SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS

FOR THE HOPE CREEK GENERATING STATION

JANUARY AND FEBRUARY 1992

The following items have been evaluated to determine:

1. If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

2. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

3. If the margin of safety as defined in the basis for any technical specification is reduced.

The 10CFR50.59 Safety Evaluations showed that these items did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These items did not change the plant effluent releases and did not alter the existing environmental impact. The 10CFR50.59 Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

DCP	Description of Safety Evaluation
4EC-3321/01	This DCP revised the instrument setpoint for temperature switches that isolate the Class 1E ventilation dampers for and supply and return ducts for the steam tunnel. The setpoint is being raised as a result of a revision to the setpoint calculation. No Unreviewed Safety Questions were involved because the setpoint change re-establishes the design basis setpoint. The setpoint change does not affect the function of the ventilation dampers and the actual setpoint is not discussed in the UFSAR.
4HC-0290/03	This DCP installed redundant recording, display, and annunciation equipment for the thermal monitor display system. This equipment provides input for daily and monthly reporting and continuously monitors effluent flow, influent temperature, effluent temperature, and net heat rate. This equipment complies with NJPDES regulations. The thermal monitor display system is non-safety related and does not interface with any systems that are required for accident mitigation and does not affect the operation of any safety-related component or system. Therefore, no Unreviewed Safety Questions were involved.

DR	Description of Deficiency Report
HTE 91-177	This DR identified potential leakage paths through gaps or separations that exist between rectangular floor penetration steel liners and the concrete floor at various locations. The separation depth is indeterminate and may not provide a watertight barrier as designed. This DR has been dispositioned "Repair". This disposition does not result in any hardware or logic changes and, therefore, does not involve an Unreviewed Safety Question.
HMD 91-194	This DR addressed a packing leak in an isolation valve in the Main Steam System Drain Header. The valve was closed to prevent steam from escaping through the packing and to reduce the potential for a high temperature condition in the steam tunnel from the packing leak. The Loss of Power Accident, the Loss of Coolant Accident, and steam header drain control during startup, shutdown, and transient conditions were evaluated. This valve can be manually open if required and is already closed for operational conditions that require it to close. Because there is no increased probability or consequences of an accident or equipment malfunction, this DR disposition does not involve an Unreviewed Safety Question.
HMD 91-199	This DR addresses a blown bellows on one of the Steam Seal Evaporator Relief Valves, causing the valve to lift below its setpoint. This valve was gagged closed and a pressure control valve was tagged open to allow the Steam Seal Evaporator to maintain overpressure protection. This disposition does not operate the system outside of its original design conditions. Additionally, failure of the Steam Seal System does not compromise any safety-related system or component or prevent the safe shutdown of the plant. Therefore, this DR disposition does not involve an Unreviewed Safety Question.

DR	Description of Deficiency Report
HTE 91-202	This DR addresses the new Maximum Critical Load and the new Design Rated Load of the Polar Crane following the load test of 134.85 tons. The Safety Evaluation associated with this DR also addresses lifting loads that exceed the new Maximum Critical Load and the new Design Rated Load. The load test was required because a new shaft was installed in the eddy current brake. This does not affect the single failure design of the polar crane. The mechanical brake was not affected by the maintenance and is still rated for 150% of the original rated full load. The load test was limited to 134.85 tons floor loading concerns, not concerns related to the capability of the crane. Since the crane was originally designed to handle loads of up to 150 tons, a test lift of 135 tons has been performed, and the only change that has been made is the replacement of the eddy current brake that did not affect the single failure proof design it is conservative to assume that the crane has the capability to lift 112 tons safely. Therefore, no Unreviewed Safety Questions are involved.
HTE 92-004	This DR disposition allows differential pressure switches on the 'A' and 'B' Hydrogen/Oxygen Analyzer System to be used-as-is. The installed switches were not designed to assure response to negative differential pressure; however, they were designed to the same proof pressures as the correct switches. The Hydrogen/Oxygen Analyzers will continue to perform their intended function. Some temporary procedure changes were implemented as additional conservative compensatory measures. Therefore, this DR disposition does not involve an Unreviewed Safety Question.

