ENCLOSURE

10 CFR 50.59 REPORT OF FACILITY CHANGES, PROCEDURE CHANGES, TESTS, AND EXPERIMENTS OCTOBER 28, 1994 - MAY 24, 1996

Pacific Gas and Electric Company
Diablo Canyon Power Plant, Units 1 and 2
Docket Nos. 50-275 and 50-323

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* These changes were completed in previous reporting periods but were not included in earlier 10 CFR 50.59 Reports. They are included in this report to update the reporting record.

SUMMARY OF 10 CFR 50.59 CHANGES FOR THE REPORT PERIOD OCTOBER 28, 1994 - MAY 24, 1996

A. Facility Changes

1. Install a Mechanical Seal Clamp Assembly on the Spare Reactor Vessel Head Penetrations
   DCP M-041295 Rev. 1 (Unit 1)

   The canopy seal weld on the reactor vessel head penetration adapter has been the location of reactor coolant system (RCS) leakage. This design change allows installation of the mechanical seal clamp assemblies (MSCAs) over the reactor vessel (RV) head penetration adapter-to-cap canopy seal. This reduces the leakage from the RV head adapter.

   Safety Evaluation Summary

   These clamps are designed to the requirements of ASME Section III, 1983 Edition with 1984 Addenda, Section NB, Class 1 Nuclear Components, and meet the material interfacing requirements for the vessel head. They are installed in accordance with the Diablo Canyon Power Plant (DCPP) ASME Section XI program plan and in accordance with approved plant maintenance procedures. This ensures that the installation meets the applicable requirements of the licensing basis of the plant.

   These clamps are seismically qualified to DCPP-specific seismic spectra and transient loading requirements. The clamp assemblies have been reviewed and do not compromise the integrity of the RCS. The MSCAs are installed on the outside of the RV head adapter so that they do not form part of the RCS pressure boundary.

   The addition of the MSCAs functions only to lower the leakage out of a RV head adapter. The design provides additional reactor coolant leak protection from the RV head. Any leakage from the clamp is classified as "unidentified leakage" to which the 1 gpm limit required by the Technical Specifications (TS) applies instead of the zero gpm leakage associated with the RCS pressure boundary. Any margins of safety implicit in the bases for the associated TS are therefore maintained.

2. Adjust the Overspeed Trip Setpoint for the Auxiliary Feedwater Turbine
DCP M-043069 Rev. 0 (Unit 1)
DCP M-044069 Rev. 0 (Unit 2)

This design change revises the auxiliary feedwater turbine (AFW) overspeed trip setpoints. The new setpoints are calculated and selected to ensure that pressure limitations of the piping system are not exceeded. The operating speed of the turbine is also limited to ensure reliable operation while still achieving the design flow.

Safety Evaluation Summary

This design change involves a setpoint change and does not require any physical modification to the plant. The modification of the AFW turbine overspeed trip setpoints does not change the operation or bases for the equipment. Additionally, the increase in the overspeed trip setpoint decreases the probability of a spurious trip on a quick start.

The range of normal operating speeds is selected to ensure design flow requirements are met while preventing overpressurization. Hence, the margin of safety as defined in the bases for the TS is not reduced because the new overspeed trip setpoint does not affect the ability of the pump to deliver design flow to the steam generators.

3. Deletion of Moisture Separators and HEPA Filter from Containment Fan Coolers
DCP H-043663 Rev. 1 (Unit 1)
DCP H-044663 Rev. 1 (Unit 2)

The moisture separators and HEPA filters were removed to decrease the maintenance requirements for the containment fan cooler units (CFCUs). To maintain the required flowrate through the CFCUs, the dampers of the CFCUs were modified to produce the same pressure drop across the CFCU during the accident mode as it was before the modification. In addition, the drain lines from the moisture separator, HEPA filters, and the charcoal filters were plugged since they were no longer needed.

Safety Evaluation Summary

The CFCUs were initially designed with moisture separators and HEPA filters. The function of the moisture separators and HEPA filters as described in the FSAR Update was to reduce airborne particulate fission particles following a loss-of-coolant accident (LOCA). However, the DCPP accident analysis did not take credit for this filtration process and, therefore, removal of the filters did not affect FSAR Update accident consequences.

The design change was made in such a way so as not to affect the heat removal capability of the CFCUs, nor increase the heat load to the component cooling water (CCW) system. In addition, it was determined that removal of the moisture separators and HEPA filters did not affect the seismic qualification of the CFCUs.

4. <u>Replace Motor Operators on Certain Valves from Rotork Motor Operators to Limitorque Motor Operators</u>
DCP J-047195 Rev. 1 (Unit 1)
DCP J-048195 Rev. 0 (Unit 2)

This design change specifies a different manufacturer to procure spare parts. Certain valves in the auxiliary saltwater (ASW) system and CCW system which had Rotork motor operators will be replaced with Limitorque motor operators since spare parts are no longer available from Rotork. Some valves will be changed to manual valves since their design purpose was to isolate pipes for maintenance purposes.

For a majority of the valves (those that are covered by TS or other design requirements), the design requirements such as stroke times are still maintained. In some cases, the previously specified stroke times are changed. However, these valves are not covered by any TS requirements, and the new stroke times have been evaluated to meet the system design requirements.

<u>Safety Evaluation Summary</u>

This change involves specifying a different model and manufacturer for certain motor-operated valves. The TS requirements for valve stroke times and other design parameters are not changed. The change was made to facilitate procurement efforts.

Some valves that were provided for maintenance purposes were changed to manual valves. In some cases, stroke times for certain valves were changed. These stroke times do not affect the TS requirements and have been evaluated to ensure that the valves continue to meet their intended design function.

In summary, the valves perform their intended design functions of opening, closing, and maintaining the capability to provide the design flow. Thus, this change does not involve a reduction in the margin of safety as defined in the bases for any TS. The change does not have any effect on the results of previously analyzed accidents or contribute to malfunction of equipment important to safety.

5. Revise Setpoints for the Relief Valves on the Hydrogen System Supply Header
DCP M-047453 Rev. 1 (Unit 1)
DCP M-048453 Rev. 1 (Unit 2)

This design change revises the setpoints provided in the FSAR Update for the relief valves on the hydrogen system supply header. The design data provided in this DCP have been established from existing design documentation. This is a document change only and no component is physically added to the plant. The document change does not alter the operation of any systems.

Safety Evaluation Summary

The setpoint change required by this design change results in a more conservative system design and increases the level of overpressure protection the affected relief valves are intended to provide for the hydrogen supply header piping. This design change does not degrade the performance of or increase the challenges to any equipment important to safety. No new failure modes are created by the revised design.

The hydrogen system affected by this design change is not safety-related and is not required for the safe shutdown of the plant. Since the design change does not compromise the operability or reduce the reliability of the hydrogen system, there is no impact on the margin of safety as defined in the basis for any TS.

6. Replace Existing Refueling Cavity Seal with a Seal of a New Design
DCP N-047861 Rev. 0 (Units 1 & 2)

This design change provides a new refueling cavity seal design to seal the gap between the reactor vessel flange and the refueling cavity floor for floodup of the refueling cavity during refueling outages. The new seal design is more quickly installed, provides a more positive seal, and extends less above the floor making it less susceptible to damage than the existing seal.

Safety Evaluation Summary

The new refueling cavity seal performs the same function as the existing seal, is only used during refueling outages when the reactor is shutdown, and is totally removed prior to plant operation, which precludes the cavity seal from becoming a missile. The hardware for the new seal is different than the existing seal, but it provides a reliable, positive seal. The new seal has a lower profile on the cavity floor and is considered to be less susceptible to damage than the existing seal.

The only credible failure would be for some leakage to develop for either type seal. Hence, the likelihood of any malfunction or failure of the new seal is

considered no greater than that for the existing seal, and the consequences of the failure are not changed by the new seal design.

Since there is no TS that directly relates to the refueling cavity seal, there is no impact on the margin of safety.

7. Replace 4-kV Bus Breakers
DCP E-048961 Rev. 0 (Unit 2)

The existing 4-kV Class 1E breakers have marginal capacity to interrupt potential short circuits. To provide the capability for these breakers to operate during worst-case short circuit conditions with margin, the 250-mVA breakers were replaced with 350-mVA rated breakers. These breakers were changed from air-magnetic to SF6 gas-filled interrupters.

Safety Evaluation Summary

There were no operational changes made to the function of these breakers. Wiring changes occurred to provide additional alarms and local indicating lights. These changes do not impact accident analysis, consequences, or create new potential accidents. As the short circuit capability of these breakers is increased, the related margin is increased.

8. Replace Existing Thermo-Lag Fire Barriers
DCP A-049070 Rev. 0 (Unit 1)
DCP A-050070 Rev. 0 (Unit 2)

This DCP replaced the existing Thermo-Lag fire barriers with qualified fire barrier materials. The NRC had identified a generic issue associated with the ability of Thermo-Lag material to provide the rated fire barrier protection in accordance with 10 CFR 50, Appendix R. Therefore, Thermo-Lag barriers were replaced.

This DCP replaced the existing unqualified Thermo-Lag material on electrical conduits and junction boxes with a qualified 3M fireproofing wrap system. In addition, for certain raceways, the Thermo-Lag material was removed and did not have to be replaced on the basis of a safe shutdown analysis.

Safety Evaluation Summary

The removal of Thermo-Lag fire barrier and its replacement with a qualified 3M material does not increase the probability of occurrence of a fire or any other accident. Since no new fire hazards are added, the consequences of a fire remain the same. Replacement of Thermo-Lag with 3M does not affect the

probability of occurrence of malfunction of the affected circuits because the 3M material provides qualified fire protection capability.

The consequence of a malfunction of equipment has not increased since the existing material was replaced with qualified material. For areas where the Thermo-Lag was removed and not replaced, analysis demonstrated that the ability to safely shut down the plant is not affected. Similarly, there is no possibility of creating an accident of a different type, nor is there a possibility of creating a different type of equipment malfunction.

Since Thermo-Lag barriers are not addressed in the TS, and replacement of the Thermo-Lag barriers was done with qualified fire protection materials, there is no reduction in the margin of safety as defined in the basis for any TS.

9. <u>Install Secondary Sample Taps</u>
DCP J-049089 Rev. 0 (Unit 1)
DCP J-050089 Rev. 0 (Unit 2)

Technical Specification 6.8.4c requires that steam generator degradation be monitored. The existing sample system had deficiencies associated with analysis of corrosion product transport due to corrosion product plateout. This design change installed four new local sample taps, which allows for a more representative sample. In addition, the design change installed the appropriate instruments to provide the capability of local on-line sampling at the new sample taps.

<u>Safety Evaluation Summary</u>

Since the sample taps do not perform any safety-related function or interface with any system required for safe shutdown or accident mitigation, it was concluded that no unreviewed safety question existed. In addition, the new sample points provide a more reliable means of analyzing for corrosion product transport and better samples for analyses.

10. <u>Replace Vital Battery(s)</u>
DCP E-049099 Rev. 0 (Unit 1)
DCP E-050099 Rev. 1 (Unit 2)

With the replacement of vital instrument inverters 1Y11 (2Y22) and 1Y11A (2Y24) with a larger capacity vital instrument uninterruptible power supply, the remaining margin on Battery 11 (22) had been consumed. This modification replaced Battery 11 and 22 LC-25 cells with larger LCUN-33 cells. This eliminated the margin limitation in Battery 11 and 22 and provided an allowance for future growth.

Safety Evaluation Summary

Replacing the 1,800 ampere-hour battery with a 2,320 ampere-hour battery does not impact accidents analysis, consequences, or create potential accidents. As the battery ampere-hours rating is increased, the margin of safety is increased.

11. Replace Containment Pressure Transmitters, PT-934, PT-935, PT-936, and PT-937
DCP J-049102 Rev. 1 (Unit 1)
DCP J-050102 Rev. 1 (Unit 2)

It was found that although the existing Barton containment pressure transmitters were environmentally qualified for a LOCA, they were not environmentally qualified for a main steam line break (MSLB) outside of containment. During a MSLB, the Bartons may have had a common mode failure and caused a spurious Phase B actuation. The DCPs replace the Barton transmitters with Rosemount 1154 transmitters that are qualified for the LOCA and MSLB.

Safety Evaluation Summary

FSAR Update Section 3.10 contained a detailed discussion of the seismic qualification of the Barton transmitters. Based on the replacement Rosemount transmitters meeting the seismic, application, and environmental qualification requirements for the installed locations, it was concluded that no unreviewed safety question existed for these replacements.

The seismic qualification was based on the shake table test performed on Rosemount 1153 transmitters documented in Wyle Report No. 45592-3. The 1153 transmitters are structurally identical to the 1154 transmitters and the seismic spectra used in the test envelopes the DCPP seismic requirements.

