Enclosure
PG&E Letter DCL-98-136

10 CFR 50.59 REPORT OF FACILITY CHANGES, PROCEDURE CHANGES, TESTS, AND EXPERIMENTS

MAY 24, 1996 - MARCH 28, 1998

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

A. FACILITY CHANGES

A. FACILITY CHANGES

A. FACILITY CHANGES

B. TEMPORARY MODIFICATIONS, ELECTRICAL JUMPERS AND LIFTED LEADS, MECHANICAL JUMPERS AND BYPASSES, AND TEST EQUIPMENT

C. PROCEDURE CHANGES

C. PROCEDURE CHANGES

D. TESTS AND EXPERIMENTS

E. EQUIPMENT CONTROL GUIDELINES

F. FSAR UPDATE CHANGES

F. FSAR UPDATE CHANGES

F. FSAR UPDATE CHANGES

G. OTHER CHANGES

G. OTHER CHANGES

A. Facility Changes

1. Restore the Fire Barrier Between Fire Area 4-B and 4-B-2
DCP A-050330, Rev. 0 (Unit 2) (LBIE Log No. 97-045)

The 2-hour fire barrier separating Fire Area 4-B and 4-B-2 is degraded because Pullbox BPG5 is partially embedded on the southside of the 2-hour plaster fire wall. Conduit k6944 is a safe shutdown circuit with a 1-hour 3M fire wrap on the north side of the fire barrier running along the north face of the plaster wall and connecting to Pullbox BPG5. To satisfy the requirement for a 2-hour plaster wall, provide a 2-hour pyrocrete enclosure behind the protuded Pullbox BPG5.

Safety Evaluation Summary

The degraded 2-hour fire rated plaster wall configuration was modified to meet the requirements of 10 CFR 50, Appendix R. This modification is required to maintain separate redundant trains of safe shutdown components, per Section IIIg of Appendix R.

2. Cathodic Protection for ASW Supply Pipelines
DCP C-049169, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-217)

This design change is for installation of a cathodic protection system for the existing Class I auxiliary saltwater (ASW) pipelines and new bypass ASW pipelines near the intake to reduce corrosion of the pipelines.

Safety Evaluation Summary

The cathodic protection system is largely buried, with most components not near the ASW pipes. The modification does not change any system interfaces and has no impact on ASW system capacity to perform its design function. The ASW system is not the cause of any FSAR Update analyzed accidents. Therefore, it is concluded that installation and operation of the cathodic protection system does not involve an unreviewed safety question.

3. Auxiliary Saltwater Bypass
DCP C-049207, Rev. 6 (Units 1 & 2) (LBIE Log No. 96-018)
DCP C-050327, Rev. 0 (Unit 2) (LBIE Log No. 97-179)
DCP C-050327, Rev. 1 (Unit 2) (LBIE Log No. 98-013)
DCP C-050327, Rev. 0 (Unit 2) (LBIE Log No. 97-157)
ASW Piping Bypass Project Letter, Rev. 0 (LBIE Log No. 97-007)

This design change installed a bypass around approximately 800 feet of Unit 1 and 200 feet of Unit 2 auxiliary saltwater (ASW) system piping. The project was initiated due to a concern that localized corrosion was occurring in the portion of

the piping buried below sea level in the tidal zone outside the intake structure. In addition, upgraded flow and temperature instrumentation was installed. The project was installed in phases. The project safety evaluation determined the change did not involve a unreviewed safety question. After interaction with the NRC, PG&E submitted a License Amendment Request (LAR 97-11) to resolve concerns with the analyzed potential for liquefaction in a small area below a portion of the Unit 1 piping. The LAR was later revised to apply only to Unit 1. The NRC has not completed their review of the LAR.

Safety Evaluation Summary

The installation of the bypass piping does not impact the ASW system design basis parameters. The rerouting and increased length of the bypass causes a small reduction of ASW flow, however the design and licensing basis functions of the ASW system are not impacted. The routing of a portion of the Unit 1 piping over an area that may be impacted by seismic induced liquefaction is conservatively included in the design. The LAR was submitted to allow the NRC to review the unreviewed safety question introduced into the Diablo Canyon FSAR Update by the liquefaction issue.

4. Intake Cove Revetment
DCP C-049310, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-015)

This DCP reinforced the rip rap revetment along the north shore of the intake cove, through the pumping of concrete into voids between the existing armor stones. This reinforcement is required in order to protect the soil surrounding and supporting buried auxiliary saltwater (ASW) bypass piping (installed per DCP C-49207) from damage associated with the tsunami-storm wave loading conditions defined in DCM T-9.

Safety Evaluation Summary

No 50.59 safety evaluation was performed for this design change, as the requirements of the safety evaluation screen were met. The safety evaluation was performed for an environmental evaluation as the performance of this work created discharges to the ocean that could impact PG&E's National Pollutant Discharge Elimination System permit. It was determined that the change did not involve an unreviewed environmental question.

5. Provide Pyrocrete Fire Barriers
DCP C-049339, Rev. 0 (Unit 1) (LBIE Log No. 97-160)

This modification involved the design of fire barrier materials, which protect redundant safe shutdown circuits from the effects of a fire. The Pyrocrete maintenance doors at Elevation 85 feet in the 12-kV switchgear room and the

Transite barriers at Elevation 76 feet in the 12-kV cable spreading room will be modified with an additional layer of Pyrocrete material to provide the required 2-hour fire rating.

Safety Evaluation Summary

The Pyrocrete maintenance doors at Elevation 85 feet in the 12-kV switchgear room and the Transite barriers at Elevation 76 feet in the 12-kV cable spreading room will be modified with an additional layer of Pyrocrete material to provide the required 2-hour fire rating to protect one train of redudant circuits for equipment per 10 CFR 50, Appendix R, Section III. G.2.

6. Unit 2 - Yard Pullbox: Install Fire Barrier
DCP C-050405, Rev. 0 (Unit 2) (LBIE Log No. 97-180)

During review of issues associated with yard pull boxes containing safe shutdown circuits, it was discovered that Transite panel separation barriers in Pull Boxes BPO 33, 33A, 33B, 33C, 43, 43A, 43B, and 43C were not qualified fire barriers. The Transite panels were not approved fire barriers. 10 CFR 50, Appendix R, Section III.G.2, requires a 3-hour rated fire barrier to be installed for one train of redundant circuits for equipment that is required to achieve and maintain safe shutdown and is located within the same fire area. As corrective action for Nonconformance Report N0001887, it was necessary to remove the unqualified Transite panels from the pull boxes and replace them with pre-cast 3-hour rated pyrocrete barriers to completely seal and separate the safe shutdown circuits in separate compartments within the pull boxes.

Safety Evaluation Summary

The 10 CFR 50.59 safety evaluation was performed because unqualified Appendix R fire barriers were replaced with qualified tested barriers. Although FSAR Update fire barrier descriptions are not particularly specific, the DCP was considered to be a modification to the fire protection system (FSAR Update Section 9.5). The design change brings fire protection in the Unit 2 yard pull boxes into conformance with NRC regulations and licensing commitments. The Unit 2 yard pull boxes fire barriers are not associated with initiation of any evaluated FSAR Update accident. The design basis accident is a pull box fire that is mitigated, not caused by, the subject barriers. The qualified pull box fire barriers ensure that at least one of the redundant safe shutdown circuits located in the yard pull boxes will be available for safe shutdown following a postulated fire. Combustible loading was not increased. New combustible materials were not added. The qualified pull box fire barriers ensure that a single fire will not affect both safe shutdown trains in the subject yard pull boxes. Installation of the barriers did not raise the possibility of a new equipment malfunction because the

new barriers were fabricated outside the pull boxes and set in place using approved procedures.

7. Install Fire Barriers in Fire Area 20
DCP C-050339, Rev. 0 (Unit 2) (LBIE Log No. 97-014)

This design change modified the design of fire barrier materials that protect redundant shutdown circuits from the effects of a fire. An additional layer of Pyrocrete was added to the Pyrocrete maintenance doors in the 12-kV switchgear room and to the Transite fire barriers in the 12-kV cable spreading room to provide the required 2-hour fire rating. The modification was a required corrective action for Nonconformance Report N0001887.

Safety Evaluation Summary

This 10 CFR 50.59 evaluation was performed because FSAR Update Appendix 9.5A takes credit for the Pyrocrete barriers and Transite panels as providing the fire protection required for Fire Area 20. Prior to this modification, the subject barriers were not qualified. The new fire barriers are qualified by testing. The FSAR Update description was modified to reflect the change. Adding qualified fire barriers to protect redundant safe shutdown circuits does not affect probability, possibility, or consequences of any analyzed or unanalyzed accident or equipment malfunction. No equipment was modified; safety margins were not affected. The change was made to bring the fire protection in Fire Area 20 into conformance with NRC regulations. A two-hour fire barrier is required to meet DCPP's commitment for fire protection in the subject area. The combustible loading in the area is less than one hour.

8. Battery 13 Replacement
DCP E-049297, Rev. 0 (Unit 1) (LBIE Log No. 96-027)

This design change was to replace Battery 13 with a larger capacity battery. The major scope of work involved replacing 60 cells, fabricating a new step rack, modifying existing battery rack end restraints, and replacing feeder cable and inter-rack cables with 6 - #4/0 AWG cable. This design changed increased the battery capacity from 1800 amp-hr to 2320 amp-hr, which restored the both the vital 125-Vdc system and vital 120-V instrument ac system with positive load growth margins.