DR	Description of Deficiency Report
HMD 92-009	This DR addresses a through-wall leak on a Station Service Water instrument line upstream of a 'B' Safety Auxiliaries Cooling System Heat Exchanger. This may be used-as-is because the area that is below minimum wall thickness is less than 0.5 inches in diameter. This DR does not involve any Unreviewed Safety Questions because the analysis of the flaw indicates that the line will maintain its integrity during a design basis earthquake. Additionally, equipment that is required to operate during or mitigate an accident and can be affected by water spray from this leak is either protected from water spray or designed to operate when wet.
HTE 92-040	This DR addresses a below minimum wall thickness reading on a Station Service Water root valve line upstream of an 'A' Safety Auxiliaries Cooling System Heat Exchanger. This DR does not involve any Unreviewed Safety Questions because the analysis of the flaw indicates that the line will maintain its integrity during a design basis earthquake. Additionally, the UFSAR analysis covers loss of safety related equipment in this area due to flooding or water spray. The pipe is planned to be restored to ASME Specification during the fourth refueling outage.
HTE 92-041	This DR addresses a below minimum wall thickness reading on a Station Service Water root valve line upstream of a 'B' Safety Auxiliaries Cooling System Heat Exchanger. This DR does not involve any Unreviewed Safety Questions because the analysis of the flaw indicates that the line will maintain its integrity during a design basis earthquake. Additionally, the UFSAR analysis covers loss of safety related equipment in this area due to flooding or water spray. The pipe is planned to be restored to ASME Specification during the fourth refueling outage.

edure ision	Description of Safety Evaluation
.NA-AP.ZZ-0008(Q) ev 2	This revision to the administrative procedure controlling design and configuration changes, test and experiments re-assigned the responsibility for document update from the Vice-President Nuclear Engineering to the General Manager - Engineering and Plant Betterment. This change does not affect any accidents or system operations; therefore, no Unreviewed Safety Questions are involved.
THC.MD-GP.KE-0001(Q) Rev 0	This temporary procedure provides guidance for the removal of a test weight from the Spent Fuel Pool. It administratively controls the use of the Auxiliary Monorail Hoist while its load setting is increased to remove the test weight from the Spent Fuel Pool. The load will not be handled over spent fuel and the Auxiliary Monorail Hoist meets the requirements for a single failure proof system. Therefore, no Unreviewed Safety Questions are involved.

Procedure Revision	Description of Safety Evaluation
NC.NA-AP.ZZ-0008(Q) Rev 2	This revision to the administrative procedure controlling design and configuration changes, test and experiments re-assigned the responsibility for document update from the Vice-President Nuclear Engineering to the General Manager - Engineering and Plant Betterment. This change does not affect any accidents or system operations; therefore, no Unreviewed Safety Questions are involved.
THC.MD-GP.KE-0001(Q) Rev 0	This temporary procedure provides guidance for the removal of a test weight from the Spent Fuel Pool. It administratively controls the use of the Auxiliary Monorail Hoist while its load setting is increased to remove the test weight from the Spent Fuel Pool. The load will not be handled over spent fuel and the Auxiliary Monorail Hoist meets the requirements for a single failure proof system. Therefore, no Unreviewed Safety Questions are involved.

UFSAR Section	Description of Safety Evaluation
6.2.4.4.3	The UFSAR stated that when leak rate testing is performed between the Main Steam Isolation Valve and the Main Steam Stop Valve, total observed leakage through both valves is assigned to that penetration. This UFSAR change assigns total observed leakage through the outboard Main Steam Isolation Valve only to that penetration when leak rate testing is performed between the Main Steam Isolation Valve and the Main Steam Stop Valve. The specified methodologies and limitations associated with leak rate testing the Main Steam Isolation Valves will not be affected by this change. There are no Unreviewed Safety Questions associated with this change because the Main Steam Isolation Valve leak rate testing methods are not affected by this proposal. Also, the function and capabilities of the Main Steam Stop Valves are not affected by this proposal. They continue to be tested in accordance with the Inservice Test Program and ASME Section XI.

Description of Safety Evaluation

H03.5-111

During Cycle-4, segments in nine control rods will exceed 34% Boron-10 depletion, which is their normal design life and the basis used for the Cycle-4 licensing analysis.

This Safety Evaluation shows that the neither the shutdown margin nor any other licensing analyses would be significantly impacted at up to 50% boron depletion. Therefore, no Unreviewed Safety Questions are involved.