The Rosemount transmitters meet the range and accuracy requirements for the application. The Rosemount transmitters and their electrical connections are qualified for all postulated accident conditions for the installed locations.

12. Refueling Water Purification System Piping Modifications
DCP N-049115 Rev. 0 (Unit 1)

This design change involved installation of a cross-tie downstream of the spent fuel pool (SFP) pumps that connects the line to the liquid holdup tanks (LHUTs) at a location upstream and downstream of the refueling water purification (RWP) pumps. This allowed use of the RWP pump to send reactor cavity/refueling canal water to the LHUTs and allowed SFP cleanup water to pass through the RWP filter prior to going through the SFP demineralizer. This involved a change

to the system description and method of operation for these systems as described in the FSAR Update.

Safety Evaluation Summary

The failure of the added Class II piping will not impact the integrity of the Design Class I piping due to the code break provided between the seismically qualified piping and the nonseismically qualified piping. Utilization of the new purification flowpath replaces the one previously used, while maintaining the same failure modes and failure probabilities. Since a piping failure has already been evaluated in the FSAR Update and passive design features protect the SFP and SFP cooling system, no unreviewed safety question was found to exist.

13. Diesel Generator Cooling Air Flow Improvement
DCP H-049117 Rev. 1 (Unit 1)
DCP H-050117 Rev. 1 (Unit 2)
DCP H-050203 Rev. 0 (Unit 2)

The emergency diesel generators (EDGs) were designed to operate in an outside ambient temperature of 90°F, at a load of 115 percent full load, and with the jacket water temperature maintained below 180°F. To operate at the above conditions, the diesel generator fan was designed to produce 140,000 cfm flow. However, flow measurements showed that the fan air flows were below the design flow rate and the fans were pulsating because they were operating in an unstable region of the fan curve.

These design changes modified the EDG fan air flow by adding straighteners and rerouting the duct work. In addition, new louvered openings were added to the 4-kV switchgear ventilation system supply air intakes. As part of these modifications, new fire walls were added in place of walls removed, which affected Fire Hazard/Appendix R Evaluation 99 and 103.

Safety Evaluation Summary

These design changes increased the radiator cooling fan air flow and restored margin for operation of the EDGs, thus increasing EDG reliability. The modified walls were evaluated and shown not to adversely impact the DCPP Fire Protection Program. All new structures, including the air plenum, flow straighteners, drywall partitions, and HVAC ducts were designed in accordance with the appropriate design requirements and criteria, including seismic, tornado, and wind requirements. In addition, the new 4-kV ventilation system flow path provides the same quantity of air to the 4-kV switchgear rooms without affecting fan operation or room temperature.

14. <u>Rerate the Residual Heat Removal Heat Exchangers and Pumps to a Higher Design Pressure</u>
DCP N-049118 Rev. 0 (Units 1 & 2)

The residual heat removal (RHR) system pressure is determined by the RCS pressure at the RHR pump suction plus the RHR pump differential pressure. Throttling the RHR pump flow increases the pump discharge head. When this pressure increase was added to the maximum pressure at the pump suction, it caused the pressure at the RHR heat exchangers and RHR pumps to exceed their design pressures. This change rerates the design pressure of the RHR heat exchangers and the RHR pump to a higher design pressure and provides for flexibility in operating the RHR system while meeting the Code requirement for design pressure.

<u>Safety Evaluation Summary</u>

The design pressures of the RHR pumps and RHR heat exchangers are increased by refining the required stress analysis to accommodate the possible increase in maximum operating pressure that can occur by throttling of the RHR pumps. The equipment still complies with the applicable Code requirements by reducing certain conservatisms used in the original analysis and using the material properties of existing equipment.

This is a document change only. Since there is no physical change to either the RHR pumps or the RHR heat exchangers, there is no increase in the probability or consequences of any FSAR Update evaluated malfunctions. The system will continue to be operated within design pressures.

Since there is no physical work associated with this DCP, no different types of accidents or malfunctions are created by this change, and the change has no impact on the margin of safety as defined in the basis for any TS.

15. <u>Remove Oil Storage Tanks from Solid Radwaste Storage Facility</u>
DCP M-049119 Rev. 0 (Common)

Eight lube oil storage tanks were located in Bay 5 of the solid radwaste storage facility as a holding station for contaminated oil. The tanks were not used and were occupying space that could be used for the storage of mixed waste.

This design change removed the tanks to make space available for the storage of mixed waste. The design change affected an FSAR Update system description and figure.

Safety Evaluation Summary

The tanks in question were Design Class III and were empty when removed. Since the lube oil storage tanks had no safety-related function or interface with safety-related systems, it was concluded that removal of the tanks did not constitute an unreviewed safety question.

16. Containment Penetration Overcurrent Protection for Seismic Monitoring System
DCP E-049134 Rev. 0 (Unit 1)

The containment electrical penetration conductors associated with the 12-volt power feed to the seismic accelerometers in containment were found to not have overcurrent protection. The DCP installed overcurrent and backup overcurrent protection for the containment electrical penetration conductors to the seismic accelerometers. FSAR Update, Table 8.3-17, contains a tabulation of containment electrical penetrations that have overcurrent and backup overcurrent protection. Although, the seismic monitoring system is Design Class II, the design change resulted in the addition of the new overcurrent protection to the FSAR Update table.

Safety Evaluation Summary

Adding the overcurrent protection did not change the function or operation of the seismic monitoring system; it merely provided the standard overcurrent protection scheme used at DCPP.

17. Diesel Fuel Oil Pump Vault Modifications
DCP C-049147 Rev. 1 (Common)

The California Underground Storage Tank Regulations (Title 23, Division 3, Chapter 16 of the California Code of Regulations) require release detection monitoring of piping that is used for the transport of vehicle fuel to ensure protection of the environment.

This design change addresses the monitoring requirements for the diesel fuel oil (DFO) piping located in the DFO pump vault. It provides for leak monitoring of the piping in the DFO pump vaults by modifying the existing drains, crediting the pump vault structures as secondary containment, and installing leak detectors. In addition, this design change also provides for replacing the existing gate-type backflow valves in the drains with a super-flow strainer with a back-water valve.

Safety Evaluation Summary

This modification does not affect the accident analyses addressed in the FSAR Update. The equipment changed is nonsafety-related and is not relied upon for mitigation of design basis accidents, nor does the equipment cause or contribute to the occurrence of any design basis accidents. Therefore, the replacement or enhancement of the existing components has no effect either on the occurrence of an accident or the malfunction of equipment.

The new leak detectors provide for a faster activation/alarm time that will minimize the impact of water/oil accumulation in the vault. The detection of water or oil in the DFO pump vaults is not part of the basis for any TS. Hence, there is no impact on the margin of safety.

18. Document Failed Core Exit Thermocouples as Abandoned
DCP J-049154 Rev. 0 (Common)

The FSAR Update, Section 7.5.1.2.2, and the Emergency Plan gave an exact number of installed core exit thermocouples. The DCP revised the FSAR Update to state that the number of operable thermocouples required per core quadrant is governed by the requirements in the TS. The instrument schematics were also revised to identify the thermocouples that were not functional and abandoned in place.

Safety Evaluation Summary

Clarifying the FSAR Update by identifying that the Technical Specification was the governing document for the core exit thermocouples was found not to create a unreviewed safety question. The TS requirements ensured the availability of the minimum number of core exit thermocouples necessary to assess core cooling following an accident. The logic and function of the temperature monitoring system were unaffected by the design change.

19. Turbine Building Residing
DCP A-049161 Rev. 0 (Unit 1)
DCP A-050161 Rev. 0 (Unit 2)
DCP C-049162 Rev. 0 (Unit 1)
DCP C-050162 Rev. 0 (Unit 2)

These DCPs resided the Units 1 and 2 turbine buildings with new siding placed over the old siding. Corrosion of the old siding necessitated that a new second layer of metal siding be placed over the existing siding to restore the integrity of the turbine building siding system.

The DCP work included (1) addition of a new metal siding layer over the existing one, (2) miscellaneous additional structural steel modifications to support the siding, and (3) additional louvers and supporting frame.

Safety Evaluation Summary

The turbine building siding and associated structural members do not cause or contribute to the occurrence of an accident. The consequences of any tornado or seismic induced failure have not increased since the equipment arrangement of the building has not changed, and the seismic qualification of the building has been maintained in accordance with the existing commitments specified in the FSAR Update.

Additional vent area in the form of louvers through the siding was provided in the turbine building to maintain compartment peak pressure below design basis limits following an MSLB. Also, an evaluation of turbine building block walls was performed that verified the continued structural integrity of the walls.

The new siding cannot cause the malfunction of equipment, nor can the modification create a situation that could result in a different type of malfunction or accident.

There is no reduction in the safety margin since the turbine building siding is not part of any TS.

20. Addition of Battery-Operated Lights
DCP E-049199 Rev. 0 (Units 1 & 2)

As a result of the NRC's disapproval of DCPP's request to use flashlights in access/egress areas, this modification added 23 battery-operated lights (BOLs) with an 8-hour capacity at various locations in Unit 1 and 2 to ensure strict compliance with 10 CFR 50, Appendix R, Section III.J, Emergency Lighting.

Safety Evaluation Summary

Installation of additional seismically qualified, emergency operating BOLs does not impact accident analyses, consequences, or create new potential accidents to be considered. The BOLs are not governed by TS.

21. Reactor Coolant Pump Bus Undervoltage Time Delay Addition
DCP E-049200 Rev. 2 (Unit 1)
DCP E-050200 Rev. 1 (Unit 2)

A reactor trip is initiated when a decrease in voltage to the 12-kV reactor coolant pump (RCP) bus occurs. These undervoltage relays have a time delay feature

that was set to zero. On December 14, 1994, a 500-kV system disturbance caused a momentary voltage dip at the 12-kV buses and initiated an undervoltage dual unit reactor trip. Such an incident also occurred in 1987.

It was determined that a time delay on the undervoltage relay of 0.5 seconds would have prevented such undervoltage reactor trips. Therefore, the time delay on the 12-kV bus undervoltage relays was changed from zero to 0.5 seconds.

Safety Evaluation Summary

These undervoltage relays are used to generate a reactor trip to mitigate a loss of RCS flow event. The total reactor trip time for RCP bus undervoltage as established in the accident analysis is not exceeded by the addition of a time delay in the undervoltage reactor trip. The TS Bases, Section 2.2, limits the time delay for the undervoltage trip signal to reach the reactor trip breakers to 0.9 seconds. The TS Table 3.3-2 limits the total delay for undervoltage reactor trip to 1.2 seconds.

By adding a time delay of 0.5 seconds (including tolerance, a maximum of 0.6 seconds), the time for the undervoltage trip signal to reach the reactor trip breakers is still within 0.9 seconds, and the total delay for the undervoltage reactor trip is still within 1.2 seconds. Therefore, the consequences of an accident previously evaluated are not changed and there is no reduction in the margin of safety as defined in the basis for any TS.

22. Convert the Mechanical Seals on Centrifugal Charging Pump 1-2 from First Generation to Third Generation Mechanical Seals
DCP N-049201 Rev. 1 (Unit 1)

The Centrifugal Charging Pump 1-2 casing was replaced with a stainless steel casing as a prudent measure to prevent deterioration of the casing. This replacement stainless steel casing has been converted to accommodate the third generation mechanical seals.

This change allows the use of the third generation seal that is an improved design that provides for a longer service life. Since the new seal does not require CCW for cooling, this design change also allows capping of CCW lines that are used to provide cooling water to the pump seals and the removal of the seal coolers from the pump skid.

Safety Evaluation Summary

The new third generation mechanical seal meets or exceeds the original mechanical seal requirement; however, external cooling is not required.

Capping of the CCW piping has negligible effect on the CCW system since the flow rate through the seal coolers is very small compared to the total CCW system flow. Although the heat load from the seal coolers is very low, a small margin is gained from eliminating the cooling water requirement for the seals. Seal life is extended as a result of the one-piece seal sleeve/pumping ring design. Therefore, the availability of the centrifugal pump is increased. Also, because CCW is no longer required for cooling, one of the failure modes that can cause unavailability of the charging pump is eliminated.

This design change is an improvement to the existing component. These modifications do not affect the charging pump and CCW system function, performance, or operability. Also, no new equipment is added and no new failure mode has been introduced.

23. <u>Change DEH P2000 Load Drop Anticipate Reset Time Delay</u>
DCP J-049206 Rev. 0 (Unit 1)
DCP J-050206 Rev. 0 (Unit 2)

This design changed the load drop anticipate (LDA) reset from 10 seconds to 2 seconds. The LDA provides anticipatory action during a loss of load. When the generator breaker opens, the overspeed protection control (OPC) LDA circuit quickly closes the governor and intercept valves in anticipation of the turbine overspeed.