Safety Evaluation Summary

This design change did not affect the design basis of the 125-Vdc system. The larger capacity did not change/affect the electrical characteristic supplied to any of the 125-Vdc loads or cause any load to be operated outside their design or testing limits. The new Class 1E battery met the original design specifications

for material and qualification (new qualification test was required). The new battery did not impose any new electrical load, and the existing battery charger, dc bus, or dc distribution panel/breakers did not require modification. No new operational or failure modes were introduced nor were there any changes required to the Technical Specification Bases. The larger battery did increase the battery room floor loading and was accounted for and accepted by Civil engineering calculations. It also increased the amount of hazardous material since the larger battery uses about 60 gallons more sulfuric acid (electrolyte); this was evaluated for in the LBIE Environmental Protection Evaluation and is to be accounted for in the DCPP Hazardous Materials Business Plan.

9. <u>Change the Tap Settings for the Vital MCC Transformers</u>
DCP E-049321, Rev. 0 (Unit 1) (LBIE Log No. 96-037)
DCP E-050321, Rev. 0 (Unit 2) (LBIE Log No. 97-125)

Calculation 357-DC evaluated and provided the optimized transformer tap voltage settings for both minimum and maximum system voltage conditions under various modes of operation. Based on the calculation the vital distribution transformers (motor control center) taps were changed. The calculation and changes were a result of adding load tap changing (LTC) 230/12-kV transformers.

<u>Safety Evaluation Summary</u>

The choice of the tap setting is to provide adequate voltage at the terminals of Class 1E equipment under design basis accident conditions. The tap settings are not covered by Technical Specifications, and the safety functions are assured since these tap changes keep the 4-kV and 480-V buses within the design required voltage levels. Technical Specification 3.8.1 states. "... ensures that sufficient power will be available to supply the safety related equipment." The technical specification requirements and their bases' margins of safety are maintained by optimizing the tap settings for these transformers.

10. <u>Replace SUT 11 With New Transformer Equiped With LTC</u>
DCP E-049322, Rev. 0 (Unit 1) (LBIE Log No. 96-042)

This design change replaced the Unit 1, 230/12-kV startup transformer (SUT) 11 with a new transformer that uses an automatic load tap changing (LTC) device to control the voltage. This change resolved the short circuit withstand capability problem of SUT 11 that was discovered following the study that was completed as a result of the Unit 1 auxiliary transformer failure in October 1995. It also helped resolve the voltage problem that existed at the 4-kV and 480-V buses when supplied from the 230-kV startup source, which was not adequate under certain design basis operating scenarios to support the operation of safety-related equipment. This design change also replaced the existing 230-kV

Disconnect Switch 211-1 with a circuit switcher that provides load break capability to allow removal of SUT 11 from service without deenergizing the 230-kV source.

Safety Evaluation Summary

The replacement of SUT 11 with a new transformer equipped with LTC for voltage control does not alter the original configuration of the electrical distribution system. It enhances the capability of the 12-kV and the 4-kV electrical distribution systems to have sufficient voltage for a successful transfer of the plant auxiliary loads to the startup source following a unit trip. This change also eliminates the potential for "double sequencing" of the 4-kV vital loads during an accident by providing adequate voltage to the 4-kV vital buses from the 230-kV source. The new transformer's design exceeds the short circuit capability requirement. So its malfunction is less likely than the old transformer. Malfunction of the LTC feature is the only new failure that was not a consideration for the old transformer. A failure modes and effects evaluation found that the possibility of a malfunction of the LTC is very unlikely and is no different than the possible failures already considered for the transformer. Malfunction of the LTC is monitored in the control room through voltage indication and annunciator alarm. Under the worst case scenario, during an accident, failure of the LTC in the boost position to maintain minimum voltage at the 4-kV vital buses would be detected by the second-level undervoltage relay and the 4-kV vital buses will transfer to the emergency diesel generators as per the original design.

11. Replacement of CFCU Timers
DCP E-049344, Rev. 1 (Unit 1) (LBIE Log No. 97-210)

All containment fan cooler unit (CFCU) timers in Units 1 and 2 were replaced with more accurate digital-type, Agastat DSC timers, along with internal wiring changes in the CFCU control circuits. The primary reason for the replacement was excessive drift in the old timers. The starting logic has also been modified such that CFCUs will auto-start with low speed under auto bus transfer conditions, regardless of the high/low speed control switch position.

Safety Evaluation Summary

The replacement DSC timers will provide equivalent (or better) performance than the existing timers. The starting logic modification makes the low speed start consistent with an existing administrative control. Therefore, this change does not involve an unreviewed safety question.

12. <u>Battery 23 Replacement.</u>
DCP E-050297, Rev. 0 (Unit 2) (LBIE Log No. 97-121)

Vital Battery 23 Replacement DCP/DCN implemented in the Unit 2 eighth refueling outage installed larger battery cells to accommodate increased dc loading from the new 120-Vac instrument uninterruptible power supply (UPS) and provide for future growth.

<u>Safety Evaluation Summary</u>

The replacement of Vital Battery 23 with larger cells increased the dc bus loading capability and supported the new larger UPS loads. There were no increases in the probabilities or consequences of any accidents previously evaluated in the FSAR Update as a result of this change. There were no unreviewed safety questions.

13. <u>On-Line Replacement of Unit 2 SUT 21 and Its Disconnect Switch 211-2</u>
DCP E-050322, Rev. 1 (Unit 2) (LBIE Log No. 97-183)
DCP E-050322, Rev. 1 (Unit 2) (LBIE Log No. 97-134)

The on-line replacement entailed removing the existing Startup Transformer (SUT) 21 and Disconnect Switch 211-2 and installing the new transformer and circuit switcher while Unit 2 was operating at power. During on-line replacement, the standby startup power to the Unit 2 12-kV startup bus was established by closing the 12-kV startup buses tie breaker. With the tie breaker closed, SUT 11 provided offsite power to both the Unit 1 and Unit 2 startup buses.

<u>Safety Evaluation Summary</u>

While the plant is in an on-line replacement configuration using one startup transformer, the design and licensing basis for the DCPP offsite power is not compromised since the shutdown power for Unit 2 in the event of an accident will be supplied by the available Unit 1 startup transformer while providing a standby power source to Unit 1. The operating instructions of Table I outlined in O-23 will be still applicable.

To handle an anticipated dual unit trip under the on-line replacement configuration, additional compensatory measures would be required, i.e., reduced 12-kV bus transfer, operating Morro Bay Power Plant (MBPP) Units 3 or 4 with a minimum voltage of 234 kV maintained at the DCPP 230-kV buses and availability of all six diesel generators to guard against a loss of offsite power. DCPP design basis allows reliance on the diesel generators for dual unit trips, since there is no common initiating event that would cause an accident in both units simultaneously.

Having a reduced 12-kV load transfer on both units provides additional offsite power margin to the vital buses in case of an accident at one unit or dual-unit trip. The reduced transfer does not result in an increase in probability or consequences of an accident.

There is an increased safety benefit and reduced risk in doing the replacement on-line since the plant configuration in Modes 1 through 4 is under much stricter and better control than in Modes 5 and 6.

There is no increased risk of plant operation while on-line replacement is being performed and the margin of safety as defined in the basis for any Technical Specification will be enhanced as a result of a better voltage control of the new transformer through its LTC. By maintaining a higher voltage at the 12-kV startup buses, the voltages at the 4-kV and 480-V busses will have better voltages providing additional margin of safety.

14. Replacement of CFCU Timers
DCP E-050344, Rev. 1 (Unit 2) (LBIE Log No. 97-211)

All containment fan cooler unit (CFCU) timers in Units 1 and 2 were replaced with more accurate digital-type, Agastat DSC timers, along with internal wiring changes in the CFCU control circuits. The primary reason for the replacement was excessive drift in the old timers. The starting logic has also been modified such that CFCUs will auto-start with low speed under auto bus transfer conditions, regardless of the high/low speed control switch position.

Safety Evaluation Summary

The replacement DSC timers will provide equivalent (or better) performance than the existing timers. The starting logic modification makes the low speed start consistent with an existing administrative control. Therefore, this change does not involve an unreviewed safety question.

15. Install Automatic Control for the LTC of SUT 21.
DCP E-050365, Rev. 0 (Unit 2) (LBIE Log No. 97-135)

This design change installed the automatic controls of the load tap changer (LTC) to the newly installed Unit 2 Startup Transformer (SUT) 21 which was used as a fixed tap transformer up to this point. The new SUT 21 and Circuit-Switcher 211-2 were installed in November 1997 under DCP E-50322. The installation of the automatic controls of the LTC put the system configuration for Unit 2 SUT 21 in the same configuration as that of Unit 1 Startup Transformer SUT 11.

