) into the Reactor Building general areas will be less than that previously analyzed in the RBER (for a "worst case" break in the Reactor Building general area), the effects of any of the identified scenarios will also be less than the consequences of the "worst case" break in the Reactor Building general area.

This EER does not increase the probability of occurrence of malfunction (or its corresponding consequences of malfunction) of equipment important to safety previously evaluated in the FSAR.

The EQ-related equipment (located in both the MSIV Pit/MS Tunnel and the Reactor Building general areas) has been qualified to its required generic qualification profile, which envelops the environmental conditions postulated within this EER. Therefore, the probability of equipment malfunction (and its corresponding consequences), as previously analyzed, would not be changed.

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.

Although the conditions associated with the accident scenarios evaluated within this EER are different than those previously evaluated in the RBER and the correspondence included in EER Attachment 3 (i.e., the vent path resulting from a MSIV Pit plug removed), the assumptions of only credible crack scenarios and limited operating conditions provide assurance that previously established limiting analyses in the FSAR continue to envelop the conditions evaluated by this EER.

Further assurance is provided by the extremely small probability of this type of critical crack scenario, and by the fact that the equipment is qualified to function as required under exposure from more severe accident scenarios.

The margin of safety as defined in the basis to any Technical Specifications will not be reduced.

There is no basis for any change to a Technical Specification requirement, and therefore there is no specific reduction to the margin of safety defined in Tech Spec bases. While the MSIV Pit plugs are not Technical Specification related, EER No. 85-0320 and OG-10 address the Pit plugs under Primary Containment Structural Integrity for administrative control only.

The margin of safety is further assured based on: (1) the extremely small probability of occurrence which exists for the credible critical crack scenarios identified within this EER; and (2) the extensive amount of margin [relative to environmental conditions postulated for these critical crack scenarios] demonstrated in environmental qualification testing/analysis as documented within BSEP QDPs.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 89-0248, Temporary Use of Fire Protection Water for Battery Discharge Tests and HRD Tester Calibration

FUNCTIONAL SUMMARY: Plant battery/chargers 2A-1, 2A-2, 2B-1, and 2B-2 are to be tested per Maintenance Surveillance Test 2MST-BATT11R, 13R, 15R, and 17R, using the Hi-Rate Discharge (HRD) tester. The HRD tester requires cooling water during performance of these tests. The only readily available source of cooling water near the battery rooms is the fire protection system supply. This evaluation determined the acceptability of using the Fire Protection system to supply the HRD tester cooling water.

SAFETY SUMMARY: The use of a temporary connection on the battery room fire protection system supply will not adversely affect previous FSAR evaluations, nor will it result in adverse operating conditions for the fire protection system since Technical Specification required water levels are maintained, and the hose station remains operable. The use of the fire protection water at the hose station will not prevent use of the station for fire fighting efforts, nor will the Technical Specification margin of safety be reduced. Blocking doors open will impair the fire barriers between the stairwell, cable spread, and battery room; however, appropriate technical specification action statements will be adhered to during the time periods when the door(s) are open.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 89-0253, Providing Design Bases and Evaluation of SW system Operability Requirements in Mode 4 and Mode 5 for Unit 2

FUNCTIONAL SUMMARY: This engineering evaluation provides a technical basis for service water pump and system operability requirements for BNP Unit 2 while in Modes 4 and 5. The current Technical Specification Interpretation does not differentiate requirements between modes. Additional operational flexibility is required to execute outage requirements.

The general approach to establishing this basis is to:

1. Define the system requirements for these modes of operation.

2. Establish by analysis the minimum equipment required to meet the established system requirements.

Since each unit is cross-tied to the other unit as an alternate feed to the other unit's Emergency Diesel Generators (EDGs), this evaluation must address requirements for both units even though only one unit may be shutdown.

SAFETY SUMMARY: EER 89-0253, Rev. 0, evaluates the design bases for BSEP Units 1 and 2 for the scenarios in which Unit 2 is shutdown and Unit 1 is any operating mode, and evaluates the capabilities of the two SW systems to meet those bases given the restrictions imposed by the Unit 2 shutdown. Operational restrictions are evaluated and recommended where necessary to maintain design basis capability of one or both SW systems. Design basis capability and/or operating restrictions applicable when both Units are at power have been evaluated in previous EERs and calculations.