The purpose of the LDA is to hold the governor and intercept valves closed until the turbine speed gets above 103 percent. With the 10-second time delay, the governor and intercept valves were held closed too long causing a RCP trip on underfrequency, with a subsequent reactor trip/turbine trip.

Per Westinghouse, a reset time delay of 2 seconds is sufficient for the anticipatory valve closure because after this time the OPC-103 percent function will provide the necessary turbine overspeed protection without causing RCP trips on underfrequency.

<u>Safety Evaluation Summary</u>

The LDA feature is part of the OPC for the DEH P2000 main turbine control computer. The DEH P2000 is nonsafety-related, Design Class II, Quality Class B. The LDA actuates on a loss of load. Once the LDA is set, the governor and intercept valves are rapidly closed and held closed until the LDA resets.

With a 10-second delay, the LDA held the governor and intercept valves closed too long thereby preventing the OPC-103 percent controller from functioning properly. The shorter delay time allows the OPC-103 percent controller to

perform its function of maintaining the turbine speed less than 103 percent, while not slowing down the turbine to the point that a reactor trip occurs.

The design change did not affect the ability of the DEH P2000 to trip the turbine. The accident analysis for loss of load assumes a turbine trip is the initiating event, since this causes a more severe transient. This design change did not affect the loss of load accident analysis. Therefore, no unreviewed safety question was found to exist.

24. <u>Revise Design Basis for Minimum Auxiliary Feedwater Flow Rates</u>
DCP M-049222 Rev. 0 (Units 1 & 2)

The AFW system operation window is limited by the maximum and minimum flow rate criteria. This limited flexibility of AFW flow rates has been a concern for plant operations. This design change adopts the results of a Westinghouse analysis as the new design basis for minimum required AFW flow rates. The changes include reducing the minimum requirement for the AFW pump flow rates from 440/880 gpm to 410/820 gpm for the motor-driven and turbine-driven pumps, respectively. Reducing these flow rates avoids steam generator overfill during a steam generator tube rupture scenario and provides a larger window for AFW system operation.

<u>Safety Evaluation Summary</u>

This design change involves document changes only and there are no physical modifications to the plant components. The change does not impact normal operation of the plant.

However, there is an effect on steam release in an MSLB outside containment event. Hence, the impact of AFW flow reduction on environmental qualification was investigated and it was concluded that previously analyzed blowdown cases represented greater energy input to equipment compartments.

A reduction in the minimum required AFW flow does not relate in any way to the initiation of an accident, does not change the likelihood of an event to occur, and does not introduce any new failure mechanisms. The results of the Westinghouse analyses demonstrate that no safety analyses acceptance criteria are violated. Therefore, the same margin exists to the design failure point or system limitation, and the margin of safety as defined in the bases for any TS will not be reduced.

25. Remove Some Straight Sections of the Steam Generator Tubes from Steam Generator 1-2
DCP N-049253 Rev. 0 (Unit 1)

Steam generator eddy current testing during 1R7 identified several large primary water stress corrosion cracking (PWSCC) indications at tube support plate locations that could exceed the structural limits for Regulatory Guide 1.121 tube burst integrity. This design change allowed the removal of 1 to 8 steam generator tubes from Steam Generator 1-2 to provide samples for visual inspection, as well as for laboratory examination and analysis to provide a direct correlation between the indications and eddy current test results and to establish structural integrity.

Safety Evaluation Summary

The process employed to remove the tube segments is designed and procedurally controlled to prevent contact with adjacent tubes and to ensure the integrity of the remaining tubes. In addition, the stabilized tube remnants in the steam generator have been analyzed to show that no compromising contacts with adjacent tubes will occur during normal operations and accidents conditions. The cold leg tube plugs are of the standard design like those already in use, and the hot leg tube sheet plugs are designed and installed as reactor coolant pressure boundary components complying with ASME Section III and XI requirements. The plugs are welded into the tubesheet cladding. This ensures the integrity of the plugs. Since the modifications only affect localized, passive, structural components, changes to the potential failure modes and their effects are limited and no dose consequences are increased.

With removal and plugging of 1 to 8 tubes, the total number of deactivated tubes in Steam Generator 1-2 will be approximately 126, which is less than 4 percent of the total and within the 7.5 percent maximum restriction associated with DCPP's peak cladding temperature margins. Also, any flow imbalance created between the four reactor coolant loops will be negligibly small. Furthermore, the removal of 1 to 8 tubes worth of flow area by plugging will still maintain the circulation capability of the loops well above that required and/or assumed in plant analyses. Hence, there is no impact on the margin of safety as defined in the basis for any TS.

26. Replace 25-kV/12-kV Auxiliary Transformer
DCP E-049254 Rev. 0 (Unit 1)

Unit Auxiliary Transformer 1-1 (UAT11) was damaged when it was unable to withstand a through fault (the current going through the transformer when there is a short in the connected load). This modification provides and installs the replacement transformer. The new transformer has more than adequate

capacity to withstand the maximum potential through fault. It also contains about 1,146 gallons more oil than the original transformer.

Safety Evaluation Summary

This replacement UAT 11 serves the same function as the previous transformer and restores the plant's 12-kV electrical distribution system to its original configuration. Therefore, this transformer does not impact accident analyses, consequences, create new potential accidents, or reduce margin of safety as defined in the basis for any TS. All environmental documents were revised to account for the additional oil contained in the new transformer. As the short circuit interrupting rating of these breakers was increased, the related margin was increased.

27. Pressurization of the Component Cooling Water Surge Tank
DCP M-049284 Rev. 0 (Unit 1)
DCP M-050284 Rev. 0 (Unit 2)

Due to the potential for the CCW fluid to flash inside the CFCUs during a large break LOCA coincident with a loss of offsite power, the CCW surge tank was pressurized to 17 psig (nominal) to increase the static head on the system.

The modification uses plant nitrogen as the primary source of pressure, with a Design Class I backup pressure source from nitrogen bottles with redundant regulators. An additional backup pressure supply is also provided by plant instrument air. The existing surge tank vent, RCV-16, which closes in the event of radioactive inleakage to the tank, will remain intact, with a downstream backpressure regulator.

Safety Evaluation Summary

The modification meets all CCW system design requirements. The safety function of the compressed gas is to maintain CCW system pressure at or above 17 psig for the first minute of the accident. Since the compressed gas maintains the surge tank pressure before a design basis accident, the components added by this modification have no active safety function to perform once the accident has occurred.

The safety function of the new components is to maintain pressure boundary integrity. As noted in the FSAR Update, the failure of these components is not postulated for the first 24 hours following a LOCA. All components used to pressurize the surge tank are seismically qualified and installed and meet the piping and instrumentation codes and standards for a Design Class I system.

The use of nitrogen to pressurize the tank will not cause any adverse effect on CCW chemistry or heat transfer capability since it is an inert gas. Although nitrogen is the preferred gas for the surge tank pressurization, compressed air can be used as a pressurization source. No adverse effects are postulated with the use of compressed air since the surge tank was previously open to atmosphere.

Based on these design aspects, the safety evaluation concluded that no unreviewed safety question existed.

28. Use of ZIRLO Cladding and 5 Percent Enrichment Fuel
DCP-N-049285 Rev. 0 (Units 1 & 2)

During refueling outages starting with 2R7 and 1R8, some of the fuel will be replaced with slightly different fuel assemblies. The new fuel assemblies will utilize a zirconium-based alloy called ZIRLO rather than the Zircaloy-4 alloy used up to this time for cladding and grid composition. Some of the fuel assemblies will also have a higher enrichment than the previously loaded fuel.

This DCP identified and addressed all the issues resulting from and associated with these differences in the fuel. Although this higher enrichment allows the fuel cycle to continue for up to 21 months, this DCP does not authorize the extension of the fuel cycle beyond 18 months. Previously issued license amendments authorize the storage of fuel with a higher enrichment in the spent fuel pool. This DCP did not include the physical work associated with the actual fuel reload because that work is controlled in accordance with plant operating procedures.

Safety Evaluation Summary

There will be no increase in the probability of an accident previously evaluated in the FSAR Update as a result of the changes in the fuel assemblies or the fuel cycle length. There is no mechanism introduced by which the frequency of an accident initiator can be increased. The fuel cladding material does not cause or contribute to the initiation of any Chapter 15 accident.

A Westinghouse evaluation confirmed that no degradation of RCS piping or components or the fuel will result from the RCS chemistry that will be used in the RCS during fuel cycles utilizing this fuel. Westinghouse performed a DCPP-specific steam generator tube integrity review that considered the higher lithium concentration that will be required in the RCS during the beginning of the fuel cycle. Westinghouse concluded that the elevated lithium concentration does not constitute a threat to the integrity of the steam generator tubes.

A Westinghouse analysis confirmed that the changes to the fuel and the proposed cycle length have a negligible impact on the following:
fuel temperature evaluations, steam line break mass and energy calculations, design transients/margin to trip, steam generator tube rupture (SGTR) analysis, reactor protection system/engineered safety feature actuation system setpoint calculations up to 18 months, containment pressure analysis, electrical system or component considerations, mechanical performance of the fuel (including corrosion resistance, creep, and Li resistance), and plant TS for the first 18 months of the cycle.

The 2 to 3 degree increase in the peak cladding temperature for some large- and small-break LOCA analyses represents a small and insignificant fraction of the available margin in these analyses. There will be no change in the LOCA hydraulic forcing function safety evaluation (blowdown reactor vessel and loop forces), the post-LOCA long-term cooling subcriticality requirement, and hot leg switchover time analysis.

Westinghouse has confirmed that the core design meets all applicable design criteria and ensures that all pertinent licensing basis criteria are met. This "precludes new risks to systems and components that could adversely affect the ability of the existing systems and components to mitigate the radiological consequences of any accident and/or adversely affect the integrity of the fuel rod cladding as a fission product barrier."

Another analysis supports the conclusion that the use of this fuel in cycles up to 21 months will not increase the radiological consequences of accidents previously evaluated in the FSAR Update.

Westinghouse has confirmed that the metal-water reaction rates for ZIRLO fuel cladding at 1800 degrees F are slightly lower than those for Zircaloy cladding.

Analyses have confirmed that the reactor vessel and its internals will not be adversely impacted by the use of this fuel. The additional radiation effects were considered.

Westinghouse has confirmed that there are no significant mechanical differences between the ZIRLO clad fuel and the fuel previously used. The ZIRLO has improved corrosion and creep resistance. Seismic analyses by Westinghouse concluded that the reactor vessel and its internals remain seismically qualified with this new fuel and reactivity will not be affected.

29. Addition of Three New Doors at Elevation 85 Feet, Turbine Building, Cold Machine Shop
DCP A-050108 Rev. 0 (Unit 2)

To create a more efficient tool storage and issuance facility, three new doorways and doors were added in the tool room area of the Unit 1 cold machine shop on the 85 foot elevation of the turbine building. Because of the new doorways the design change also made a document change to identify the new Fire Zone 16 boundary and to reflect the 3-hour rating requirement for the south wall of the tool room. The three new doors do not perform as flood or high energy line break barriers.

Safety Evaluation Summary

Extending Fire Zone 16 to the 3-hour, fire rated wall located approximately 10 feet to the south does not compromise the DCPP Fire Protection Program. The fire rating of Fire Zone 16 will remain 3 hours. The fire zone boundary extension does not impact any safety-related equipment or circuits. The change to the DCPP Fire Protection Program will not impact the ability of DCPP to achieve and maintain safe shutdown in the event of a design basis fire.

30. Install Resized Restricting Orifice at Safety Injection Pump 2-2 Discharge
DCP N-050235 Rev. 1 (Unit 2)

This design change resized the restricting orifice (RO) for Safety Injection Pump (SIP) 2-2 and installed a pressure tap (PX-650) downstream of the RO to allow monitoring of the integrated performance of the pump and RO. The safety evaluation was written to show that ECCS performance was not adversely affected.

Safety Evaluation Summary

The safety evaluation determined that the characteristics of SIP 2-2, in conjunction with RO-269, would be verified to be bounded by the maximum and minimum pump performance curves used in the accident analysis by testing prior to restart from 2R7. In addition, total pump flow and branch line flow balance would also be verified by testing prior to declaring the pump operable. These tests were completed and verified proper pump performance and system flow results.

31. Replacement of Process Protection System Equipment
DCP J-041540 Rev. 2 (Unit 1)
DCP J-042540 Rev. 0 (Unit 2)

This design change replaces the existing process protection system (PPS) Hagan 7100 analog equipment with the Eagle 21 microprocessor-based digital equipment supplied by Westinghouse. Eagle 21 is designed as a form, fit, and functional replacement for the Hagan 7100 PPS equipment.