Safety Evaluation Summary

The replacement of SUT 21 with a new transformer equipped with LTC for voltage control does not alter the original configuration of the electrical distribution system. It enhances the capability of the 12-kV and the 4-kV electrical distribution systems to have sufficient voltage for a successful transfer of the plant auxiliary loads to the startup source following a unit trip. This change also eliminates the potential for "double sequencing" of the 4-kV vital loads during an accident by providing adequate voltage to the 4-kV vital buses from the 230-kV source. The design of the new transformer exceeds the short circuit capability requirement. So its malfunction is less likely than the old transformer. Malfunction of the LTC feature is the only new failure that was not a consideration for the old transformer. A failure modes and effects evaluation determined that the possibility of a malfunction of the LTC is very unlikely and no different than the possible failures already considered for the transformer. Malfunction of the LTC is monitored in the control room through a voltage indication and annunciator alarm. Under the worst case scenario, during an accident, failure of the LTC in the boost position to maintain minimum voltage at the 4-kV vital buses would be detected by the second level undervoltage relay and the 4-kV vital buses will transfer to the emergency diesel generators as per the original design.

16. Alternate Power Source to Spent Fuel Pool Pumps
DCP E-050381, Rev. 0 (Unit 2) (LBIE Log No. 97-216)

This design change adds alternate Class 1E power sources to spent fuel pool (SFP) Pumps 21 and 22. The alternate Class 1E power source will be available for use during electrical bus outages and maintenance periods and reduce the need for the use of temporary power jumpers to maintain one SFP pump available.

Safety Evaluation Summary

Providing an alternate Class 1E power source to the SFP pumps reduces the need for energized jumpers during outages. Previous jumper installation has caused increased wear and degradation of cables and terminations, as well as personnel hazards while installing the jumpers. Adding the alternate power source does not affect accidents or safety margin and therefore does not involve an unreviewed safety question.

17. Removal of Flow Controllers from Unit 1 AB and FHB HVAC System Supply and Exhaust Inlet and Downgrading Associated Flow Instruments from "Class I" to "Class IC"
DCP H-049326, Rev. 0 (Unit 1) (LBIE Log No. 97-205)
DCP H-050326, Rev. 0 (Unit 2) (LBIE Log No. 97-185)

This design change deleted automatic control of inlet air flow to the auxiliary and fuel handling building ventilation system fans. Controllers modulating inlet vanes of Fans E1, E2, S31, S32, E4, E5, E6, S1, and S2 to maintain predetermined flow from these fans are removed and they are replaced by manual pressure regulators that main inlet vanes at a predetermined (almost open) position. Also the change downgrades associated flow elements (sensors) from Class I to Class IC.

Safety Evaluation Summary

The design change simplifies operation and improves availability and reliability of the Unit 1 auxiliary building and fuel handling building HVAC systems. This change impacts the description in Section 9.4.2.2 of the FSAR Update. The change does not alter design intent and functionality and was determined to not involve an unreviewed safety question. Recommendation to remove the controllers is based on experience with the HVAC system operation.

18. Upgrade Debris Screens to Design Class 1
DCP H-050401, Rev. 0 (Unit 2) (LBIE Log No. 98-007)

The DCPP Operating License allows opening of the 48-in. containment purge valves during power operation. PG&E had committed in Supplement 9 of the Safety Evaluation Report to install debris screens on the containment side of the valves to ensure that debris will not lodge in the valve seat to prevent full closure of the valves in the event of an accident. This debris prevention function is safety related. However, currently, the Q-List, Design Criteria Memorandum (DCM) T-16, and DCM S-23A classify the debris screens as Design Class II, Seismic Category I. Design Change Package H-50401 upgrades the debris screens from Design Class II to Design Class I. This upgrade is required to ensure that the debris screens will perform their safety-related function.

Safety Evaluation Summary

The containment purge valves may be open during plant Modes 1-4 for purging of the containment. In the event of an accident, the valves must be able to fully close to maintain the integrity of the containment. The consequences of an accident evaluated in the FSAR Update is based on full closure of the isolation valves within the predetermined stroke time. This ensures that the off-site 10 CFR 100 dose guideline value is not exceeded. The safety-related function

of the debris screens is to prevent debris generated during an accident from lodging in the valve seat such that it may affect closure of the valves. Upgrading the design classification of the debris screens and associated components to Design Class I will ensure that this safety-related function can be fully met and that the consequences of an accident would not be increased.

19. <u>Met Tower Instrument Upgrade</u>
DCP J-049101, Rev. 1 (Unit 1) (LBIE Log No. 97-189)

This design change replaced the obsolete meteorological instrumentation on both the primary and backup meteorological towers. The supplementary measurement instrumentation is mounted permanently on top of the backup meteorological shack. This design change was implemented when the equipment was removed from service for its bi-annual calibration.

<u>Safety Evaluation Summary</u>

The design change upgrades the meteorological instrumentation to maintain the requirements of Regulatory Guide (RG) 1.97. The met instrumentation is nonsafety related, is used for monitoring purposes only, and is not part of any accident scenarios previously evaluated in the FSAR Update. The meteorological instrumentation does not interface with any equipment important to safety. All requirements required by RG 1.97 are maintained.

20. <u>Safety Parameter Display System Replacement</u>
DCP J-049123, Rev. 0 (Unit 1) (LBIE Log No. 96-005)

This design change removed and replaced the existing Unit 1 emergency response facility data system (ERFDS). The ERFDS/SPDS is being replaced to solve the following problems: (a) software errors exist which result from operational changes or latent software defects in the original B&W software, (b) parts are not available for hardware failures impacting system availability, and (c) ERFDS software changes can only be purchased at great expense from the original vendor.

<u>Safety Evaluation Summary</u>

The SPDS is functionally and spatially incapable of creating design basis events comparable to those evaluated in the FSAR Update. The replacement of the SPDS and the abandonment of the ERFDS tape function and use of the plant process computer function in the Technical Support Center and emergency operations facility do not introduce any new equipment, configurations, or hazards not previously evaluated.

21. Replacement of Plant Vent Gross Gamma Monitors RM/RE-29
DCP J-049193, Rev. 0 (Unit 1) (LBIE Log No. 96-022)
DCP J-050193, Rev. 0 (Unit 2) (LBIE Log No. 96-023)

This design change replaces the entire analog RM-29 radiation monitoring channel with more modern digital equipment. Components included in this change include the local detector/preamp assembly (RE-29) and control room readout module (RM-29). The local indicator (RI-29A) is no longer required and will be permanently removed. This design change also installs a rigging support structure local to the detector assembly to add in disassembly of heavy lead shields during calibration activities.

Safety Evaluation Summary

This design replaces the existing obsolete RM-29 radiation monitor with a more reliable digital radiation monitor having the same range, functionality and peripheral interfaces. Removal of RI-29A (referenced in FSAR Update Chapter 2) is acceptable as it is not required per RG 1.97 and does not impact the capabilities of RM-29. The rigging support will be seismically qualified to meet seismically induced systems interaction requirements. This design change does not affect Technical Specifications, Emergency or Security Plans, Effluents, Environmental Protection, Fire, or Quality Assurance Programs.

22. Connect PGA Panel Alarm and Condenser DP Signal to Control Room
DCP J-049218, Rev. 1 (Unit 1) (LBIE Log No. 96-038)

This design change replaced existing local condenser delta-P pressure indicators on the generator auxiliaries (PGA) panel with indicating transmitters. The transmitter signals are processed by a new local panel, PK011, and sent to the plant process computer (PPC). The PPC provides control room indication of the condenser delta-P signal and alarm capability to the main annunciator system (MAS). Individual alarm signals from the PGA panel were transmitted to the MAS by a new remote multiplexer in PK011.

Before the modification, individual PGA panel alarms and condenser DP indications were available only locally. A grouped alarm from the PGA panel was input to the MAS for display in the control room. On receiving the grouped alarm, an auxiliary operator was dispatched to identify the specific generator alarm. The associated delay could have led to a unit trip or generator damage.

The safety-related portion of the modification installed raceway supports in the auxiliary building.

Safety Evaluation Summary

The safety evaluation was performed because FSAR Update Section 3.10.2.9 described the MAS as seismically qualified. The new remote multiplexer does not require seismic qualification. Information provided to the control room by the nonsafety-related instrumentation in PK011 is not required for safe plant shutdown or to mitigate the effects of an accident. Isolation is provided where needed to prevent a failure in the multiplexer from degrading operation of the MAS, PPC, or main condenser. The FSAR Update was changed to clarify that the nonseismic portion of the MAS will not adversely affect operation of the seismic portion. Verification activities ensured that no rebar was cut or damaged due to addition of the raceway supports. Any penetrations violated for the pulling of cable were resealed per applicable DCPP procedures. There is no impact on the frequency or consequences of any accident or equipment malfunction. The PGA instruments are not part of any TS-required function; there is no impact on any TS safety margin.

23. Reclassify FCV-430, 461, 495, 496 & 601 to Allow Crediting Remote Operation
DCP J-049259, Rev. 0 (Unit 1) (LBIE Log No. 97-116)
DCP J-050259, Rev. 0 (Unit 2) (LBIE Log No.97-117)

This DCP upgrades the motor control loops for the subject valves from Design Class II to Design Class ID by recognizing that they were originally procured, installed, and maintained in accordance with Design Class ID requirements.

Safety Evaluation Summary

The 50.59 evaluation concluded that upgrading the design classification of these motor control loops for these valves does not impact how they will perform their safety function.