The Unit 1 and Unit 2 SW system computer models generated as part of the effort to address the NRC DET questions were utilized to evaluate the design capabilities of the SW systems in response to design basis accidents and/or transients. The models have been representative of SW system response. The specific analyses addressed as part of this EER are formalized in calculations G0050A-10, Rev. 1, and G0050A-12, Rev. 0.

To generate a meaningful analysis of the SW system using the KYPIPE SW system models requires a clear definition of those design basis accidents and/or transients which are the most limiting with respect to SW system capability. In general, the function of the SW system is to provide component cooling for those safety-related components listed in FSAR Table 9.2.1-1 during any accident or transient requiring the component to operate. The accidents and transients which must be evaluated are discussed in Chapter 15 of the UFSAR. In particular, section 15.0.A contains commonality charts which define the requirements for SW and other systems for the various operating modes. The commonality charts also indicate which events must be able to withstand a single failure.

Calculations G0050A0-12 and G0050A-10 used a single failure logic approach to determine which design basis accidents/transients, when coupled with the worst case single active failure, would be the most limiting for the SW system. These limiting cases were then analyzed incorporating any additional limitations caused by the shutdown status of Unit 2 (specifically, the need for Unit 1 to carry all four Diesel Generators when the Unit 2 SW header is down).

Based on a review of Chapter 15 and the most limiting single active failures, the worst case scenarios for SW system design performance were identified and analyzed in Calculations G0050A-10 and G0050A-12. In several instances, these analyses identified areas in which the SW pumps were being operated beyond an acceptable flow rate or in which the SW system could not deliver sufficient flow to required safety-related components. Operating restrictions were evaluated to ensure safe operation of the SW pumps and design flow capacity to all required components. With imposition of the operating limits via this EER, design capability of the SW system is ensured.

This EER allows operation of RHR SW at flows above the indicated restrictions for the purpose of performing required testing. Technical Specifications allow one SW pump to be out of service for up to 7 days. Placing that pump out of service puts the SW system in an abnormal condition for up to 7 days. The extension of the RHR SW flow places the SW system in a similar situation since the additional RHR SW demand on the system can be treated as the decrease in system capacity due to the pump out of service. And, since testing continues for only a matter of hours, the risk is small in comparison to the allowable risk under existing Technical Specifications for a pump out of service.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 89-0305, Evaluation of Receipt Inspection Non-Conforming Items

FUNCTIONAL SUMMARY: During the receipt inspection of the Reactor Coolant Recirculation Nozzle safe-ends purchased on P.O. 557385, QC and Engineering noted 26 items requiring clarification or supporting documentation as required by Specification 248-153, Rev. 0. This EER will address the appropriate items requiring an engineering evaluation. The item numbers refer to the appropriate non-conforming item numbers from the MRR No. 89-2930, Attachment No. 1. This EER also addresses the equivalent nonconforming items from MRR 89-2951, Attachment No. 3.

SAFETY SUMMARY: The acceptance of the safe-ends as received will not change the RCR system operating characteristics and will meet or exceed the original design criteria. The received safe-ends are equal to the ones ordered in strength requirements and IGSCC resistance. Therefore, long term reliability and structural integrity of the RCR system will not be compromised by the installation of the as-received safe-ends. Based on these considerations the following conclusions are drawn relative to 10CFR50.59 requirements:

1. The probability of occurrence or consequences of any accident previously evaluated in the FSAR will not be increased.

2. The probability of occurrence or consequences of malfunction of equipment important to safety previously evaluated will not be increased.

3. The probability of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the FSAR will not be increased.

4. The margin of safety as defined in the bases to any technical specifications will not be reduced.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 89-0337, Temporary Disabling of the No. 1 DG Engine Driven Jacket Water Pump

FUNCTIONAL SUMMARY: The Diesel Generator Engine driven jacket water pump is being temporarily disabled until such time as permanent corrective actions can be developed, planned, scheduled, and implemented. The disabling is the result of excessive vibration of the Engine driven jacket water pump during operation. The vibration is felt to be the result of a problem with the pump internals or the pump drive interface.