Safety Evaluation Summary

LAR 92-05 submitted for the PPS modifications based on the 10 CFR 50.59 safety evaluations presented by Westinghouse was accepted and approved by the NRC. Replacement of the PPS equipment does not impact accident analysis, consequences, or create new potential accidents.

32. Deletion of Gross Failed Fuel Detector High Alarm
DCP J-043240 Rev. 0 (Unit 1)
DCP J-044240 Rev. 0 (Unit 2)

Regulatory Guide 1.97, Revision 3 only requires an indication function for the gross failed fuel detector (GFFD). Since the GFFD has been producing nuisance alarms, its high alarm annunciation in the control room was deleted. The required local indication will remain.

Safety Evaluation Summary

The GFFD performs only a postaccident monitoring function. Its annunciation in the control room is not required. The deletion of the GFFD high alarm annunciation does not affect any safe shutdown function, nor is the GFFD governed by TS.

33. Downgrade the Safety-Related Portion of the Boric Acid Heat Tracing System
DCP N-045376 Rev. 0 (Unit 1)
DCP N-046376 Rev. 0 (Unit 2)

With the reduction in boron concentration from 12 to 4 weight percent, providing heat tracing to the boric acid system is not required. This modification downgrades the safety-related heat tracing to nonsafety-related and lowers the temperature settings for the boric acid storage system to a temperature above 65 degrees F, which is the TS 3.1.2.6 limit.

Safety Evaluation Summary

The solubility limit of the 4 weight percent boric acid is close to room temperature. In accordance with TS 3.1.2.6, the boric acid system is required to be maintained above this temperature (65 degrees F). This would allow sufficient time for operator intervention once the temperature alarm was sounded.

Since the boric acid system can be maintained above its solubility limit through the use of the boric acid storage tank (BAST) heaters and through normal ambient temperatures, availability of boric acid for accident mitigation will not be compromised. A seismic evaluation of the BAST, filters, and the boric acid transfer pumps, along with an evaluation of the stress levels that these heat traced lines would be exposed to by this temperature reduction, was performed and found to be acceptable.

Therefore, downgrading the safety-related heat tracing to nonsafety-related and lowering the temperature settings for the BAST and the heat tracing to a temperature above 65 degrees F will not have any impact and will not reduce the margin of safety as defined in the basis for any TS.

34. Reduction of the Fuel Handling Building Ventilation Supply Air Flow Rate
DCP H-045932 Rev. 0 (Unit 1)

The fuel handling building (FHB) ventilation supply air flow was reduced to facilitate the FHB ventilation system's ability to maintain 1/8-inch wg negative pressure differential as required by the TS. The flow reduction was accompanied by rebalancing of the air flow distribution to maximize the cooling effect of the supply air.

Safety Evaluation Summary

It was demonstrated that with the reduced air flow rates and the system rebalance, the FHB ventilation system was still able to maintain the design room temperature for the FHB. Thus, there was no adverse affect on the equipment inside the FHB.

B. Temporary Plant Modifications, Electrical Jumpers and Lifted Leads, Mechanical Jumpers and Bypasses, and Test Equipment

1. <u>Steam Generator 1-3 Narrow Range Level Channel 537 - Recorder Installation</u>
   Jumper #94-66 (Unit 1)

   This jumper allowed the temporary installation of test equipment (recorder) to Steam Generator 1-3 Level Channel 537 to isolate a problem associated with a channel signal spiking. This test equipment was connected to the normal test points at the front of the Eagle rack and was installed in a manner that ensured the continued seismic qualification of the equipment. The leads were run to maintain circuit separation criteria and were installed in one rack for one channel only.

   <u>Safety Evaluation Summary</u>

   The installation of a recorder using the normal test points did not have any impact on the operation of the equipment nor did it increase the probability of the malfunction of the equipment. The margin of safety was not decreased by the addition of this temporary installation, nor was the probability or consequence of any accident previously evaluated in the FSAR Update increased. The installation of the recorder met the Seismically Induced Systems Interaction Program requirements and did not reduce the margin of safety.

2. <u>Component Cooling Water Surge Tank - Removal of Relief Valve RV-45 and Functional Bypass of RCV-16</u>
   Jumper #95-14 (Unit 1)

   Relief Valve RV-45 was removed to perform an inspection of the component cooling water surge tank to check for foam and signs of biofouling. This condition rendered RCV-16 ineffective for isolating the surge tank in the event of the release of any radiological material. The jumper permitted bypassing of the function of RCV-16 to close on a high radiation signal while Relief Valve RV-45 was removed and the tank was open to atmosphere.

   <u>Safety Evaluation Summary</u>

   The function of RCV-16 is to close on a high radiation signal. Since this function was bypassed, in the event of a high radiation alarm, the control room has an annunciator alarm. In addition, there is also an alarm for high surge tank level. This would allow time for the source of the inleakage to be identified and isolated before a large quantity of water leaks onto the roof. Also, the manual

valve CCW-1-89 upstream of RCV-16 will be opened and any discharge can be routed to an appropriate drain to minimize the potential for an uncontrolled release. There was no impact on the margin of safety due to the removal of RV-45 since there are no Technical Specifications (TS) or Equipment Control Guidelines associated with the function of the relief valve.

3. Control Room Ventilation System - Damper VAC-2-M-4 Manual Closure
Jumper #95-16 (Units 1 & 2)

This jumper was installed to manually close the control room ventilation system (CRVS) operating filter train recirculation damper, VAC-2-M-4. The jumper supported installation of a blank-off plate to allow maintenance of FU-39, in accordance with DCP H-49243. This modification put the CRVS in a configuration different from that described in the FSAR Update and prevented normal operation of the system. To prevent unfiltered air from entering the control room if a loss-of-coolant accident were to occur, this damper should be blocked in the closed position. The damper is normally closed for CRVS operating Modes 1 and 2 and is open for CRVS operating Modes 3 and 4.

Safety Evaluation Summary

The control room is required by design to have a positive air pressure of 1/8 inch water gauge to prevent the introduction of unfiltered air through the recirculation duct. The jumpered damper is normally closed for CRVS operating Modes 1 and 2 and open for CRVS operating Modes 3 and 4. Normally in CRVS Mode 4, damper VAC-2-M-4 is open to allow recirculation air to be processed through the charcoal filters. However, if this damper were open with the filter train FU-39 isolated, recirculation air from the control room would cause the pressurization fans to backup, resulting in a reduction of air pressure in the control room. With the damper closed in CRVS operating Mode 4, there would be zero recirculation and 100 percent outside air. This would ensure sufficient air to maintain the design requirement of positive pressure in the control room.

Installation of the jumper did not prevent the system from performing its intended function and did not create any new malfunctions of equipment. Also, the margin of safety was not reduced by the installation of the jumper.

4. Diesel Generator 1-1, 1-2, and 1-3 Surveillance Test Procedure M-15, Part B - Performance with Kilowatt Sensing Relay Disabled
Jumper #95-52 (Unit 1)

Surveillance Test Procedure M-15, Part B, verifies the automatic loading of the diesel generators to meet Safety Guide 9 requirements with regard to transient voltage and frequency recovery times during load applications. The

manufacturer of the kilowatt sensing (KWS) relay, Basler Electric, has advised that the pickup values as published in the product bulletin are not correct, and the tested minimum pickup value is 25 to 30 percent, as opposed to the published pickup value of 10 percent. The jumper disabled the KWS relay to obtain frequency, voltage, and kilowatt data for the Unit 1 seventh refueling outage surveillance testing.

Safety Evaluation Summary

Disabling the KWS relay only during surveillance testing was a temporary modification and did not affect the ability of the diesel generators to perform their intended functions. The starting and loading capabilities of the diesel generators were verified through periodic surveillances not associated with the KWS relay. The revised configuration did not change, degrade, or restrict the operational capability of the diesel generators.

The margin of safety of the diesel generators was not reduced by installation of the jumper, and the jumper did not render the diesel generators unable to perform their intended safety function for the period of time the jumper was installed.

5. Service Cooling Water Heat Exchanger Cooling - Firewater Supply
Jumper #96-08 (Unit 2)

The normal cooling for the service cooling water (SCW) heat exchanger is provided by the salt water system. During the Unit 2 seventh refueling outage, both the circulating water pumps were secured for maintenance and testing. This required an alternate cooling supply for the SCW heat exchanger. This cooling supply was provided by the fire water system. The supply was connected through a regulator to lower the water pressure from 89 psig to 40 psig.

Safety Evaluation Summary

The SCW heat exchanger is normally provided cooling water supply by the salt water system as described in the FSAR Update. The jumper provided an alternate cooling supply from the firewater system during the refueling outage. This cooling enabled the SCW heat exchanger to perform its intended function for Modes 5 and 6. The 90 psig fire water supply pressure is well within the SCW heat exchanger design pressure. The installation was seismically installed and there was no impact of the jumper on the operating requirements of the SCW or the firewater systems. The firewater system was evaluated to (1) ensure its capability to supply the cooling flow while still providing fire protection capabilities, and (2) ensure its integrity for the possible failure modes of the jumper configuration. Potential turbine building flooding issues were also

addressed. The jumper had no impact on any safety parameter and there was no reduction in the margin of safety as defined in the basis for any Technical Specifications.

6. Diesel Generator 2-1 Lube Oil Heater - Alternate Vital Power Supply
Jumper #96-13 (Unit 2)

The normal power supply to Diesel Generator 2-1 auxiliary power panels is from Bus G-Panel MPG 31 and Bus F-Panel MPF 28. Panel MPF 28 provides power to the lube oil heater. Bus F was required to be cleared for maintenance during the outage, and an alternate source of power was to be provided to the lube oil heaters to maintain the oil temperature above 90 degree F. The jumper provided temporary power from Bus G Panel MPG 31 to maintain the lube oil temperature

Safety Evaluation Summary

The temporary power supply to the lube oil heaters was from a different source than that described in the FSAR Update. Energizing the lube oil heaters from a different power source had no impact on the ability of the heaters to perform their intended function. In case of a fault or disruption of power supply, Diesel Generator 2-2 would satisfy the Technical Specification requirement of having one diesel generator OPERABLE. The jumper did not reduce the margin of safety as defined in the basis for any Technical Specification and had no impact on the ability of the diesel generator to perform its intended function.

7. Diesel Generator 2-3 Tachometer Package YM3-23 - Additional 24-Vdc Power Supply
Jumper #96-14 (Unit 2)

This jumper involves the temporary installation of a 125-Vdc to 24-Vdc converter and associated wiring to supply the Diesel Generator 2-3 tachometer package YM3-23 with a 24-Vdc power supply during plant Modes 5 and 6 only. The temporary power supply was an alternate power supply for the normal 120-Vac source to maintain the operability of the tachometer package and Diesel Generator 2-3 in the event of a loss of the 120-Vac source during testing of Inverter IY21. Diesel Generator 2-3 and its tachometer package currently have only one source of power (120 Vac).

Safety Evaluation Summary

The jumper provided an additional source of power to maintain the ability of tachometer package YM3-23 (in control panel GQD 23) to support operability of Diesel Generator 2-3. This additional power supply did not degrade the performance of the tachometer package or the diesel generator. The function of

the system was not affected by the jumper and the additional power supply, and operation of the diesel generator was not changed. The temporary jumper retained the function of the tackometer package and maintained operability of the diesel generator as required by the Technical Specifications.

The margin of safety of the diesel generator system was not reduced by installation of the jumper, and the jumper did not render the diesel generator unable to perform its safety function.

8. <u>Diesel Generator 2-3 Lube Oil Heater - Alternate Vital Power Supply</u>
Jumper #96-16 (Unit 2)

The normal power supply to Diesel Generator 2-3 auxiliary power panels is from Bus F-Panel MPF 56 and Bus H-Panel MPH 49. Panel MPH 49 provides power to the lube oil heater. Bus H was required to be cleared for maintenance during the outage, and an alternate source of power was to be provided to the lube oil heaters to maintain the oil temperature above 90 degrees F. The jumper provided temporary power to maintain the lube oil temperature from the Bus F Panel MPF 56.

<u>Safety Evaluation Summary</u>

The temporary power supply to the lube oil heaters was from a different source than that described in the FSAR Update. Energizing the lube oil heaters from a different power source had no impact on the ability of the heaters to perform their intended function. In case of a fault or disruption of power supply, Diesel Generator 2-1 would satisfy the Technical Specification requirement of having one diesel generator OPERABLE. The jumper did not reduce the margin of safety as defined in the basis for any Technical Specification and had no impact on the ability of the diesel generator to perform its intended function.