24. Addition of "GO" Pushbuttons to SSPS Safeguards Test Cabinets
DCP J-049298, Rev. 0 (Unit 1) (LBIE Log No. 96-021)
DCP J-050298, Rev. 0 (Unit 2) (LBIE Log No. 97-123)

This design change adds two "GO" push buttons to each of the safeguards test cabinets in the solid-state protection system. This allows testing the steam generator main feedwater supply valves FCV-510/1510, 520/1520, 530/1530, and 540/1540. This design change was implemented during the eighth refueling outages for Units 1 and 2.

Safety Evaluation Summary

The new test pushbuttons do not affect protection circuits. The indirect safety function of the pushbuttons only affects the testing should the pushbutton

contact fail open. The added pushbuttons are Class I devices and are not expected to contribute to the evaluated adverse condition. There is no change in the way the steam generator main feedwater supply/bypass valves operate. The change does not degrade or prevent feedwater isolation. Failure of the test pushbuttons does not affect normal or accident operation of the valves.

25. <u>Modify SCMM Annunciator Alarm</u>
DCP J-049302, Rev. 0 (Unit 1) (LBIE Log No. 96-001)

This change interlocks the sub-cooled margin monitor (SCMM) lo-margin alarm with an existing reactor power permissive, P-10, to maintain the alarm during the appropriate low power operation modes. This alleviates the nuisance alarm at normal power operational modes.

<u>Safety Evaluation Summary</u>

This design change adds another function associated with reactor power permissive P-10. This will result in a revision to FSAR Update Table 7.2-2, "Protection System Interlocks." This wiring change does not affect the Emergency Plan, nor does it affect the Security Plan. The design change will not impact the Technical Specifications. The applicable annunciator response procedure will no longer apply during normal plant operation modes above 10 percent reactor power. This design change will not result in a test, experiment, condition or configuration that will affect the operation of the plant.

26. <u>Utilize Gamma-Metrics Post Accident Monitors as Alternate Source Ranges</u>
DCP J-049320, Rev. 0 (Unit 1) (LBIE Log No. 97-033)

This design change added continuous visual indication in the control room for the post accident neutron flux monitors to be used as additional source range channels during Mode 6. The indication is provided by connecting an isolated output of the Gamma-Metrics monitors to the plant process computer. This change was implemented during the Unit 1 eighth refueling outage.

<u>Safety Evaluation Summary</u>

The Gamma-Metrics post-accident neutron monitors provide the same level of quality assurance, redundancy, and necessary display range as the normal source range monitors. Because they do not have alarm and audio circuit capability, one normal source range channel must remain operable. The additional channels are used for indication only during Mode 6. The additional channels provide no control or protective functions. Should either operable channel (i.e., normal source range or the Gamma-Metric channels) fail, the actions specified by Technical Specification 3.9.2 will be taken.

27. Replace Rod Insertion Recorder YR-412 With PPC Recorders
DCP J-049346, Rev. 0 (Unit 1) (LBIE Log No. 97-166)

Replace Class II rod insertion recorder (YR-412) with two new miniature recorders that will be used for a different function. The new miniature recorders (YR-800 and YR-801) will be electrically connected to the plant process computer and used as required by the operators. The new recorders will be installed at the same location as the old recorder on Control Console CC1. The function of the new recorders is Class II. This DCP will be implemented during the Unit 1 ninth refueling outage.

Safety Evaluation Summary

The associated instrument recorders and electrical components have a non-safety related function (Class II). The recorders are used for monitoring purposes only. They are not required for the safe shutdown of the plant. The new recorders will be mounted on Control Console CC1, which is a safety-related panel. The new recorders will be seismically mounted. These recorders do not have any impact on the rod control function and do not contribute to the effects of any inadvertent control rod bank withdrawal or control rod ejection.

28. Control Room Shift Foreman Workstation Modifications Phase 2
DCP J-049353, Rev. 0 (Unit 1) (LBIE Log No. 97-168)
DCP J-050353, Rev. 0 (Unit 2) (LBIE Log No. 97-169)

This modification installed a permanent workstation for the Shift Foreman in the Unit 1 main control room. The former Unit 1 Shift Foreman's office was modified to provide office space for the Shift Supervisor and the Assistant Shift Foreman. The permanent workstation in the primary control room area provides the Shift Foreman with a clear "command and control" presence in the main control room, with good visual and audible access to control room operators and contact with plant operation.

This modification addressed recent INPO and NRC criticism of the former Shift Foreman/Control Room arrangement. Previously, the on-duty Shift Foreman was located in an office area adjoining the main control room. That location met the requirement for being within the control room isolation boundary (Reference NUREG-0700, Rev. 0, Guideline 6.1.1.6.a), but did not provide the preferred "good visual and voice contact with the primary operational area." The new configuration enhanced the Shift Foreman's ability to oversee plant operations, and to maintain a more formal and professional atmosphere in the control room.

Safety Evaluation Summary

No plant equipment was modified, so that new or different accidents or malfunctions were not introduced, and the frequency of analyzed accidents was not increased. There was no change to any Technical Specification safety margin. The modifications were performed with the plant at power. Most of the work was away from the control boards and outside the Control Operator area to reduce the possibility of distractions or accidental equipment actuations that could cause an accident or impair the ability of the operator to respond to an accident or event. During the modifications, the Shift Foreman was continuously provided with all information necessary to respond to plant evolutions or events so that the potential consequences were not affected.

Phase 1 of this work (DCP J-49351) temporarily relocated the Shift Foreman to the Shift Control Operator area. Control room drawings were relocated from the existing cabinet in the center aisle to an area behind the vertical boards. Access through the center aisle was restricted. These factors did not limit the ability of control room personnel to respond appropriately. The limited access areas of the control room were accessible as required. Drawing relocation caused no significant personnel response delay.

Phase 2 removed the temporary Shift Foreman workstation and installed the permanent workstation. The modification did not affect any systems, structures, or components that are relied upon to mitigate accidents. Improved physical presence enhanced the ability of the Shift Foreman to exercise command and control. The more formal and professional control room atmosphere strengthened the ability of the control room crew to respond to normal plant evolutions as well as to the potential accidents and events evaluated in the FSAR Update. The new workstation location did not significantly affect control room personnel access.

29. Replace RWST Range Code 6 Rosemount Level Transmitters
DCP J-049363, Rev. 0 (Unit 1) (LBIE Log No. 97-037)

This change will replace the existing 2-LT-920 & 921, Rosemount Model 1153HD6RC transmitters, with Rosemount Model 1153HD5RC transmitters. This change will decrease the instrument and channel uncertainty and increase the minimum indicated refueling water storage tank (RWST) volume at the low level alarm to greater than the 120,650 gallons of RWST volume assumed in Table 6.3-5 of the FSAR Update.

Safety Evaluation Summary

Increased accuracy of the RWST level instruments does not increase the probability of accidents. The Emergency Plan is not affected by the RWST

accuracy. The accuracy of the new transmitters exceeds the minimum accuracy required by the Technical Specifications. The Security Plan is unaffected by the accuracy of these transmitters.

30. Safety Parameter Display System (SPDS) Replacement
DCP J-050123, Rev. 0 (Unit 2) (LBIE Log No. 96-035)

This modification removed and replaced the original Technical Support Center, Emergency Operations Facility (EOF) , and the control room emergency response facility data system (ERFDS)/ SPDS hardware supplied by Babcock & Wilcox (B&W) The original ERFDS functions are now divided between two systems:

- A new computer system provided by this design change provides color graphic SPDS displays

- The ERFDS "Data Recording and Recall" functions used for post-trip review are now performed by the Plant Process Computer (PPC)

The ERFDS was replaced to solve the following problems:

Software errors resulting from operational changes or latent software defects in the original B&W design required operators to implement workarounds due to errors in the critical safety function status tree displays. Such workarounds inhibit or adversely affect the operators' ability to respond effectively to an emergency or a plant transient situation. Hardware failures caused by aging and obsolete components occurred with sufficient frequency that system availability was being adversely affected. In many cases direct replacement parts were not available. The original SPDS was implemented in firmware. PG&E did not possess the development tools needed to make changes. Changes could be purchased only from the original vendor at substantial cost and long lead-time. Even minor changes such as scaling limits or engineering units required an expensive firmware replacement.

Safety Evaluation Summary

The SPDS, PPC, and EOF/TSC activities and functions cannot initiate any accidents or cause any equipment malfunctions or failures. Similarly, these activities and functions cannot affect any Technical Specification safety margins.

However, the SPDS provides information to control room, TSC, and EOF personnel to aid in the development of accident evaluations and responses, and in making decisions regarding protection of the health and safety of the public. If the SPDS displays do not accurately reflect the plant configuration, assessment and response functions by operations and management may be delayed or

degraded. The replacement ERFDS/SPDS is designed to facilitate maintenance of its displays to maintain fidelity to an evolving plant configuration. Thus, increased consequences due to inaccurate SPDS displays are not a concern.