SAFETY SUMMARY: The Motor Driven jacket water pump (MDJWP) is a backup to the Engine Driven Jacket water pump (EDJWP) that is being disabled. The MDJWP has the same capacity as the EDJWP and can therefore meet the system cooling requirements. The MDJWP will automatically start and deliver design flow after the No. 1 DG has energized the Bus E1. The EDJWP is being disabled in such a manner that the structural integrity of the jacket water piping will not be compromised during a seismic event. With the EDJWP disabled, there will be a loss in the redundancy of jacket water cooling supply. This, however, is not a safety concern since an individual DG is not designed to be single failure proof, and the loss of the MDJWP will only impact the #1 DG. In order to ensure that the MDJWP remains available, compensatory actions will be implemented.

The FSAR states that a single failure analysis has been conducted that shows that no single failure in the standby AC system auxiliaries could result in the loss of more than one DG set. The loss of the MDJWP for DG No. 1 will only require result in the loss of operability of the No. 1 DG. The FSAR also evaluates for situations where 3 DGs are operating and under worse case loading conditions, in no case is the continuous rating of any DG exceeded even though it is allowable for the rating to be exceeded by 10%. Based on these conclusions, no unreviewed safety question exists due to the temporary disabling of the No. 1 DG EDJWP.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: EER 88-0517, Leakage Investigation by Adding Florescent Additive to the 2B Recirculation Pump Motor Oil

FUNCTIONAL SUMMARY: To help determine the origin of an oil leak on the 2B Reactor Recirculation Pump Motor, a yellow florescent dye was added to the oil reservoir.

SAFETY SUMMARY: The small amount of florescent dye added to the Reactor Recirculation pump motor oil reservoir will not adversely affect the performance of the pump motor. Degradation of the oil will not result in levels of potassium hydroxide per gram in excess of manufacturer limits. Failure of this pump is bounded by the analysis of the Reactor Recirculation pump motor seizure. The bearing lubrication will not be significantly affected by the addition of the dye. Although the degradation of the dye will not result in excessive levels of potassium iodide, an analysis has been performed which shows no appreciable degradation of the dye due to radiation doses expected.

CHANGE TO THE FACILITY AS DESCRIBED IN THE FSAR

TITLE: SW Modification Review: Required Revision to Regulatory Guide 1.97, Variable D22

FUNCTIONAL SUMMARY: Service Water Plant Modification Review for Brunswick Steam Electric Plant Units 1 and 2 identified a concern regarding the scale range change for the Residual Heat Removal Service Water Flow Indicators. These Flow Indicators monitor cooling water flow through the RHR Heat Exchanger tubes. The review comments on the As Built indicator range of 0-8400 gpm versus the range of 0- 12000 gpm submitted in the Brunswick Response to Regulatory Guide 1.97, Variable D22.

Variable D22, Cooling Water Flow to ESF System Components, concerns are as follows:

A. Item (a) Instrument range

   0 - 110% of design flow is recommended. The Brunswick submittal is 0 to 150%, 0 - 12000 gpm for RHR Service Water.

B. RHR Heat Exchangers Service Water Flow Transmitters E11-FT-N007 A/B. The Brunswick submittal is a Range of 0 - 800 in. water

C. RHR Heat Exchangers Service Water Flow Indicators E11-FI-R602 A/B. The Brunswick submittal is a Range of 0 - 12000 gpm.

The maximum tube design flow for the RHR Heat Exchangers is 8000 gpm. The RHR Service Water Pumps, two redundant loops, two pumps per loop, have a design flow of 4000 gpm each. The flow transmitter and flow indicator in each loop, are calibrated for a range of 0 - 8400 gpm (0 - 434 in. water).

The loop calibration provides an instrument range of 105% design flow. Two RHR Service Water Pumps per loop produce a collective flow of 8000 gpm tube flow through the RHR Heat Exchangers. This range of interest provides accurate and reliable RHR Service Water performance indications.