C. Procedures

1. Chemistry Administrative Procedure: Chemical Additions to the Closed Cooling Water Systems
CAP O-6 Rev. 3 (Units 1 & 2)

This procedure is a part of a chemistry control program designed to provide flexibility in dealing with chemistry, corrosion, and microfouling concerns in the closed cooling water systems. This procedure revision introduces a new biocide (Isothiazolin), biodispersent (Nalco 7348), iron dispersent (Nalco 7302), and antifoam (Nalco 7471) to the closed cooling water systems to control micro-organisms.

Safety Evaluation Summary

This procedure revision involves chemicals that have been tested to ensure that the potential for foaming has been minimized and that there would not be any material compatibility or chemical interaction issues. Chemical control improves the availability of the closed cooling water systems and, therefore, improves availability of the equipment important to safety cooled by the closed cooling water systems.

This change deals with chemicals added to the system to maintain chemical control.. The addition of the reviewed chemicals will not adversely affect component cooling water system hydraulics, heat transfer capability, or pump operation. The temperature limits in the Technical Specifications bases are not affected. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

2. Chemistry Administrative Procedure: Alternative Steam Generator Layup and Startup Chemical Additions
CAP O-14 Rev. 0 (Units 1 & 2)

Based on a Westinghouse Steam Generator Group recommendation, this procedure revision allows the use of dimethylamine (DMA) and ammonia in layup and startup activities to increase return of sludge and to loosen sludge in the steam generators. The amine currently used for this purpose is ethanolamine (ETA) with hydrazine.

Safety Evaluation Summary

A small amount of the new chemical for pH control is only used during layup. Ammonia use is already permitted from past operations. The new chemical will have very little effect on the steam generators and will help cleanup of the steam generators during shutdown. Therefore, layup chemicals with concentrations in

the low parts per million will not have any effect on safety analysis, nor will the chemicals cause a new type of accident.

3. Department Level Administrative Procedure (DLAP): Control of the Surveillance Testing Program
AD13.DC1 Rev. 1 (Units 1 & 2)

This procedure revision adds MS-902 to the list of valves allowed to be open on an intermittent basis, subject to administrative controls of OP-12. The revision also adds valve closure time requirements that were previously listed in Technical Specification Table 3.6-1 (License Amendments 73 and 72) that was relocated improperly to the Inservice Testing Program Plan.

Safety Evaluation Summary

Opening MS-902 will supply back-up air to the chemical and volume control system valves 8149A, B, and C and FCV-459 and 460 in accordance with Abnormal Operating Procedure AP-9, "Loss of Instrument Air," and will allow normal letdown to be reestablished. Hence, the probability of an accident may be reduced because use of backup nitrogen to maintain letdown will help avoid unnecessary shutdown and resultant thermal cycles in the event of loss of instrument air that exceeds the Technical Specification 3.6.3 limiting condition for operation of 4 hours.

Opening MS-902 beyond the Technical Specification limiting condition for operation does not increase the consequences of any accidents because a dedicated operator would be stationed at the valve to close it upon direction from the control room. There are no limits to the number of openings of other manual containment isolation valves that are allowed to be opened intermittently under administrative control. Opening MS-902 would only occur in an abnormal operating situation. There is no impact on the margin of safety.

4. Operating Procedure: Chemistry Control Limits and Action Guidelines for the Secondary Systems
OP F-5:II Rev. 10 (Units 1 & 2)

Changes to secondary chemistry in this procedure revision reflect increased conductivity caused by ethanolamine, a new recommendation from Westinghouse, and new INPO chemistry guidelines for feedwater iron and dissolved oxygen. This procedure revision also adds information on instruments and VB3 alarms and the use of steam generator blowdown demineralizers.

Safety Evaluation Summary

The use or non-use of steam generator blowdown demineralizers has no effect on accident mitigation. All chemistry changes are very small and reflect current operation and are conservative. This procedure revision does not affect the accident analysis, the type of accidents previously evaluated, malfunctions of equipment important to safety, or the margin of safety.

5. Transfer to Hot Leg Recirculation
EOP E-1.4 Rev. 11 (Unit 1)
EOP E-1.4 Rev. 4 (Unit 2)

The post-LOCA hot leg switchover time was reduced from 13.5 hours to 10.5 hours in Revision 11 (4-26-94) to Emergency Operating Procedure (EOP) E-1.4. This safety evaluation was prepared after the revision had been approved in response to a PG&E NQS Audit 963460052 finding that the previous 13.5-hour transfer to hot leg recirculation time was changed in EOP E-1.4 (Rev. 11) to 10.5 hours while the FSAR Update Sections 6.3 and 15.4 still reflected the 13.5-hour criteria. Westinghouse letter PGE-94-584, "Hot Leg Switchover Time Assessment," provided the justification for changing the hot leg switchover time criteria to 10.5 hours.

Safety Evaluation Summary

This change involves the post-LOCA alignment of an accident mitigation system that is not the cause of or initiating event of an accident. Therefore, the probability of occurrence of an accident is not changed.

Switchover to hot leg recirculation slightly decreases the heat removal capacity of the ECCS due to an increase in system resistance and corresponding decrease in ECCS flow. This affect, not previously modeled, was included in WCAP-13907 and -13908 with a bounding 10 hours used since the earlier switchover to hot leg recirculation is more critical to containment temperature response. The Westinghouse analysis and ISAG Calc. 921214-1 Rev. 2 show insignificant changes to the containment temperature/pressure response and sufficient core cooling available with switchover at 10.0 hours. Consequently, there is no affect on the containment integrity or qualification of equipment. With regard to boron precipitation, the earlier switchover to hot leg recirculation is more conservative in preventing this phenomenon from occurring. Therefore, the consequences of a previously evaluated accident are not changed.

Hot leg switchover is performed to prevent boron precipitation and a potential reduction of heat transfer from the fuel which may reduce the effectiveness of ECCS cooling. An earlier time to switchover reduces the possibility of boron precipitation. The same equipment is used under similar conditions to

accomplish this task. Therefore, the probability of malfunction of equipment important to safety is not changed.

Switchover to hot leg recirculation is part of post-accident long-term ECCS cooling and, therefore, an accident of a different type is not created.

Technical Specifications 3.5.2 and 3.5.3 provide for the operability of ECCS subsystems, with provisions for long-term core cooling capability in the recirculation mode. The switchover to hot leg recirculation is provided for as part of the recirculation capability but the switchover time limit is not discussed in the bases of the Technical Specifications. An earlier switchover time is more conservative in preventing boron precipitation and does not compromise essential decay heat removal since the decay heat load will have exponentially decayed from the event initiation. Therefore, there is no reduction in the margin of safety as defined in the bases of any Technical Specification.

D. Tests and Experiments

1. <u>Surveillance Test Procedure: Weekly and Monthly Fire Valve Inspection</u>
   STP M-67A Rev. 20 (Units 1 & 2)

   This test procedure involves the inspection of fire water valves. This procedure revision extends the inspection period from weekly to monthly for sealed fire water valves.

   <u>Safety Evaluation Summary</u>

   The change to the inspection frequency of sealed fire water valves from weekly to monthly does not affect combustible loading or ignition sources as previously evaluated in the FSAR Update. In addition, the inspection frequency of the fire water valves does not affect the function of structures, systems, or components described in the FSAR Update. Therefore, systems currently credited to mitigate the effects of a fire will be unaffected by this change.

   The change in inspection frequency of the fire water valves does not affect the operation of equipment important to safety. The design and function of the sealed fire water valves have not been changed. Therefore, this procedure revision does not affect the accident analysis, the type of accident previously evaluated, the malfunction of equipment important to safety, or the margin of safety.

2. <u>Surveillance Test Procedure: NUREG-0737; Safety Injection System Pump Suction Leak Reduction</u>
   STP M-86A1 Rev. 5 (Unit 1)
   STP M-86A1 Rev. 3 (Unit 2)

   This test procedure revision allows an option of pressurizing the safety injection system (SIS) suction piping with residual heat removal (RHR) Pump 2 in the test mode (Mode 6). The proposed configuration allows the piping to be pressurized quickly and without as much temporary setup as is necessary using a hydro pump, which is the current practice.

   <u>Safety Evaluation Summary</u>

   This revision does not involve any material changes to the systems involved. Test pressures are within design requirements for the SIS suction piping. The clearance boundary on the SIS ensures flow is not diverted from the RHR system. The SIS components are not required to perform a safety function in the operating mode in which the test is performed (Mode 6). Operation of the RHR

system is within its design parameters. This test does not affect the ability of RHR Train 2 to deliver flow to the core.

The test alignment actually increases the margin of safety by taking suction from the refueling cavity rather than the RHR sump, thus decreasing the chance that RHR suction may be lost during testing. The test does not create any new accident scenarios. Pressurization of SIS suction piping with a RHR pump simulates previously evaluated operation in the post-LOCA recirculation alignment.

3. <u>Surveillance Test Procedure: Flow Balancing Component Cooling Water to Equipment on the Centrifugal Charging Pump Skid</u>
STP PEP M-200 Rev. 0 (Units 1 & 2)

This test procedure installs and removes ultrasonic flow meters that have been calibrated by Technical and Ecological Services to allow testing of the component cooling water (CCW) flow to the individual components on centrifugal charging pump (CCP) pump skids. The test can be performed in any mode provided the RHR heat exchangers are not in service. The test measures flows to the equipment and adjusts the flows to meet minimum design requirements.

<u>Safety Evaluation Summary</u>

This test procedure allows for engineering control of the valves that affect the flow balance of CCW to the CCP skid equipment. This ensures that CCW flow is maintained within required limits to the CCP skid components important to safety. Therefore, this procedure does not affect the accident analysis, the type of accident previously evaluated, the malfunction of equipment important to safety, or the safety margin.

4. <u>Surveillance Test Procedure: Determination of Recirculation Flow through the Centrifugal Charging Pump Miniflow Orifice</u>
STP PEP M-222 Rev. 0 (Units 1 & 2)

This test procedure determines CCP recirculation orifice flow capacity during normal charging and correlates that flow capacity to post-LOCA conditions. If the orifice flow capacity is found to be excessive, the valve downstream of the recirculation orifice is throttled to lower the orifice flow to prevent the CCP from exceeding the allowable total pump flow, while maintaining the required minimum flow.

<u>Safety Evaluation Summary</u>

A computer analysis, using the PEGISYS computer code, determined that recirculation orifice flow capacity could be adjusted within the limits given in this

procedure without adversely impacting the other critical parameters of the emergency core cooling system (ECCS) flow balance. Preventing excess flow and maintaining the required minimum flow ensure that the CCPs are available to perform their accident mitigation function.

Since the PEGISYS computer analysis of the parameters affected by this procedure shows that all of the requirements of Technical Specification 4.5.2.h, including total pump flow, will be met following the completion of a valve adjustment to limit recirculation flow, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

5. <u>Surveillance Test Procedure: Setting of the Centrifugal Charging Pump 2-1 Miniflow Orifice Flowrate</u>
STP PEP M-223 Rev. 0 (Unit 2)

This test procedure measures the flow through the CCP 2-1 recirculation orifice during performance of Surveillance Test Procedure (STP) P-CCP-A in Mode 6 and throttles the valve downstream of the CCP recirculation orifice to lower flow. This ensures that the CCP 2-1 recirculation orifice design resistance is restored and that the CCP 2-1 satisfies STP-V-15 acceptance criteria.

<u>Safety Evaluation Summary</u>

Throttling the valve downstream of the CCP 2-1 recirculation orifice limits the orifice flow capacity. This ensures that the CCP will not exceed its maximum flow limit during accident conditions while maintaining the required minimum flow. Preventing the excess flow and verifying that the required minimum flow is available ensure that CCP 2-1 is available to perform its accident mitigation function.

There are no new credible accidents created by the performance of this test procedure and it does not affect the malfunction of equipment important to safety. Since the requirements of Technical Specification 4.5.2.h are verified to be met following the completion of a valve adjustment during the performance of STP V-15, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

6. <u>Surveillance Test Procedure: Residual Heat Removal Heat Exchanger 1-1 Performance Test</u>
STP PEP M-238 Rev. 0 (Unit 1)

This test procedure installs and removes test instruments that allow testing of the heat transfer capability of the RHR heat exchanger. The test is typically performed during a plant cooldown in Mode 4 and it measures RHR and CCW flows into the RHR 1-1 heat exchanger. It also measures temperatures in and

out of both RHR and CCW sides of the RHR 1-1 heat exchanger. The results are used to evaluate the condition of the CCW system regarding microbiological fouling/blockage concerns.