31. Connect Condenser Delta-P Signals and Main Generator Auxiliary Panel (PGA Panel) Alarms to the Control Room
DCP J-050218, Rev. 0 (Unit 2) (LBIE Log No. 97-136)

The purpose of this design change is twofold: (1) Permanently connect the condenser delta-P (DP) signals to the plant process computer (PPC) via a new remote multiplexer panel, PK011. By having the condenser DP signals in the control room on the PPC, operators can have early warning of an upward trend by using the PPCs variable alarm capability. (2) Connect the main generator auxiliary alarms (panel PGA) to the Control Room via the same remote multiplexer panel, PK011, used in Purpose 1. The existing Rochester alarm system in panel PGA is replaced with a Ronan supplied lampbox (PK21), which is driven by the main annunciator system (MAS) via isolated data links. The purpose is to provide the operators with individual alarms (vs grouped or general alarms) associated with the main generator in order to promote timely response to system troubles. In addition, the 20+ year old Rochester annunciator system is replaced.

Safety Evaluation Summary

The safety evaluation deals mainly with Purpose 2 above. The condenser DP connection to the control room (Purpose 1) did not require a revision to the FSAR Update. The connection of the PGA panel to the MAS required a clarification to the FSAR Update statement that the MAS is seismically qualified. This design change added a paragraph to the FSAR Update to clarify that the main generator alarms connected to the MAS via a remote multiplexer are not seismically qualified; however, these connections are isolated by qualified means. There is no failure mechanism of the data link, remote multiplexer, or remote visual annunciator drivers that can adversely impact the function of the MAS following an earthquake. The main generator alarms provided by the remote multiplexers are Design Class II and are not needed to maintain the plant in a safe shutdown condition or to mitigate the consequences of seismic events.

The existing text was clarified by making two additional minor changes that did not impact the conclusion: (1) The MAS is seismically qualified to remain functional after an earthquake. (2) The alarms associated with the main generator are not seismically qualified, but will not adversely impact the system function following an earthquake. The main generator alarms are not needed to mitigate the consequences of seismic events.

32. Modify SCMM Annunciator Alarm
DCP J-050302, Rev. 0 (Unit 2) (LBIE Log No. 96-002)

This change interlocks the subcooled margin monitor (SCMM) lo-margin alarm with an existing reactor power permissive, P-10, to maintain the alarm during the appropriate operational modes, low power operations. This alleviates the nuisance alarm at normal power operational modes.

Safety Evaluation Summary

This design change adds another function associated with reactor power permissive P-10. This will result in a revision to the FSAR Update Table 7.2-2, Protection System Interlocks. This wiring change does not affect the Emergency Plan, or the Security Plan. The design change will not impact the Technical Specifications. The applicable annunciator response procedure will no longer apply during normal plant operation modes above 10 percent reactor power. This design change will not result in a test, experiment, condition or configuration that will affect the operation of the plant.

33. Replace RWST Range Code 6 Rosemount Level Transmitters
DCP J-050363, Rev. 0 (Unit 2) (LBIE Log No. 97-036)

This change will replace the existing 2-LT-920 & 921, Rosemount Model 1153HD6RC transmitters, with Rosemount Model 1153HD5RC transmitters. This change will decrease the instrument and channel uncertainty and increase the minimum indicated refueling water storage tank (RWST) volume at the low level alarm to greater than the 120,650 gallons of RWST volume assumed in Table 6.3-5 of the FSAR Update.

Safety Evaluation Summary

Increased accuracy of the RWST level instruments does not increase the probability of accidents. The Emergency Plan is not affected by the RWST accuracy. The accuracy of the new transmitters exceeds the minimum accuracy required by the Technical Specifications. The Security Plan is unaffected by the accuracy of these transmitters.

34. Canopy Seal Clamp Assemblies at Spare CRDM Nozzles
MMP M000036-1, Rev. 1 (Unit 2) (LBIE Log No. 98-028)

This Maintenance Modification Package (MMP) is created to allow for installation of canopy seal clamp assemblies (CSCAs) and dummy can adapters onto canopy seal welds on the reactor vessel closure head CRDM penetrations. The CSCAs are designed to encapsulate defective canopy seal welds, functioning as

an additional barrier to prevent migration of reactor coolant from a weld defect onto the carbon steel reactor vessel closure head.

Safety Evaluation Summary

The CSCAs are safety-related components that are designed and fabricated for reactor coolant system pressure, temperature and loading conditions in accordance with applicable ASME Codes, and interfacing instructions for the CRDM nozzles. Installation of the CSCAs will be in accordance with DCPP's ASME Section XI Program Plan and controlled with approved plant maintenance procedures.

The CSCA serves as a backup device to prevent the leakage of reactor coolant through a defective canopy seal weld from corroding the carbon steel reactor vessel closure head. In this capacity, the CSCA functions to reduce leakage from the defective canopy seal weld and thus prevent the leakage from increasing. Leakage, if present, from the CSCA will be detected by the leakage detection method. Any leakage attributed to the clamp will be classified as "unidentified" to which a 1 gpm limit required by the Technical Specifications will apply. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

35. Steam Generator Mechanical Plugging
MMP M000043-1, Rev. 2 (Units 1 & 2) (LBIE Log No. 98-021)

This change authorizes the mechanical plugging of steam generator tubes that have been identified for removal from service as a result of tube inspections. The repair consists of installing erosion/corrosion resistant mechanical rolled plugs at the steam generator tubesheet.

Safety Evaluation Summary

The integrity of the reactor coolant system (RCS) and of the steam generators is maintained by the installation of these plugs. There is no change in design or functions of the steam generators. The integrity of the plugs is assured by the qualification of the process used to install the plugs and by evaluations to confirm the design will perform the intended function. Similar mechanical plugs are already in service in the steam generators.

The plugs are installed in a way compatible with the overall integrity of the tubesheet. These modifications only affect localized passive structural components. This change does not authorize the removal and plugging of tubes in excess of the present limit of 15 percent of the plugged steam generator tubes. This change does not result in a reduction in the margin of safety as defined in the Technical Specification bases.

36. Installation of Framatome Weld Plugs in Steam Generator Tubes
MMP M000044-1, Rev. 2 (Units 1 & 2) (LBIE Log No. 97-076)

This design change allows installation of Framatome weld plugs in defective steam generator tubes at DCPP Units 1 and 2.

Safety Evaluation Summary

Installation of a welded plug to remove defective tubes from service maintains the integrity of the reactor coolant pressure boundary for all normal and postulated accident conditions. The weld plug material and weld filler material used for installation are compatible with the tube/tubesheet and are not susceptible to degradation that caused the tube to become defective.

37. Steam Generator Tube Pull (FTI)
MMP M000055-1, Rev. 1 (Units 1 & 2) (LBIE Log No. 98-022)

This change authorizes the removal of steam generator (SG) tubes from each of the SGs. The removed SG tubes will provide samples for visual inspection as well as provide samples for laboratory examination and analysis which can aid in better understanding of tube degradation and failure mechanisms. Also, the results will provide a direct correlation between the indications and eddy current test results.

Safety Evaluation Summary

The process employed to remove the tube segments is designed and procedurally controlled to prevent contact with adjacent tubes. Further, the tube remnants remaining in the SG have been analyzed to show that no compromising contacts with adjacent tubes will occur during normal operations and accident conditions. With removal and plugging of some tubes per this MMP, the total number of plugged tubes in the SGs will still be a small fraction of the total tubes.

The removal of flow area by plugging still maintains the circulation capability of the loops well above that required and/or assumed in plant analyses. This change does not compromise the operability of the SGs including the flow and heat transfer capability and pressure boundary integrity during normal operation and postulated accidents. Therefore, this change does not result in a reduction in the margin of safety as defined in the Technical Specification bases.

38. Installation of Framatome U-Bend Stabilizer in Steam Generator Tubes
MMP M000057-1, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-090)

This design change allows installation of Framatome U-bend stabilizers in defective steam generator (SG) tubes requiring stabilization.

Safety Evaluation Summary

Installation of Framatome U-bend stabilizers in defective SG tubes, in conjunction with tube plugging, maintains adjacent SG tube structural and leakage integrity by preventing damage to adjacent tubes during normal and accident flow induced vibration loading. Stabilization prevents the possibility of a tube section becoming a loose part in the secondary system. The integrity of the tube plugs is maintained. The stabilizer material is not susceptible to degradation.

39. Determinate Defective Pressurizer Heaters
MMP M000058, Rev. 2 (Unit 1) (LBIE Log No. 97-197)
MMP M000058, Rev. 1 (Unit 1) (LBIE Log No. 97-027)
MMP M000059, Rev. 2 (Unit 2) (LBIE Log No. 97-198)

This Maintenance Modification Package (MMP) allows pressurizer heaters that have failed to be disconnected. This allows other heaters that are fed from the same circuit breaker to be returned to service. The initial pressurizer heater capacity was 1800 kW. This MMP allows failed heaters to be disconnected as long as the connected capacity is at least 1340 kW with Heater Groups 1 and 4 each having at least 276 kW connected and Groups 3 and 4 each having at least 345 kW connected.

Safety Evaluation Summary

The pressurizer heaters are nonsafety-related. The purpose of the pressurizer heaters is to control pressurizer pressure during heat up and power operation and to support natural circulation of the reactor coolant system during the loss of offsite power. 150 kW is required to support natural circulation. Technical Specifications define the minimum pressurizer heater capacity as 150 kW from two groups which can be supplied by vital emergency power. This MMP maintains this 150 kW by limiting the number of heaters that can be disconnected. Therefore, all margins of safety implicit in this Technical Specification requirement are maintained by this MMP.