SAFETY SUMMARY: The RHR Service Water Pumps can fulfill their intended Nuclear Safety Related function during a Design Basis Event by flowing service water to the RHR System Heat Exchangers. Safety related design, installation, and testing practices, along with the use of Q-list materials where required, ensure that there is no reduction in reliability of the system.

The new 0 - 8400 gpm range will add positive flow control monitoring based on true design flow margins. The enhanced flow control indication will provide the control room with information to evaluate RHR cooling and prevent throttling RHR Heat Exchanger service water flow below recommended minimum design flow limits following a DBE thereby, reducing the consequences of an accident.

The flow of RHR Service Water through the RHR Heat Exchanger is accomplished with redundant Safety Related loops which precludes the occurrence of malfunction following a DBE or during Shutdown cooling. Worst case failure of the components installed could only effect components within the same division which would leave the other division to perform the safety function. The change in range of flow indications does not introduce any credible failure potentials that could effect both divisions.

Accurate design flow indication will help to ensure that the RHR Heat Exchangers will provide containment cooling during Shutdown cooling or following a Design Basis Accident and will be available to supply post accident cooling needs. The design flow range indication serves to detect low RHR Heat Exchanger cooling water flow and mitigate inadequate core cooling. The new 0 - 8400 gpm range of interest has no credible potential for increasing the consequences of occurrence of malfunction of equipment important to safety as previously evaluated in the FSAR.

Accurate throttling of the service water flow to the RHR System based on the new scale indications will ensure that containment or shutdown cooling will be adequately maintained. The Safety Related function of the RHR Service Water System remains unchanged.

The Technical Specification margin of safety will not be reduced by changing the range of RHR Service Water flow indication. The RHR Service Water System will perform its safety functions as previously analyzed within the margins of safety given in the technical specification bases.

CHANGE TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: Site Organizational Changes

FUNCTIONAL SUMMARY: Several site organization changes were made during 1989, including those resulting from restructuring following an Organizational Analysis. Changes included reorganizing the Operations Staff personnel, creating a Technical Assistant to the Plant General Manager position, moving design functions to the Nuclear Engineering Department, and changing titles from supervisors to managers to better describe functions.

SAFETY SUMMARY: The changes made were the result of restructuring in the CP&L organization due mainly to the Organizational Analysis that CP&L initiated in 1989. The changes were made to functionally enhance the operations of CP&L at its Nuclear Facilities. The changes do not represent an unreviewed safety question in that approval authorities for tasks related to nuclear safety have not been downgraded, nor have they resulted in a reduction in commitment. The changes will allow the plant to operate more efficiently, reliably and safely by streamlining organizations and their interface.

CHANGE TO PROCEDURES AS DESCRIBED IN THE FSAR

<u>TITLE:</u> OSP-89-013, Supplemental Fuel Pool Cooling Mode

<u>FUNCTIONAL SUMMARY:</u> This procedure is being written to provide a method for cooling the Spent Fuel Pool with the RBCCW system out of service while in Mode 4 (Cold Shutdown), using the RHR System in the Fuel Pool Cooling Assist Mode of operation.

<u>SAFETY SUMMARY:</u> This procedure provides for using the RHR system in the Fuel Pool Cooling Assist (FPCA) mode of operation while in operating conditions 4 or 5. This is provided by operating the B Loop of the RHR system in shutdown cooling and then providing an additional flow path for the Spent Fuel Pool (SFP). This procedure exists in OP-17 for Operational Condition 5.

The UFSAR specifies that the RHR system can supplement the Fuel Pool cooling and cleanup system only if the reactor is shutdown and in the refueling mode. This assures the availability of the RHR system for core standby cooling service as required. The extension of the use of the RHR system in Fuel Pool Cooling Assist is justified by the following:

> Sufficient core cooling for a loss of coolant accident is provided by having 2 Low Pressure systems available in accordance with Technical Specification 5.3.1 and 5.3.2. This provides actions to be taken if one of the systems is lost. This action is not changed by using the RHR system, which is in shutdown cooling, for FPCA. This system can be manually realigned within the time frames of the Technical Specifications if required, or actions can be taken to terminate evolutions which have a potential to drain the vessel.
>
> Another operational transient (occurrence) which could occur is a loss of shutdown cooling. The UFSAR requires that shutdown cooling be reestablished using redundant equipment, and provides methods of performing this evolution. There will be sufficient time to reestablish shutdown cooling by securing the FPCA if required.