Safety Evaluation Summary

In Mode 4, which is when this test is performed, one RHR train or one reactor coolant system loop is required to be operating. This test is performed with both RHR trains operating but separated by closing RHR-1-8726A. The alignment of the "A" train of RHR in this test by closing RHR-1-8726A isolates this train from the heat exchanger bypass loop, but this does not affect the design basis for any analyzed accidents, nor does it change the probability of occurrence of any analyzed accident since it does not restrict either train of RHR from operating or from injecting into all four cold legs.

The alignment of the "A" train of RHR in this test does not prevent the RHR system from performing its intended design basis function. The only possible consequence is the possibility of exceeding the allowable cooldown rate during shutdown cooling. The cooldown rate is controlled by operator action to throttle the flow through both heat exchangers and through the bypass loop for the "B" RHR train. The Technical Specification basis for the RHR system in Mode 4 is that only one operable ECCS subsystem is required. This basis is not affected by this test procedure; therefore, there is no reduction in the margin of safety.

7. Temporary Procedure: Implementation of DCP E-49099 - Battery 11 Replacement
TP TA-9501 Rev. 0 (Unit 1)

This temporary procedure implements Battery Bank 11 replacement with LCUN-33 type cells in accordance with DCP-E-49099. Energizing vital DC Bus 11 and its associated loads from nonvital Battery 17 is required to keep some loads functional during battery replacement. Battery 17 was chosen since it has ample margin on its respective battery and battery charger. Most loads that are being deenergized and jumpered to nonvital Battery 17 are considered "Functional" and not "Operable." Technical Specification related loads that are being jumpered from the nonvital battery have been evaluated for operability, and appropriate compensatory measures are taken in the event the nonvital power from Battery 17 is lost.

Safety Evaluation Summary

This work is performed during Modes 5 or 6. Powering vital DC Bus 11 Class 1E loads from nonvital Battery 17 does not increase the probability of occurrence of an accident, nor does it have any impact on the consequences of the accidents described in the FSAR Update. Should the temporary nonvital DC power from

Battery 17 fail to DC Bus 11 or its loads, the effects would not increase the consequence of malfunction on the vital DC Bus 11 or its loads. The clearance and the jumping process is a controlled evolution where individual loads are deenergized and reenergized one at a time. In addition, this procedure contains contingency actions to take should loss of power from Battery 17 occur during the replacement period. The jumpering to a nonvital battery does not take credit for the limiting condition for operation (LCO). It has been determined that having the battery out of service does not violate Technical Specification LCO 3.8.3.2.

8. <u>Temporary Procedure: Feed and Bleed of the Component Cooling Water System</u>
TP TB-9512 Rev. 0 (Unit 1)

This temporary procedure proposes an activity to feed and bleed the CCW system to reduce chemical concentrations in the CCW coolant. It utilizes installed system components in conjunction with a temporary drain line to drain CCW coolant out the plant discharge via the ASW discharge line. Makeup to the system is provided via the normal source using a manually controlled valve.

During feed and bleed, the level in the CCW surge tank is maintained in the normal operating range and automatic level control is not affected. Discharge flow is within the makeup capability of the makeup water system. An operator is stationed at the discharge flow control and isolation valves to immediately isolate discharge flow in the event of a low CCW surge tank level due to loss of makeup flow, system leak, or an emergency. Operators are directed to immediately secure discharge of CCW overboard if any indication of a primary coolant leak into the CCW system is detected. Operators are also directed to secure CCW discharge flow to the ASW system if ASW flow is lost to prevent any build up of CCW coolant on the ASW side of the heat exchanger. A check valve is placed in the temporary drain line to preclude seawater backflow into the CCW system.

<u>Safety Evaluation Summary</u>

The feed and bleed of the CCW system, along with the associated precautions and limitations, provide adequate assurance that safe CCW and ASW system operation will not be impacted. The length of time the system is left with dilute corrosion control inhibitors until chemicals are added to reestablish chemical concentration to effective levels is minimized. This results in continued operation of both the CCW and ASW systems at their regular effectiveness with no degradation of the margin of safety.

9. Temporary Procedure: Transportation of Replacement Transformers from the Intake Area to Parking Lot # 1
TP TD-9503 Rev. 0 (Unit 1)

The scope of this temporary procedure is to deliver four Unit 1 main transformers to DCPP, off-load the transformers, and transport them to a laydown area at Parking Lot # 1. This procedure provides that the barge be brought in the cove during daylight when the ocean swells are less than or equal to 4 feet and wind speed is less than 12 mph. The barge is to be located at the south end of the cove during the unloading period and will be anchored in the intake cove to four anchor points during unloading of the main transformers. Moving the transformers from the barge to the shore is performed when the elevation between the barge and the shore is within 2 feet. The transformers are then transported directly to the laydown area at Parking Lot # 1.

Safety Evaluation Summary

Communications are established between the control room, intake operators, and the vessel operators during any movements or maneuvers that could potentially impact plant operations. The control room would be aware of any changes or abnormal conditions and respond quickly to place the plant in a safe condition. The FSAR Update Chapter 15 accident analyses and evaluations have been reviewed and it was determined that there is no impact on the consequences of these accidents. The process of bringing a barge into the cove, offloading transformers, and transporting them to Parking Lot # 1 does not change the facility design, function, or method of performing any plant operation. Therefore, the transformer move does not reduce the margin of safety as defined in the basis for any DCPP Technical Specifications.

10. Temporary Procedure: Providing Vital 125-Vdc Power from SD13 to SD11 Loads
TP TD-9507 Rev. 0 (Unit 1)

This temporary procedure provides the installation of Class 1E DC jumpers from SD13 that temporarily reconfigures the vital DC power to Diesel Generator (DG) 1-3, 4-kV Bus F and its associated safeguards relay boards in Modes 5 and 6. This reconfiguration is necessitated due to the unavailability of Battery 11 when it is being replaced in Modes 5 and 6. Normally DC Bus 11 is configured to be connected to DG 1-3, 4-kV Bus F control and its associated safeguards relay board. As Battery 11 is being replaced during Modes 5 or 6, Bus 13 is to be connected to DG 1-3 via its normal/backup transfer switch and to the 4-kV Bus F and its associated safeguard relay board via Class 1E jumpers.

Safety Evaluation Summary

This temporary jumper modification only exists during Modes 5 or 6. Powering the 4-kV Bus F controls and associated safeguards relay board from the same backup vital DC source (Battery 13) as the associated DG 1-3 does not increase the probability of occurrence of an accident, nor does it have any impact on the consequences of the accidents described in the FSAR Update. DC Bus 13 loading and Battery 13 sizing evaluations with the added loads have been performed and found to be satisfactory. The jumpering of 4-kV Bus F and associated safeguards relay board to the same backup vital DC source (Battery 13) as DG 1-3 ensures the operability of associated components and provides the greatest flexibility in meeting the LCOs. At all times, the previously evaluated Technical Specification LCOs 3.8.1.2 and 3.8.2.2 are met or exceeded for the applicable mode.

11. Temporary Procedure: Provide Vital 125-Vdc Power from SD21 to SD22 Vital Loads
TP TD-9607 Rev. 2 (Unit 2)

This temporary procedure provides the installation of Class 1E DC jumpers from SD21 that temporarily reconfigures the vital DC power to DG 2-1, 4-kV Bus G and its associated safeguards relay boards in Modes 5 and 6. This reconfiguration is necessitated due to the unavailability of Battery 22 when it is being replaced in Modes 5 and 6. Normally DC Bus 22 is configured to be connected to DG 2-1, 4-kV Bus G control and associated safeguards relay board. As Battery 22 is being replaced during Modes 5 or 6, DC Bus 21 is to be connected to DG 2-1 via its normal/backup transfer switch and to 4-kV Bus G and associated safeguard relay board via Class 1E jumpers.

Safety Evaluation Summary

This temporary jumper modification only exists during Modes 5 or 6. Powering the 4-kV Bus G controls and associated safeguards relay board from the same backup vital DC source (Battery 21) as the associated DG 2-1 does not increase the probability of occurrence of an accident, nor does it have any impact on the consequences of the accidents described in the FSAR Update. DC Bus 21 loading and Battery 21 sizing evaluations with the added loads have been performed and found to be satisfactory. The jumpering of 4-kV Bus G and associated safeguards relay board to the same backup vital DC source (Battery 21) as DG 2-1 ensures the operability of associated components and provides the greatest flexibility in not only meeting the Technical Specification LCOs but also the outage safety plan requirements. During all times, Technical Specification LCOs and outage safety plan requirements are met or exceeded for the applicable outage window.

12. Temporary Procedure: Implementation of DCP-E-50099 Battery 22 Replacement
TP TD-9609 Rev. 0 (Unit 2)

This temporary procedure implements Battery Bank 22 replacement with LCUN-33 type cells in accordance with DCP-E-50099. Temporarily energizing vital DC Bus 22 and its associated loads from nonvital Battery 27 is required to keep some loads functional during Battery 22 replacement. Battery 27 was chosen as it has ample margin on its respective battery and battery charger. Most loads that are being deenergized and jumpered to nonvital Battery 27 are considered "Functional" and not "Operable."

Safety Evaluation Summary

This work is performed during Modes 5 or 6. Powering DC Bus 22 Class 1E loads from nonvital Battery 27 does not increase the probability of occurrence of an accident, nor does it have any impact on the consequences of the accidents described in the FSAR Update. Should the temporary nonvital DC power from Battery 27 fail to DC Bus 22 or its loads, the effects would not increase the consequence of malfunction on the vital DC Bus 22 or its loads. The clearance and the jumping process is a controlled evolution where individual loads are deenergized and reenergized one at a time. In addition, this procedure contains contingency actions to take should loss of power from Battery 27 occur during the replacement period. The jumpering to a nonvital battery does not take credit for the LCO. It has been determined that having the battery out of service does not violate Technical Specification LCO 3.8.3.2.

E. Equipment Control Guidelines

1. Spent Fuel Pool Cooling System
ECG 13.1 Rev. 3

This Equipment Control Guideline (ECG) revision modifies the completion time of "immediately" for the condition of "One spent fuel cooling pump inoperable with the core not fully loaded in the reactor vessel" by adding a note to the guideline. The note allows a spent fuel pool cooling pump to be taken out of service for the purpose of installing or removing power supply jumpers. Prior to removing a pump from service in accordance with the note, Operations personnel shall ensure close coordination with Maintenance personnel to minimize the amount of time the pump is out of service. Additionally, Operations shall verify that the calculated maximum temperature of the pool, assuming no spent fuel pool cooling for the time the pump is expected to be out of service, will not exceed 175 degrees F.

Safety Evaluation Summary

The spent fuel pool cooling pumps, or loss thereof, are not an initiating event for any accident evaluated. Heat removal from the spent fuel pool is assumed via heat transfer from the surface of the pool. The fuel handling building ventilation system is designed to filter building exhaust in the event that decay heat removal occurs via heat transfer from the pool surface. Makeup to the pool occurs from a borated Class I water source. As stated in the FSAR Update, Section 9.1.3.3.1, the spent fuel pool cooling pumps provide no emergency function during an accident. Therefore, allowing one pump to be out of service in order to install or remove jumpers for its power supply will not increase the probability of occurrence of an accident nor impact the consequences of an accident.

Changing the power requirements of a spent fuel pool cooling pump could only result in the loss of one pump in the event that offsite power were lost. However, this would not impact the integrity of the spent fuel pool or reduce the negative reactivity of the pool. Therefore, the possibility of an accident of a different type than any previously evaluated will not be created.

The loss of both spent fuel pool cooling pumps is assumed to occur in the FSAR Update, Section 9.1.3.3.1. Allowing one pump to be out of service in order to install or remove jumpers for its power supply would, at worst, only result in the loss of cooling if loss of the other pump occurred. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated is not created.

The spent fuel pool cooling pumps are not addressed in any Technical Specifications or the basis for any Technical Specifications. Therefore, the change does not result in a reduction in the margin of safety as defined in the basis for any Technical Specifications.

2. <u>Component Cooling Water Surge Tank Pressurization System</u>
ECG 14.1 Rev. 1

The component cooling water (CCW) surge tank pressurization system provides pressure on the CCW surge tank to maintain CCW pressure above the post-LOCA saturation pressure in the containment fan cooler unit coils. This precludes the possibility of CCW flashing and subsequent water hammer during the design basis LOCA, even if double sequencing occurs. Therefore, the purpose of the ECG revision is to provide administrative controls to ensure that the CCW pressurization system and the CCW system are capable of performing their required functions.

<u>Safety Evaluation Summary</u>

The proposed changes to the ECG would not affect the probability of occurrence of an accident. Compliance with Technical Specification 3.0.3 and an orderly unit shutdown, if needed, is an expected evolution for the Diablo Canyon units.