40. Replace 4-kv Potential Transformer Primary Fuse
MMP M000066-1, Rev. 1 (Unit 2) (LBIE Log No. 98-012)

Licensee Event Report (LER) 1-97-010-01, "Unplanned Start of Diesel Generator 1-1 Due to a 4160-V Bus H Startup Feeder Phase Potential Transformer Opened Fuse," discusses the event, root cause and possible upgrade of the primary fuse from 1/2 ampere to 1 ampere. Maintenance Modification Package M000066, Rev.1, allows the replacement of Units 1 and 2, 4 kV vital and nonvital potential transformer primary fuses. The replacement fuses have a 1 ampere current rating in place of the existing 1/2 ampere rating. While the 1/2 ampere rating is acceptable, the 1 ampere rating increases the inrush current margin to allow more reliable operation. The replacement fuses have the same physical characteristics as the existing fuses.

Safety Evaluation Summary

The plant configuration and operational logic remains unchanged.
The failure analysis concluded that the most likely cause of failure was accumulated fuse element degradation due to current surges on the fuse over the life of the plant. Increasing the replacement fuse inrush capability implements the analysis recommendation. The safety function of the 4-kV potential transformer is to provide a signal, for control and instrumentation, which is proportional to the voltage of the bus or feeder. The safety function of the primary fuse is to carry load/inrush current and provide short circuit protection to the potential transformer.

The replacement fuse maintains coordination with existing fuses and relays, is safety related, and meets the requirements of PG&E Design Class I and IEEE 308 Class IE. Since the 1 ampere replacement fuse has three times the inrush capability of the 1/2 ampere existing fuse, the probability of occurrence of a fuse blow malfunction during operational transients is decreased, and the availability of potential transformers to perform safety related instrumentation and control functions is increased. The 4-kV primary fuse has no effect on any radiation barrier or offsite dose..

41. Replacement of Diesel Fuel Oil Tank 0-1
DCP M-049160, Rev. 0 (Units 1 & 2) (LBIE Log No. 96-004)

This modification consists of replacing the 40,000 gal. single-walled diesel generator fuel oil tank 0-1 and associated piping up to the pump vaults with a new 50,000 gal. double walled tank and new piping. The design change package also installed a leak detection system for the tank and piping. The overall function of the tank and piping remained the same. The purpose for the replacement was to meet the new California Code of Regulations.

Safety Evaluation Summary

The removal of one of the two diesel fuel oil storage tanks from service has been reviewed by the NRC. To permit replacement of the tanks, the NRC has issued License Amendments 108 and 109 to permit operation of the plant for up to 120 days with a single operable diesel fuel oil tank. The fuel supply to each of the diesel generators is being preserved for the duration of the construction activities required to implement the design change. Construction procedures and work plans assure that the function of the 0-2 diesel fuel oil supply system and the capability of performing that function are not altered during the diesel fuel oil tank 0-1 replacement activities. The final configuration of the diesel fuel oil system following the implementation of the design change assures that the system design, function, and method of performing its function are unchanged or enhanced.

42. Requalification of the CCWS for a Maximum CCW Post-Accident Supply Temperature of 140°F
DCP M-049291, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-158)
DCP M-049291, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-077)

This design change package (DCP) established a new postaccident temperature limit profile for the component cooling water system (CCWS). This was achieved by upgrading individual components' temperature qualifications where necessary and revising the associated design and licensing documentation. This DCP was performed to document previously unrecognized margin between the postaccident CCW supply temperature profile and the documented CCWS equipment temperature limitations.

Safety Evaluation Summary

The bases for the 50.59 conclusions are (1) that raising the qualified post-accident system temperature limit does not effect how the system will actually respond to an accident, and (2) the new elevated temperature limits do not affect the capability any CCW (or related system) equipment to perform its safety function.

43. Removal of Halon from SSPS Rooms
DCP M-049295, Rev. 0 (Unit 1) (LBIE Log No. 96-010)
DCP M-050925 Rev. 0 (Unit 2) (LBIE Log No. 96-011)

These design changes removed the Halon fire suppression system from the Unit 1 and Unit 2 solid-state protection system (SSPS) rooms. It was difficult to maintain the leak-tightness of the ceiling of these rooms as required to maintain the Halon concentration in the event of a fire. In addition, the release of Halon to the atmosphere has adverse environmental consequences, and it is no longer

commercially available. Since the system is not required to satisfy Appendix R safe shutdown requirements, it was not replaced with an alternative fire suppression system. However, the existing fire detection system and alarms were left in place.

Safety Evaluation Summary

The absence of the Halon system has no effect on the probability of a fire occurring in the SSPS rooms. The removal of the system does not increase the analyzed consequences of a fire in the SSPS rooms because the current analysis assumes the loss of an SSPS train as a result of a fire. Removal of the Halon suppression system does not impact the ability to achieve and maintain safe shutdown of the plant; once the reactor is tripped, the equipment in the SSPS room is no longer required to maintain safe shutdown. Manual actions and redundant safe shutdown components, not the Halon system, are credited for mitigating the effects of a fire in this area. The operation of the SSPS computers will not be affected by the change, and the consequences of a fire in this area are unchanged from those already evaluated.

44. CCP 1-1 and 1-2 Gear Oil Cooler Replacement
DCP M-049312, Rev. 0 (Unit 1) (LBIE Log No. 96-014)
DCP M-050312, Rev. 0 (Unit 2) (LBIE Log No. 96-015)

This design change installed new centrifugal charging pump (CCP) gear oil coolers on the 1-1 and 1-2 CCPs to enhance heat transfer and raise temperature qualification.

Safety Evaluation Summary

This 50.59 evaluation concludes that there is no unreviewed safety question concerning replacement of the coolers. This conclusion is reached because (1) the coolers transfer as much (or more) heat from the CCP gear oil, enhancing the CCP's ability to perform its design function, and (2) the coolers are installed in the same configuration and to the same design qualification as the previous coolers.

45. CCW-1-TCV-130 Replacement (HOT TAP)
DCP M049319 (Units 1 & 2) (LBIE Log No. 97-089)

See LBIE 97-046 (Procedure MP I-38-M.1, Units 1 & 2, Rev.0 in "Procedures" section of report). This was a revision to LBIE Log No. 97-046 that changed the mode that work could be conducted. The mode was change from 0 to Modes 5, 6, or 0.

Safety Evaluation Summary

This change only applies to Mode 6 with fuel removed from the reactor vessel. Therefore, because the reactor coolant system is still depressurized, there is no possibility the change involves an unreviewed safety question (see LBIE Log No. 97-046).

46. Revise Design Basis for ASW Pump Motors
DCP M-049385, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-137)

In 1988, the auxiliary saltwater (ASW) pump impellers were replaced with those of larger diameter under Design Change Package (DCP) M-39834. The increase in impeller size has caused the pump motor to operate beyond its nameplate rating (400 hp). This condition (450 hp) of operation was evaluated by plant engineering and accepted. Based on a subsequent engineering Calculation M-854 and its supporting test data, the power supplied by the motor, while operating in a single pump/two heat exchanger configuration, can be as high as 465 hp (conservatism included).

PG&E electrical engineering and Westinghouse (the motor manufacturer) has evaluated that the motors are capable of operating at 465 hp without exceeding their design limits. This evaluation was documented in two engineering memos and a Westinghouse letter. DCP M-049385 is used to accept the extended rating (465 hp) as the new limit for the ASW pump motor operation and to revise the Design Criteria Memorandum S-17B, as well as the FSAR Update sections.

Safety Evaluation Summary

There are no known impacts on the equipment important to safety for accident events due to the ASW pump motor operating in a higher rating up to 465 hp. The increase in ASW pump motor hp can cause a higher consumption in diesel fuel and can affect the adequacy of diesel fuel inventory. Such impacts have been evaluated in engineering Calculation M-786, Rev. 8, and the new hp rating (465 hp) has been used to establish the minimum required fuel storage to meet the license bases.

The increase in ASW pump motor hp may affect diesel loading and loading sequence, motor protective relay setpoints, feeder cable ampacity and voltage drop, and the motor stator temperature rise. These issues have been addressed by electrical engineering and Westinghouse (the motor manufacturer) and concluded that the ASW motors are capable of operating at 465 hp without exceeding the design limits for the motors and the diesel generators. This evaluation was documented in engineering memos. The increase in ASW pump motor hp will affect the internal thermal load for the pump vaults. This increase has been analyzed by engineering and documented in HVAC Calculation 82-6,

Rev. 5. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the FSAR Update will not be increased.

47. Revised Peak CCW Temperature Following a Design Basis Accident
DCP M-049386, (Units 1 & 2) (LBIE Log No. 98-070)

The limiting component cooling water (CCW) temperature transients following a design basis accident (DBA) have been reevaluated by Westinghouse in WCAP-14282, Revision 1, dated December 1997. Revised WCAP-14282 captures previous CCW heatup evaluations contained in several documents and incorporates the latest design input while using the same methodology used in past analyses.