The design and operation of the RHR system is not changed from the requirements in the UFSAR, and the system operating procedures, and therefore the probability of malfunctions or accidents are not changed. The FPCA operation is not covered in the Technical Specifications and therefore the margin of safety is not changed as long as the requirements or T/S 5.3.1 are met.

CHANGE TO PROCEDURES AS DESCRIBED IN THE FSAR

TITLE: OSP-89-014, Rev. 0; Loading of Fuel Channel Liner into the Chem. Nuclear 8-120B Cask

FUNCTIONAL SUMMARY: This procedure provides the guidance for the special process of loading a liner full of irradiated flow channels into a shipping cask. The disposal liner filled with irradiated flow channels will be remotely raised above the spent fuel pool and drained for approximately 30 minutes. The liner will then be loaded into the cask, which is staged on the east side of the pool. Access to the refuel floor will be restricted during this evolution.

SAFETY SUMMARY: Crane operator controls for this evolution have been staged on the 95' elevation with all functions which have been incorporated in the cab controls. To provide redundancy, two hoist units, main brakes, hoist motors and gear trains are supplied. The brake system consists of a disc mechanical brake, one electrical shoe brake, and a service brake on the trolley. The mechanical rigging which mates the loaded liner to the lifting hook has been proof tested to more than twice the expected load. This redundant braking system, together with the other redundant components of the crane and lifting slings ensures that dropping the loaded liner is not a credible accident.

Separate 120 volt power feeders will be installed to the main hoist lowering contact, the trolley west contact, and the main contact in the crane cab to provide a back-up system for the crane hoist and trolley operation. As a third back-up to the crane controls, an electric hoist will be in place to pull the trolley back over the spent fuel pool. The power for this hoist will be supplied from an alternate power supply. This will allow the liner to be manually lowered into the spent fuel pool with a loss of power to the crane hoist and trolley.

The 117' elevation will be a locked high radiation area with controlled access during this evolution. The liner movements, from the time it is raised to within 7 feet of the pool surface until it is lowered into the cask, will be performed remotely on the 95' elevation. Dry-runs of this process will be performed prior to the actual lift. Radiation surveys will be taken during the evolution on the other elevations in the reactor building to ensure safe access to areas required for operation of the reactor.

Worst case calculations have been performed which indicate that the reactor building ventilation isolation detector, located on the 80' west elevation of the reactor building, will not be affected by this evolution. Area radiation monitors on the refuel floor are anticipated to alarm as a result of this evolution. However, operating personnel will be kept informed of the planned evolution such that shutdown of the reactor will not be required.

The FSAR has analyzed the handling of shipping casks on the refueling floor and basis is provided to ensure that dropping of the liner is not a credible accident. The cask/liner movement will be controlled so as not to adversely affect any safety systems. The margin of safety as defined in the basis of the Technical Specifications is not reduced, as the movement of loads greater than 1600 pounds over the pool is not permitted. Contingency measures included in the procedure mitigate the possibility of an accident, thus ensuring that this evolution does not constitute an unreviewed safety question.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

<u>TITLE</u>: OSP-86-074, Rev. 1; Service Water Pump Balancing

<u>FUNCTIONAL SUMMARY:</u> The test involves the additional balancing of the Service Water Pumps to determine if the problem of high vibration can be corrected.

<u>SAFETY SUMMARY:</u> This test provides a means to attempt balancing of the Service Water Pump and Motor as a unit. There are no changes to the operation of the pump, or its controls. The test is performed to improve the reliability of the pump and motor, and the performance of this test has no adverse affect on plant safety.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: OSP-89-045, Rev. 0; Control Building HVAC Damper Test with the 2-VA-U-V18 Throttled

FUNCTIONAL SUMMARY: This test verifies the operability of the Control Building HVAC Dampers with the system in an abnormal line-up; the damper air supply valves in the throttled position.