The loss of the CCW pressurization system could render the CCW system inoperable only if double sequencing were to occur. However, double sequencing occurring with a LOCA is not considered part of the Diablo Canyon licensing basis. Thus, a 7-day allowed outage time is reasonable if only the CCW pressurization system is inoperable. Double sequencing, however, should be considered during the times when the 230-kV system is known to be degraded (as defined by System Operations Instruction O-23).

If the 230-kV system were degraded at the initiation of an event, double sequencing would occur. If the CCW pressurization system were inoperable, the CCW system integrity would be challenged as a result of flashing in the CFCU coils during the resulting double sequencing.

Consistent with the requirements of Technical Specification 3.0.3, the plant should be placed in a safe condition if a safety function, such as the operability of the CCW system, cannot be satisfied. The ECG requires that Technical Specification 3.0.3 be entered for the CCW system if the 230-kV system were degraded and the CCW pressurization system were inoperable.

If the possibility of double sequencing were eliminated, the CCW system would perform its required function regardless of the operability of the pressurization system. Therefore, an allowance to exit Technical Specification 3.0.3 upon restoring the 230-kV system, the CCW pressurization system, or preventing

double sequencing by opening the 4-kV vital bus transfer to startup cutout switches ensures that the CCW system will be capable of performing its required function. Therefore, the consequences of an accident previously evaluated are not increased.

Degraded 230-kV system voltage and the potential for double sequencing is another LOCA scenario where CCW flashing could occur. While the required design basis of the plant only requires consideration of a complete loss of offsite power coincident with the LOCA, the ECG appropriately includes an action statement to declare both trains of CCW inoperable (and enter TS 3.0.3) if both the CCW pressurization system and the 230-kV system are known to be degraded.

The ECG provided administrative controls that ensure availability of the pressurization system to maintain the ability of the CCW system to accommodate a design basis LOCA, with double sequencing. Although short durations of system unavailability are allowed under the ECG allowed outage time, the CCW system remains capable of performing its design function under the re-evaluated LOCA coincident with loss of offsite power conditions.

Under the conditions of a degraded 230-kV system, the ability of the CCW system to accommodate a LOCA and double sequencing cannot be demonstrated. As a result, with the CCW pressurization system out of service and the 230-kV system in a degraded condition, Technical Specification 3.0.3 restrictions will be applied to the CCW system. Under the described conditions, the margin of safety as defined in the basis for any Technical Specification will be maintained.

F. FSAR Update Changes

1. <u>Tornado Effects on Auxiliary Feedwater System</u>
   Section 3.3.2

   This change clarifies the capability of the auxiliary feedwater (AFW) system to resist tornado and tornado-induced effects based on failure analysis. There are no physical modifications or setpoint changes involved. As described in the failure analysis, the AFW piping and valves in the FE and FW plant areas were identified as potentially vulnerable to tornado wind and missile effects. Piping calculations documented that this piping can withstand wind velocities over 300 mph. However, this piping is susceptible to damage due to tornado-induced missiles. On the basis that only one missile can occur at a time, such damage is limited to a single component or train, in a worst-case. Hence, the AFW system functionality is not affected when redundant trains are available.

2. <u>Structural Assessment of Containment Exterior Shell</u>
   Section 3.8.1 and Associated Figures

   This change updates the containment assessment information to more accurately represent the basis used for the structural assessment of the primary containment exterior shell. Several text revisions are included to provide the updated information, including changes on loads due to thermal expansion, liner anchorages, junction of cylinder and base slab, and acceptance criteria for accident conditions. In addition, several figures are deleted and several others are revised to reflect updated information. Updated engineering calculations confirm that the structural integrity of the primary containment is within allowable acceptance criteria limits and no safety margins are impacted.

3. <u>Design Basis of Backup Air/Nitrogen Supply Systems</u>
   Section 9.3.1 and Table 3.9-9

   This change involves the design basis for the backup air/nitrogen supply systems that were added to the facility. An analysis was performed to ensure a complete safety review of the addition. The addition of backup compressed gas systems to air-operated valves does not prevent the valves from performing their intended safety function. Further, this addition supports the ability of the valves to perform their safety functions in the event that the compressed air system is lost due to a seismic event or loss of offsite power.

4. <u>Containment Integrity Analysis</u>
Section 6.2 and Appendix 6.2B

This change incorporates the 1993 containment integrity analysis for loss-of-coolant accidents (LOCAs) and main steam line break (MSLB) accidents in the FSAR Update. The major change is the deletion of Appendix 6.2B, which is replaced by Appendix 6.2C. Appendix 6.2C summarizes Westinghouse analysis results documented in WCAPs 13907 and 13908. Other changes are made to Section 6.2, including tables and figures, to provide proper referencing and to delete obsolete information. The containment integrity analysis was updated to increase analytical margins and to reduce the calculated heat load for the component cooling water system. Several methodology changes were performed to provide these benefits. Changes were made to both the LOCA and MSLB analyses.

The primary change to the LOCA analysis is the use of a new mass and energy release methodology that credits condensation inside reactor coolant system (RCS) piping, hence decreasing steam release and increasing water flow. The primary change to MSLB is crediting the ramp-down of feedwater flow as the feed isolation valve closes. Although the analysis is updated, the methodology is essentially unchanged (except for the inclusion of the NRC-approved mass and energy release model) and the safety limits and technical specification are unaffected.

5. <u>Containment Isolation and Emergency Core Cooling System Valves</u>
Sections 6.2.4, 6.3.2, and Table 6.3-1

These changes revise discrepant statements in the FSAR Update regarding the containment isolation valves and emergency core cooling system (ECCS) valves. The statements involve the minimization of leakage of postaccident radioactive fluid through valve stem packing to the room atmosphere and to the environment. The discrepant statements are not due to design changes, but result from the fact that Diablo Canyon Power Plant (DCPP) was built and is operated differently from the generic description provided in the FSAR Update.

The FSAR Update previously discussed design and operating features that, if implemented, would contribute to meeting NRC leakage criteria (Appendix J). The fact that some of these features are not used and that DCPP has used other design and operating features to meet the criteria does not affect plant safety provided the leakage criteria are met. The changes to the FSAR Update clarify that different features are used but do not affect compliance with these criteria. No unreviewed safety question results from the use of design and operating features different from those originally anticipated provided the criteria are met.

6. Regulatory Guide 1.1 Net Positive Suction Head Margin
Section 6.3.2 and Table 6.3-11

This change revises the static head assumption to be consistent with the guidance of Regulatory Guide (RG) 1.1, and to increase the residual heat removal (RHR) pump runout limit from 4500 gpm to 4900 gpm for the worst-case, post-LOCA alignment. The change maintains and demonstrates that an adequate net positive suction head (NPSH) margin exists as required by RG 1.1. There are locked nuts installed on the flow control valves to limit the maximum pump flow rate to be less than 4900 gpm. The RHR system has been full-flow tested to 4900 gpm to properly set flow control valves and to verify system performance. Containment sump and equipment inside containment are designed for and analyzed to a higher flood water level. Increased RHR pump flow will not impact the post-LOCA ECCS performance because (a) there will be more flow to the core to provide cooling; (b) it will not boost centrifugal charging pumps or safety injection pumps beyond their runout limits since the boosted pressure from the RHR pump decreases as flow increases; (c) RHR pumps have been tested by the vendor to a flow well beyond the 4900 gpm limit; (d) the increased brake horsepower is within the motor capability; and (e) net positive suction head margin exists. This change will maintain RHR pump performance and does not affect other systems that are important to safety. Therefore, no unreviewed safety question exists.

7. Miscellaneous Electrical System Revisions
Sections 8.1, 8.2, 8.3, and 9.5; Associated Tables and Figures; and Appendices 8.3A, 8.3B, and 9.5D

Various changes were made to reflect the existing configuration. These changes are necessary to clarify current plant conditions as described in the FSAR Update to be consistent with current plant conditions, procedures, and analyses. Changes include revisions to reflect the latest stability study performed on the PG&E transmission system; additional information on operator actions required to open the motor-operated disconnect switch; deletion of Appendix 8.3A; clarified diesel generator loading sequence and reason for the time delay based upon existing schematic and logic diagrams; increased the time for pressurizer heaters to transfer power supplies from less than 10 minutes to the design basis value of less than 60 minutes; additional discussion of nonvital uninterruptible power supplies, battery operated lights, and other descriptions of various aspects of the electrical system; additional information in Appendix 8.3B on special cables for pressurizer heaters; additional information in Appendix 9.5D on emergency lighting for fire protection; additional discussion in Section 9.5 on lighting and communications equipment; and several other less significant changes throughout Chapter 8.

The revisions are intended to clarify current plant conditions and do not affect accident analyses as analyzed in the FSAR Update. These revisions do not result in any physical or procedural changes. The revisions have been analyzed to verify that they meet existing design criteria, or are described in design criteria documents or the Technical Specifications, and no unreviewed safety questions exist.

8. Spent Fuel Cask Drop Analysis
Section 9.1.2

This change clarifies the spent fuel cask drop analysis. A statement is added that requires the performance of a more detailed evaluation of the structural effects of a cask drop on spent fuel pool integrity when a specific transfer/shipping cask is selected. The change also clarifies that spent fuel cask movement in the fuel handling building will be governed by procedures that specify the rigging configuration for the particular cask selected has no impact on the environment, the emergency plan, or the security plan. The performance of a more detailed evaluation and the use of cask-specific procedures will not have any effect on safety-related equipment. No other existing systems or components will be affected. A more detailed evaluation and the use of procedures will ensure that the results of a cask drop are no more severe than those presently described based on the current evaluation or spent fuel handling practices. Therefore, no unreviewed safety questions exist.

9. Reactor Refueling Operations
Section 9.1

This revision is made to indicate that the reactor may routinely be either partially or fully off-loaded during refueling to support outage operations; that when performing a full core off-load, the spent fuel pool temperature is maintained below 140°F by administrative controls; and that as a consequence of this temperature limit, an assembly off-load rate greater than that originally assumed in the analysis described in the PG&E Reracking Report is acceptable.

The full core off-load offers a number of safety advantages over the partial core off-load with in-core shuffle due to the following factors: (a) a less complex pattern of fuel movement with less chance for misplacing a fuel assembly within the core; (b) the ability to remove fuel assemblies from the outside of the core inwards, reducing the likelihood of fuel assembly damage during movement; (c) the opportunity to perform a visual inspection of all the fuel assemblies that will be reloaded into the core; and (d) the opening of a core off-load window, during which maintenance and testing activities can be more safely performed on the RCS and RHR systems, and which minimizes operation in the mid-loop configuration. For these reasons, performing a full core off-load is considered a

safer way to perform refueling than a partial core off-load with in-core shuffle, and has hence been adapted as the preferred method of fuel movement.

10. Spent Fuel Pool Cooling Pumps
Section 9.1

This revision clarifies the analysis of the spent fuel pool cooling pump with regard to the power supplies, in that the stand-by pump is temporarily aligned to a nonvital supply during outages or maintenance periods. These pumps have no emergency function during an accident. The loss of these pumps is not an initiating event for any accident evaluated in the FSAR Update. Heat removal from the spent fuel pool is assumed via heat transfer from the surface of the pool. Makeup to the pool occurs from a Class I water source. Consequently, providing for the temporary alignment of the standby spent fuel pool pump to a non-vital power supply does not pose an unreviewed safety question.

11. Auxiliary Saltwater System and Component Cooling Water System Analysis
Section 9.2 and Associated Figures

This revision documents the effect of a revised analysis (in Appendix 6.2C) on the auxiliary saltwater and component cooling water (CCW) systems. The change involves the methodology used to determine the consequences associated with FSAR Update accident analyses, including the effects of malfunction of equipment important to safety. The new LOCA analysis predicts higher CCW temperatures in the post-LOCA recirculation phase due to higher sump water temperatures. This condition has been analyzed and actions have been taken to ensure that the CCW system will support the required equipment to mitigate a design basis LOCA. Compliance with the accident success criteria is maintained, and no unreviewed safety question exists.

12. Chloride and Dissolved Hydrogen Analysis Techniques
Section 9.3

This revision updates the description of chloride and dissolved hydrogen analysis techniques. The analyses are performed via remote grab samples rather than in-line as previously described. The method of sampling does not affect the safety systems and therefore does not pose an unreviewed safety question.

13. HVAC System Descriptions
Sections 9.4 and 12.2

These changes correct the discrepancies identified during the FSAR Update review performed as a result of a nonconformance report. The revisions involve HVAC systems. The changes are intended to reflect current plant conditions

and do not involve physical modifications. The HVAC systems in the plant will not cause the occurrence of an accident as defined in the FSAR Update, and the revisions do not affect the safe operation of the plant or systems required to mitigate an accident. Therefore, no unreviewed safety question exists.