Safety Evaluation Summary

The evaluation specifically addressed the following topics: (1) incorporation of WCAP-14282, Rev. 1, into the licensing and design bases, (2) establishment of an elevated ultimate heat sink temperature limit of 70°F, (3) revision of the normal maximum operating CCW temperature from 120°F to 80°F for Modes 1-3 and 95°F for Modes 4-5, (4) a revised bases for operation of two RHR trains of CCW/ASW in the cold leg recirculation phase, (5) a revised bases for the CCW heat exchanger saltwater inlet valve (1/2-FCV-602 and -603) required 8-hour hold time, and (6) clarification that during post-LOCA split-train operation, operator action is required to recover from specific active failure scenarios. None of these topics involve physical changes to the plant. The evaluation concluded that the proposed changes do not involve an unreviewed safety question.

48. EDG Rooms - CO_2 Manual Actuation Switches: Relocation
DCP M-050366, Rev. 0 (Unit 2) (LBIE Log No. 97-186)

The emergency diesel generator CO_2 manual actuation switches were moved from the south wall of the turbine building to a location outside the diesel generator rooms in the corridor. The relocation was required because the switches were originally located in Pyrocrete boxes that did not conform to a tested configuration. Offsite power circuits were located in vicinity of the switches. A fire in the turbine building had the potential to disable the diesel generators and damage the offsite power circuits.

Safety Evaluation Summary

This 10 CFR 50.59 safety evaluation was performed because FSAR Update Chapter 9, Appendix 9.5A, specifically stated that the switches are located in Fire Zone 19-A, and are enclosed in Pyrocrete to prevent hot shorts. The switch relocation affected the description of Fire Zone 19-A. Ability to achieve and

maintain safe shutdown depends on availability of power to the equipment required for safe shutdown. Relocating the CO_2 manual actuation switches provides increased protection because the switches are now separated from a fire in the turbine building by 2-hour fire barriers. The CO_2 manual actuation switches are not associated with initiation of any accident. The increased separation from potential turbine building fires enhances the ability to protect the diesel generators and reduces the potential for hot shorts to impair operation of the diesel generators. Probability and consequences of accidents or equipment malfunctions are not increased.

49. Convert CCP 1-1 to 3rd Generation Seal Configuration
DCP N-049231, Rev. 0 (Unit 1) (LBIE Log No. 96-016)

The DCP changed/replaced the pump case and internal assembly for Centrifugal Charging Pump (CCP) 1-1 with a like-for-like pump case and internal assembly that has been equipped with the 3rd generation seal configuration. The old seal design (first generation) was a multicomponent assembly requiring external cooling by CCW. Although no known problem is associated with maintenance at DCPP for the 1st generation seals, the conversion to 3rd generation was done as an enhancement, which could increase pump availability.

Safety Evaluation Summary

The replacement of CCP 1-1 with the pump casing and internal assembly from CCP 2-1 is considered a like-for-like replacement. The capability of the CVCS system to meet the functional requirements of the accident analysis is unaffected by this change.

The new 3rd generation mechanical seals meet or exceed the original mechanical seal requirements, with the exception that external cooling is not required. Seal life is extended as a result of the one piece seal sleeve/ pumping ring design. Therefore, the availability of the CCP is increased. Also, because CCW is no longer required for cooling, one of the failure modes that can cause unavailability of the CCP is eliminated.

50. Installation of FE-999
DCP N-049364, Rev. 0 (Unit 1) (LBIE Log No. 97-031)
DCP N-050364, Rev. 0 (Unit 2) (LBIE Log No. 97-032)

This DCP installs a new flow element, FE-999, in the charging injection flowpath downstream of existing FE-917. This was done because of the non-ASME standard installation of FE-917 which is documented in AR A0414083. The effect of this nonstandard installation is such that the bias corrections that would be necessary to correct the reading from FE-917 would restrict the allowable range of settings during the ECCS flow balancing of STP V-15.

Safety Evaluation Summary

While the new flow element will provide a minor restriction to the flow in the charging injection header, it is not the most limiting restriction in the line. Adequate flow is verified by testing each outage by STP V-15.

NDE inspections were performed for the welds on the piping and fittings to assure installation in compliance with applicable construction codes. Testing is performed each outage for injection flow balancing and post loss-of-coolant accident recirculation leakage. All applicable design and licensing standards for the piping and components were complied with to assure all requirements were met in the installation.

Therefore, the installation of this additional flow orifice did not affect the probability or consequences of any accident, new or previously reviewed, nor did it affect the basis for any Technical Specification. There is no affect on the licensing basis of the plant.

51. Gross Failed Fuel Detector Removal (Note: This design change has not been implemented)
DCP N-049369, Rev. 0 (Unit 1) (LBIE Log No. 97-167)
DCP N-050369, Rev. 0 (Unit 2) (LBIE Log No. 97-203)

This design change deletes the gross failed fuel detector (GFFD) system from the NSSS system. The GFFD process skid, and GFFD control console instrumentation will be physically removed. In addition, component cooling water piping, sample tubing, and associated supports will be modified accordingly. This design also includes electrical changes to the GFFD control console and skid power supply and signal wiring. The GFFD control console will remain in-place as it houses main steam line radiation monitors and loose parts monitor pinger circuits.

Safety Evaluation Summary

The GFFD system is not related to any accident previously evaluated in the FSAR Update. The GFFD is a nonsafety-related device, originally designed to monitor reactor coolant during normal operation (for purposes of detecting potential fuel defects). The GFFD provides no accident monitoring function and removal of the GFFD will not affect the capability to obtain a post-accident reactor coolant sample. Process line changes resulting from the GFFD removal are designed to assure leak-tight integrity of the sampling system tubing, and the tubing will continue to be seismically supported to meet system post-HOSGRI cold shutdown requirements. The supply/return piping to the GFFD sample cooler has been redesigned to ensure CCW header "C" efficiencies are not

adversely impacted, and the associated piping will continue to be seismically supported. Design changes to the GFFD control console have been seismically qualified to maintain integrity of the main steam line radiation monitors (which are housed within the GFFD control console) and adjacent inter-linked cabinets.

There is no Technical Specification (TS) requirement to provide for continuous on-line sampling of reactor coolant for purposes of assessing core conditions during normal power operation. Currently, reactor coolant sampling is performed on a batch basis, thus meeting the sampling frequency requirements of TS 3.4.8 (RCS specific activity). Frequent TS-required grab sampling will continue to be performed for failed fuel detection. Note that DCPP continues to search for an instrument to detect for severe failed fuel failures in accordance with Safety Evaluation Report Supplement 6. Therefore, elimination of the GFFD system, considering the existing sampling program, will not result in any margin of safety reduction.

52. <u>Install Zinc Injection Subsystem</u>
DCP N-049408, Rev. 1 (Units 1 & 2) (LBIE Log No. 98-069)
DCP.N-049408 (Units 1 & 2) (LBIE Log No. 98-025)

Install a skid-mounted, zinc acetate injection subsystem designed to inject zinc into the reactor coolant system (RCS) to inhibit stress corrosion cracking in the Alloy 600 steam generator tubes.

<u>Safety Evaluation Summary</u>

This design change installs equipment to inject zinc into the RCS. The new equipment has no impact on any FSAR Update accidents. With regard to boron dilution, the limited capacity of the zinc injection pumps (less than 2 gallons per hour) is insignificant when compared with the 262 gpm dilution flow considered in the uncontrolled boron dilution accident at power. The zinc injection equipment will not interact with or impact the operation of any equipment important to safety. Therefore, it is concluded that no unreviewed safety question is involved.

53. <u>Design Criteria for CVCS Evaporator Feed Demineralizers Resin Loading</u>
DCP N-049429, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-045)

This design change revises the design criteria for the chemical and volume control system (CVCS) evaporator feed demineralizers to allow variation of the combination of anion and cation resin used in the demineralizers to optimize operation.

Safety Evaluation Summary

The proposed changes in resin loading have no effect on accidents analyzed in the FSAR Update and do not impact operation of equipment important to safety. Therefore, it is concluded that no unreviewed safety question is involved.

54. ECCS Pressure Reducing Orifice
DCP N-050286, (Unit 2) (LBIE Log No. 98-006)
DCP N-050286, Rev. 0 (Unit 2) (LBIE Log No. 97-152)

This DCP modified both the charging and safety injection lines of the emergency core cooling system (ECCS). A pressure reducing orifice assembly and trimming orifice were installed in each charging injection line. The charging injection throttle valves 8810A-D, flow orifices (FE 924-927), interconnecting piping, and orifice flanges were replaced. In each safety injection cold leg, a pressure reducing orifice assembly was installed and the flow orifices (FE 974-977) were replaced.

Safety Evaluation Summary

The addition of pressure reducing orifice assemblies and trimming orifices coupled with the replacement of the charging throttle valves, flow elements, and orifice flanges is to prevent pump runout of centrifugal charging pumps (CCPs) and safety injection pumps (SIPs), as well as to avoid potential ECCS flow blockage during the sump recirculation phases. The ECCS delivers flow to the reactor vessel for core cooling and to provide additional shutdown capability following an accident. The ECCS performance is evaluated by using the minimum and maximum pump curves coupled with the maximum and minimum system resistances, which results in the minimum and maximum ECCS injection profiles. System resistance provided by the ECCS throttle valves in each injection line minimizes the spill flow through the broken line and prevents pump runout during a postulated LOCA. The addition of passive pressure reducing orifice assemblies and passive trimming orifices results in distribution of the system resistance previously provided by the single throttle valve. The replacement of charging injection throttle valves and flow elements provides an enhanced design of the existing components.