SAFETY SUMMARY: Administrative controls as required by the Technical Specifications will be adhered to during the performance of this test so as to maintain evaluated safety margins discussed in the FSAR. Equipment important to safety as related to the performance of this procedure will be monitored in accordance with Technical Specification requirements and measures taken to return it to service within the required time constraints. The change does not affect any initiating systems involved with an assumed Control Building event which would require the initiation of the CBEAF system. The effects on the mitigating system equipment (dampers) will be controlled so as to not involve an unreviewed safety question.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 1SP-89-006, Rev. 1, Differential Pressure Testing of Valves 1-E41-F004, F041, F042, and 1-E51-F010, F029, and F031 to Meet the Requirements of IEB 85-03, Supplement 1.

FUNCTIONAL SUMMARY: Includes removal of control logic fuses and lifting of wires to allow valve operations at current plant conditions, i.e., high torus level, control logic A out of service, low steam pressure.

SAFETY SUMMARY: This SP contains provisions for the performance of this test only during modes 4 or 5, when the HPCI and RCIC systems are not required. The procedure performs diagnostic testing of the HPCI and RCIC MOVs. Testing is performed on "out of service" equipment. Operability status of equipment outside of the test boundary is not affected. No scenario is introduced that would involve an unreviewed safety question.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-89-005, Rev. 0, Unit 2 SW Vital Header Flow Verification Test.

FUNCTIONAL SUMMARY: This test is being run to obtain flow data in response to NRC Violation 50-325, 324/88-40. The test places all of the Unit 2 Service Water vital header B-Loop cooling loads in service to obtain flow data.

SAFETY SUMMARY: This test does not alter the SW system, but places it in a lineup which would be required should an accident occur that requires SW Vital Header cooling. After completion of the test, the SW Vital Header is returned to its normal (standby) lineup per operating instructions.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-89-011, Rev. 0, Alternate Drywell Leakage Calculation.

FUNCTIONAL SUMMARY: This procedure provides an alternate method of measuring drywell leakage in lieu of the installed flow measurement and integrating system only while Maintenance Surveillance Test 2-MST-LKDET-21R is in progress.

SAFETY SUMMARY: Measurement of identified and unidentified leakage is used to monitor the integrity of reactor coolant pressure boundary. The alternate method of drywell leakage measurement will adequately detect changes in leakage and will quantify the leakage for analysis of leak rates per Technical Specification requirements. No new malfunctions or accidents are created as a result of this change.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-89-016, Rev. 0, Fast Closure of the Outboard MSIV 2-B21-F028B, Following Packing Adjustment.

FUNCTIONAL SUMMARY: This procedure provides guidance to the operator to fast close outboard MSIV 2-B21-F028B at power condition in order to prove operability following maintenance on the valve packing.

SAFETY SUMMARY: The FSAR states that the MSIVS may be tested and exercised individually to the fully closed position by reducing reactor power to 70 percent full power. This special procedure includes in its p .requisites section for reactor power to be less than 70% rated. The procedure has an additional initial condition that specifies the inboard isolation valve to be closed prior to closure of the outboard valve. The valve is being closed to meet Technical Specification requirements for containment isolation. Since the "B" main steam line is already isolated, fast closure of the outboard valve will not pose any pressure or level transient on the RPV and therefore no systems will be challenged. Main Steam line break analysis remains unaffected as a result of this change. The system is being tested within its design basis, providing assurances that adequate margins of safety are maintained. Auto closure of the MSIV following an isolation signal is unaffected by this change.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-89-032, Unit 2 Vital Header Flow Test

FUNCTIONAL SUMMARY: This procedure will gather pressure and flow data for the Unit 2 SW Vital Header. The test flow paths are those allowed by the system Operating Procedure. The procedure establishes SW flow through the room coolers and seal coolers served by the Unit 2 Vital Headers. Pressure and flow data will be recorded to support a SW System Hydraulic Analysis. The following components will be affected by initiating Nuclear SW flow through them:

RHR A and C Room and Seal Coolers
RHR B and D Room and Seal Coolers
Core Spray A and B Room Coolers

Flow will be established by opening the circuit breakers that supply the solenoid valves which control the air-operated service water supply valves on the subject components.