14. <u>Equipment Control Guideline 18.7</u>
Appendix 9.5H

This revision is made to reflect the current Fire Protection Program identified in Equipment Control Guideline (ECG) 18.7. This ECG currently provides the required administrative controls for fire barrier penetrations in fire area boundaries protecting safety related equipment. In order for the ECG to effectively apply to fire area boundaries separating redundant safe shutdown equipment, the ECG was revised to address both safe shutdown and safety-related areas. The changes to the FSAR Update are needed to be consistent with ECG 18.7. The changes clarify the types of fire rated assemblies applicable to the ECG and the acceptable methods of compensatory measures. Any changes to the ECG are reviewed according to procedures to ensure that the Fire Protection Program commitments are maintained and that the ability to safely shut down the plant is not adversely affected. Therefore, the changes to the FSAR Update would not pose an unreviewed safety question.

15. <u>Secondary Chemistry Limits</u>
Section 10.4 and Table 10.4-2

Changes to the secondary chemistry show increased conductivity caused by ethanolomine. Revisions in chemistry limits are needed in this section to reflect current industry standards, and to reflect the use of steam generator blowdown demineralizer usage since it is not normally on-line in current plant practice. Chemistry controls are conservative and the changes are in a conservative direction. No mitigation of accidents is provided by the usage of the steam generator blowdown demineralizer, and no equipment important to safety is affected by the revisions. Therefore, no unreviewed safety question exists.

16. <u>Tritium Concentrations</u>
Section 11.2

This change corrects an inappropriate limit in the FSAR Update regarding tritium concentrations. The previous discussion indicates that the concentrations in the primary coolant are maintained below 1 μCi/cc. The actual analyzed concentrations range from 1 to 2 μCi/cc. Offsite doses from routine effluents remain well below the design objectives of 10 CFR 50, and the basis for acceptance of the liquid radwaste system is not affected. Therefore, no unreviewed safety question exists.

17. <u>Liquid Radwaste System</u>
Table 11.2-10

This change revises the FSAR Update to be consistent with PG&E Design Criteria Memorandum S-19, for the liquid radwaste system. The changes involve additions and revisions to Table 11.2-10 involving the containment, reactor cavity, and RHR pump room sumps; volume of the auxiliary building sump and the miscellaneous equipment drain tank; operating parameters of radwaste equipment; and addition of two radwaste filters to the original three. Information is being provided to reflect plant conditions in more specific detail and no physical changes are made to the plant. The changes affect equipment and parameters that are not evaluated in an accident scenario. Therefore, no unreviewed safety question exists.

18. <u>Gaseous Radwaste System</u>
Table 11.3-1

This change clarifies details in the FSAR Update to be consistent with PG&E Design Criteria Memorandum S-24 for the gaseous radwaste system. The changes involve revisions to Table 11.3-1 to revise the waste gas compressor discharge pressure; revise the surge tank design pressure and add the design temperature; and specify the PG&E pipe specification for the surge tank material. Information is provided to reflect plant conditions in more specific detail and no physical changes are made to the plant. The changes affect equipment and parameters that are not evaluated in an accident scenario. Therefore, no unreviewed safety question exists.

19. <u>Sampling and Monitoring Program</u>
Section 11.6 and Associated Tables

This change clarifies the organizational responsibility and details regarding the sampling and monitoring program. No physical modifications are made to plant systems or equipment, and no equipment important to safety is affected. Only administrative details are changed and the changes do not adversely affect accidents analyzed in the FSAR Update. Therefore, no unreviewed safety question exists.

20. <u>Auxiliary Feedwater System Flow Requirements</u>
Section 15.1 and Associated Tables and Figures

This revision involves a change in the AFW flow requirements. The AFW flowrate in the FSAR Update is based on the flow rate provided in the basis of the Technical Specifications (TS), which had been 440 gpm for the motor-driven pump and 880 gpm for the turbine-driven pump. The basis of the TS has been revised to allow 410 gpm for the motor-driven AFW pumps and 820 gpm for the

turbine-driven pumps. The change to the FSAR Update is necessary to be consistent with the new basis of the TS. The reduced AFW flow rates have been reviewed against accident analysis and other concerns (such as ATWS) by Westinghouse. Reducing the required flow rates provides a wider margin for AFW pump performance during surveillance testing. This change also allows for throttling AFW flow to provide greater margin to steam generator overfill during a tube rupture event. Since the change does not impact normal operation of the plant, there are no FSAR Update accidents that would be more likely to occur due to a reduction in the minimum required AFW flow.

A reduction in the AFW flow does not relate to the initiation of an accident or introduce any new failure mechanisms. There is an effect on steam release in a MSLB outside containment, hence the impact on AFW reduction on environmental qualification of equipment was investigated. The investigation determined that previously analyzed blowdown cases represented greater energy input to equipment compartments. Therefore, no unreviewed safety question exists.

21. <u>Reactor Coolant System Heat Input Assumptions</u>
Table 15.1-1

This revision changes the value of RCS heat input assumed by the software associated with the heat balance procedure. Specifically, the RCS heat input, which is due primarily to RCP operation, is changed from 10 million Btu/hr to approximately 12.2 million Btu/hr per loop. This change corrects an unnecessarily conservative estimate of the RCS heat input. This change does not affect rated thermal power, but does increase the NSSS power slightly. The increase is negligible compared to valve capacities or power uncertainty. The new value allows slightly more core power in reality, but the core power remains limited by the licensing values. Hence, this change has no effect on the procedure or any item that uses an assumed core power licensing limit.

Although no written procedure is changed, the 50.59 review was performed because the RCS heat input assumptions result in a slight increase in core power and causes a change in a value in Table 15.1-1. However, the change to an input parameter of the RCS/secondary calorimetric procedure is not an accident initiator nor does it affect the potential for an accident to occur. The function of this procedure is to benchmark the NIS power range neutron detectors and is not changed. The existing procedure provides no automatic actuations. Existing seismic and environmental qualifications remain valid. The FSAR Update analyses are performed based on a maximum core power rating that is not impacted by this change. Based on these considerations, there is no unreviewed safety question.

22. Spurious Safety Injection Actuation
Section 15.2

This change involves the analysis for spurious safety injection actuation. The previous analysis does not address pressurizer overfill. The new analysis demonstrates that overfill does occur prior to safety injection termination, but the pressurizer safety valves will operate reliably for the fluid conditions that result during the water release period. This section is revised to reference the new spurious safety injection analysis. In addition, operator action is credited for safety injection termination within 16 minutes. The revision references an additional analysis in the FSAR Update. The change does not affect operating procedures or the physical condition of the plant. The change relates to the mitigation of a specific accident and does not impact accident initiation during normal operation. The results of the analysis demonstrate successful event termination without initiating an additional accident.

The analysis does credit a specific time for operators to respond to an inadvertent safety injection signal, and credits pressurizer safety valve water release. Previously, pressurizer overfill was not an analyzed consequence discussed in the FSAR Update. The new analysis biases the input variables in a manner to promote overfill. The conclusion of the new analysis is that overfill will occur, but the pressurizer safety valves are qualified to reseat after safety injection termination. The consequences of fluid flow through the pressurizer safety valves to the pressurizer relief tank are bounded by previous evaluations. Hence, the existing success criteria continue to be maintained. Therefore, there is no unreviewed safety question.

23. Feedwater Break Accident Analysis/Pressurizer Overfill
Section 15.4 and Table 15.4-8

This revision adds further description of the feedline break accident analysis in this section. The results of the existing analysis show pressurizer overfill after 75 minutes. However, overfill will not occur since the predicted time is sufficient to allow operators to take appropriate actions, such as terminating safety injection, depressurizing the secondary side, and establishing normal charging and letdown. Therefore, specific information is added in the FSAR Update to credit operator intervention to prevent pressurizer overfill.

The FSAR Update analysis represents a high degree of conservatism and does not necessarily reflect how plant conditions would be under actual accident conditions. The reliance on operator action to preclude pressurizer overfill after a feedline break does not result in any physical changes or operational changes, and there is no impact on normal operation. The proposed change relates to the mitigation of a specific accident and does not impact accident initiation. The reliance on operator action to prevent pressurizer overfill involves a change in

performing a function. However, the end result, mitigation of a feedline break without pressurizer overfill, is the same as previously analyzed. The operator action is included or implicit in other FSAR Update evaluations. Therefore, there is no unreviewed safety question.

24. Environmental Consequences of Postulated Rupture of Liquid Holdup Tank
Section 15.4

This change revises the analysis of the environmental consequences of a rupture of a liquid holdup tank (LHUT), based on PG&E Calculation N-160. The reanalysis demonstrated that the postulated offsite dose from a ruptured LHUT is less than 0.5 rem. This supports the classification of LHUT as Design Class II, and no seismic requirements are necessary. This analysis is also conservative with respect to the original assumptions for the LHUT documented in the FSAR Update. The reanalysis is added as a reference in this section and is utilized to revise Table 15.5-56 (in another change, described below). No modifications to the plant are involved and no unreviewed safety question exists.

25. Reclassification of the Liquid Holdup Tank
Section 15.5 and Associated Tables

This change revises the analysis of the environmental consequences of a rupture of a LHUT. Changes to the FSAR Update text are involved, along with deletion or revision of several tables. As described in change 24 above, the reanalysis decreases the projected offsite dose and justifies the reclassification of LHUT as Design Class II, nonseismic items. No modifications to the plant are involved and no unreviewed safety question exists.

26. Steam Generator Pressure/Temperature Limitation - Equipment Control Guideline 4.3, Rev. 0
Chapter 16, Table 16.1-1

Technical Specification 3/4.7.2 was relocated into a new Equipment Control Guideline (ECG) as part of a change to the DCPP Operating Licenses (Appendix A), as approved by the NRC in License Amendments 106 (Unit 1) and 105 (Unit 2), dated July 6, 1995. The ECG is identified in a new Table 16.1-1 of the FSAR Update.

Safety Evaluation Summary

There were no changes to the requirements of the relocated Technical Specification. There were no changes to existing plant systems, equipment, or practices. There are no safety implications associated with this administrative change.

27, Flood Protection - Equipment Control Guideline 17.3, Rev. 0
Chapter 16, Table 16.1-1

Technical Specification 3/4.7.13 was relocated into a new ECG as part of a change to the DCPP Operating Licenses (Appendix A), as approved by the NRC in License Amendments 106 (Unit 1) and 105 (Unit 2), dated July 6, 1995. The ECG is identified in a new Table 16.1-1 of the FSAR Update.

Safety Evaluation Summary

There were no changes to requirements of the relocated Technical Specification. There were no changes to existing plant systems, equipment, or practices. There are no safety implications associated with this administrative change.

28. Area Temperature Monitoring - Equipment Control Guideline 23.1, Rev. 0
Chapter 16, Table 16.1-1

Technical Specification 3/4.7.11 was relocated into a new ECG as part of a change to the DCPP Operating Licenses (Appendix A), as approved by the NRC in License Amendments 106 (Unit 1) and 105 (Unit 2), dated July 6, 1995. The ECG is identified in a new Table 16.1-1 of the FSAR Update.

Safety Evaluation Summary

There were no changes to requirements of the relocated Technical Specification. There were no changes to existing plant systems, equipment, or practices. There are no safety implications associated with this administrative change.

29. Sealed Source Contamination - Equipment Control Guideline 39.6, Rev. 0
Chapter 16, Table 16.1-1

Technical Specification 3/4.7.8 was relocated into a new ECG as part of a change to the DCPP Operating Licenses (Appendix A), as approved by the NRC in License Amendments 106 (Unit 1) and 105 (Unit 2), dated July 6, 1995. The ECG is identified in a new Table 16.1-1 of the FSAR Update.

Safety Evaluation Summary

There were no changes to requirements of the relocated Technical Specification. There were no changes to existing plant systems, equipment, or practices. There are no safety implications associated with this administrative change.

[illegible]

30. Snubbers - Equipment Control Guideline 99.1, Rev. 0
Chapter 16, Table 16.1-1

Technical Specification 3/4.7.7 was relocated into a new Equipment Control Guideline (ECG) as part of a change to the DCPP Operating Licenses (Appendix A), as approved by the NRC in License Amendments 106 (Unit 1) and 105 (Unit 2), dated July 6, 1995. The ECG is identified in a new Table 16.1-1 of the FSAR Update.

Safety Evaluation Summary

There were no changes to requirements of the relocated Technical Specification. There were no changes to existing plant systems, equipment, or practices. There are no safety implications associated with this administrative change.