SAFETY SUMMARY: The SW supply valves are designed to fail in the open position (safe position). The open condition allows service water flow in the event of an accident. The vital headers thus remain operable.

Initial flow and pressure data will first be taken with no flow to provide baseline data about the idle condition. Next, flow will be established through the vital headers. Then pressure and flow data will be taken with one NSW pump in operation. A second NSW pump will be started and additional data taken. The flow paths are those designed to be used in accident conditions. No hydraulic jumpers or unanalyzed lineups will be established.

No equipment modifications will be made which could affect the design of the pumps. The NSW pumps will be operated at flows in excess of 2500 gpm/pump, even with 2 pumps in operation. The vendor has formally stated that the pumps are reliable at minimum flow rates as low as 2000 gpm/pump.

Procedural controls will be in place to ensure that the system will be maintained in a lineup consistent with its accident operational requirements. No pressure boundaries will be affected. Pressure data will be obtained from installed instrumentation. Technical Specification requirements for operable SW pumps will be maintained for the duration of the test. Design margins will be maintained. The accident condition heat removal capability will not be degraded by this change.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-89-041, Rev. 0, Cleaning the Reactor Pressure Vessel Bottom Head Drain Line

FUNCTIONAL SUMMARY: This is a new procedure to clean out flow restrictions in the bottom head drain line to meet GE SIL 251, and reduce RPV differential temperature indications. Work activities will open the reactor coolant pressure boundary after the reactor fuel has been removed from the reactor vessel and placed in the Spent Fuel Pool. Access of the RPV bottom head drain line will be made by installing a temporary isolation valve at the first tee downstream of the reactor vessel and "hot tapping" on the back side of the tee. This method reduces radioactive leakage collection significantly. The line will be visually inspected and cleaned using special tools designed and fabricated by Power Cutting Incorporated, which use high pressure water jets to fracture and flush the restricting debris to the drywell sump. Upon flush completion a final inspection will be performed inside the pipe, then the line will be freeze plugged and the access modified tee replaced to restore the line to its original configuration in accordance with Specification 248-117. The new welds will be shop tested, as applicable, prior to start-up.

SAFETY SUMMARY: This procedure will be performed while reactor fuel has been removed from the RPV and is in storage in the Spent Fuel Pool. The bottom head drain/Reactor Water Cleanup System will be out of service during work activities. This work will not create any new concerns for previously evaluated accident consequences. The cleaning will improve temperature differentials in the bottom of the core, thus improving plant reliability. Some of the contingency activities associated with this test have been evaluated for possibility of creating malfunctions/accidents not previously evaluated in the FSAR. These events have been evaluated and found to present no nuclear safety concerns beyond those previously evaluated in the FSAR. They are listed below:

1. During high pressure cleaning the "mole" and water lance becomes bound inside the pipe by debris and cannot be removed from an inaccessible section of the piping.

2. Following high pressure cleaning, the post work fiberscope visual examination identifies a large metallic object (nut, washer, etc.) requiring removal.

3. Following high pressure cleaning, minimum wall thickness of the pipe is exceeded in an inaccessible area undervessel (thru wall leak), requiring a pipe repair.

Temporary access fittings installed as a result of this procedure will be removed, and the system restored to its original condition prior to work completion and startup.

TEST OR EXPERIMENT NOT DESCRIBED IN THE FSAR

TITLE: 2SP-89-042, Rev. 0, Differential Testing of HPCI and RCIC Valves to Meet Requirements of Generic Letter 89-10.

FUNCTIONAL SUMMARY: Diagnostic testing of the following motor operated valves in the HPCI and RCIC systems at differential pressure conditions will be performed in modes 4 or 5 to meet the requirements of Generic Letter 89-10:

HPCI: 2-E41-F004, F041, and F042

RCIC: 2-E51-F010, F029, and F031

SAFETY SUMMARY: This procedure only affects the involved systems during modes 4 and 5, when the systems are not required to be operational. As such, no new malfunctions/accidents are introduced. Provisions have been made to ensure that the systems are returned to operable status prior to start-up. Performance of this procedure will increase reliability of the involved components by ensuring proper operation during expected conditions.