Since ECCS is not considered an accident initiator and the addition of the passive components does not create new failure modes, this modification does not impact the possibility of an accident nor the consequence of an accident as previously evaluated in the FSAR Update. The addition of the passive components to share the system resistance does not reduce the margin of safety as defined in the basis for any Technical Specification.

55. 21-Month Cycle, Unit 2 Cycle 8
DCP N-050382, Rev. 0 (Unit 2) (LBIE Log No. 97-163)

This design change authorized the extension of Unit 2 Cycle 8 from 18 months nominal to 21 months nominal.

Safety Evaluation Summary

The LBIE for this design change reviewed the following items:

- The use of 1.25 grace period for the not yet surveillance test procedures affected by the approved License Amendment Requests 95-07, 96-10, 97-01, and 97-07
- Changes to instrument setpoint and postaccident monitoring calculations
- Design change notice for changing the pressurizer level high trip setpoint
- Evaluation of steam generator tube integrity
- Impact on major plant systems and components
- Changes to DCMs and FSAR Update
- Effect on Emergency Plan

Based on this review, there were no 10 CFR 50.59 safety issues or unreviewed safety questions identified.

56. Replace Containment Recirculation Sump Screen
DCP N-049317, Rev. 1 (Unit 1) (LBIE Log No. 97-084)

This DCP modifies the Unit 1 outer containment recirculation sump screen (top, sides, and front inclined sections) by replacing the existing mesh with a 1/8-in. mesh opening. This modification is necessary because DCPP has the potential to pass debris through the sump screen that could potentially block flow through the safety injection to cold leg and charging injection to cold leg throttle/runout valves during the recirculation phase of a loss-of-coolant accident (LOCA).

Safety Evaluation Summary

Rescreening with a smaller mesh size will improve the sump's capability to filter out debris, and when combined with another modification to increase the minimum opening in the ECCS injection lines, the possibility of ECCS flow blockage will be minimized. The sump's function of providing a source of long-

term cooling water following a LOCA is not compromised for the following reasons: the sump's structural and seismic integrity is maintained, the required free flow area is maintained under minimum LOCA water level and debris blockage conditions, adequate RHR pump NPSH is maintained, vortex suppression is maintained, no new seismically induced system interaction or high energy line break concerns are created, material selection maintains structural/ functional integrity under all conditions, and the new design does not interfere with the sump level instrumentation.

57. Unit 1 Cycle 9 Reactor Core Reload
DCP N-49368, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-081)

This design change authorized the reloading of the Unit 1 core in a specific pattern of new and partially spent fuel, which is known as the Cycle 9 reload core.

Safety Evaluation Summary

The LBIE for this design change relies, in part, on information provided by the Westinghouse Reload Safety Evaluation that is specific for the core design of Unit 1 Cycle 9. There is no change from previous core designs that triggered the need for prior licensing review. Based on a review of the FSAR Update and associated Chapter 15 accident analysis, there were no 10 CFR 50.59 safety issues or unreviewed safety questions identified.

58. Unit 2 Cycle 9 Reactor Core Fuel Load
DCP N-050368, Rev. 0 (Unit 2) (LBIE Log No. 98-024)

This design change incorporates the new core design from Westinghouse for operation of Unit 2 Cycle 9. This is done routinely for each reload cycle since cores eventually become less reactive and need the addition of new fuel to start a new cycle.

Safety Evaluation Summary

The fabricator provided a 10 CFR 50.59 safety evaluation that verified there are no impacts to the reference safety analyses in the FSAR Update, no unreviewed safety questions, and no impacts on the plant Technical Specifications. This core design meets all the design criteria for maintaining its design basis function. The features implemented in this design are similar to those implemented in previous cores.

59. Unit 2 Overpressurization Protection of Penetrations 49 and 50
DCP P-050371, Rev. 0 (Unit 2) (LBIE Log No. 97-156)

This plant modification added holes to the upstream side of the inner containment isolation ball valves in the liquid radwaste system on containment penetrations 49 and 50. This change protects the integrity of the isolated penetrations against failure in the event a design basis accident were to cause heating, expansion, and pressurization of the fluid trapped between the isolation valves. The change also provides a rupture disc upstream of the valve on penetration 50 to ensure a lower pressure relief path is available during a design basis accident. This change is in response to evaluation of the concerns of Generic Letter (GL) 96-06.

Safety Evaluation Summary

The change is necessary and capable of ensuring the containment isolation system will meet its design and license basis requirements in the event of a design basis accident as detailed by GL 96-06. That is, the change must be implemented to ensure the penetration is not overpressurized due to expansion of the trapped fluid during a design basis accident.

No new containment isolation failure modes were introduced by the change and the designed failure of the rupture disc in the event of a design basis accident will not affect the consequences of the event. Further, the design and procurement quality of the rupture disc ensure that its failure and the subsequent radwaste spill inside containment would not occur for the range of radwaste system operating conditions.

60. Manipulator Crane Parking Position Limitation
DCM S-42B (Units 1 & 2) (LBIE Log No. 97-054)

This evaluation considers the movement of the manipulator crane from its eastmost position over the refueling cavity during operating Modes 1 through 5. Hosgri correspondence to the NRC stated that the crane would be "parked at east end of its travel during this mode (power operation to cold shutdown)." The Hosgri Report states that the crane will not be used in Modes 1 through 5. The capability to move the crane during these modes is necessary to facilitate preventative maintenance and testing of the crane itself or to allow access to structures, systems, or components adjacent to but blocked by the crane (e.g., hatches, lower cavity area, fuel transfer system upender, and cart winches).

Safety Evaluation Summary

The manipulator crane and its support rails remain qualified for Hosgri independent of the parked position of the crane. Civil Engineering has evaluated the crane and containment for seismic and seismically induced system interaction (SISI) effects and have concluded that the configuration is acceptable provided the crane is not parked closer than 5 feet to structures, systems, and

components in its travel path. The SISI spacing criterion described above would assure that the manipulator crane would not strike these components, thereby precluding secondary interactions.

61. Avoidance of Unnecessary Thermal Transients on Alternate Charging Nozzle
DCM S-8, Rev. 6 (Units 1 & 2) (LBIE Log No. 97-128)

This change adds a new entry in the Precautions and Limitations section of the chemical and volume control system design basis document to reflect the Westinghouse recommendation that use of alternate charging be minimized during normal power operation to avoid unnecessary thermal transients on the alternate charging nozzle, and to state that, on this basis, the Inservice Test Program stroke testing of normal charging line Valves 8146 and 8147 should be performed on a cold shutdown frequency rather than quarterly. Excessive use of alternate charging could contribute to eventual fatigue failure of the alternate charging nozzle, resulting in a loss-of-coolant accident.

Safety Evaluation Summary

The normal charging flow path on which these air-operated, fail-open valves are located is isolated by a safety injection signal, so the postaccident position of these valves is inconsequential for purposes of accident mitigation. Hence, the stroke times of these valves has no effect on the consequences of an accident.

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

1. Jumper to Provide Non-1E Power to ABVS Supply Fan, S-32,Rev. 0, Rev. 1, and Rev. 2
(Unit 1) (LBIE Log No. 97-094)

This LBIE evaluates the activity of installing a jumper to allow the operation of the auxiliary building ventilation system (ABVS) Supply Fan S-32 from a nonvital electrical bus during the Unit 1 eighth refueling outage. Fan S-32 provides ventilation cooling air to rooms housing emergency core cooling system (ECCS) components in the auxiliary building. Normally, Fan S-32 is powered by vital Bus H, which would not be available. The redundant supply Fan S-31 was also not available due to the outage of Bus F. This jumper was applicable during Modes 5 and 6. Technical Specification 3/4.7.6 requires the ABVS to be operable in Modes 1 through 4 to ensure that radioactive materials leaking from the ECCS equipment within the auxiliary building following a LOCA are filtered prior to reaching the environment . The ABVS also has the support function to provide ventilation cooling to the areas containing safety-related equipment that is required to be operable to mitigate the consequences of certain design bases accidents and to provide safe shutdown . During defueled condition, Mode 6 or Mode 5, the ABVS has a support function to provide cooling air to the engineered safety features (ESF) equipment rooms served by the ABVS. The installed jumper would allow the ABVS to provide sufficient cooling to the ESF equipment rooms as required during the applicable modes. This jumper was issued as Rev. 0, Rev. 1, and Rev. 2. Only Rev. 2 was installed.

Safety Evaluation Summary

The jumper allowed the ABVS to perform its support function of providing ventilation cooling function to the ESF equipment. The jumper was adequately sized for the expected fan motor loads. In this configuration, Supply Fan S-32 would supply the design bases air flow required to maintain the ESF equipment at their normal operation temperature. The ambient room temperature of the ESF pump rooms (SI, CCW, RHR, charging, and containment spray) in the auxiliary building are monitored. If failure of the non-vital power supply occurred, the pumps would remain operable until the temperature increased to 30°F above the limits stated in Equipment Control Guideline (ECG) 23.1 (133°F for charging and RHR, 141°F for CCW). If the room temperatures reached these limits, corrective actions to provide adequate cooling would be taken to restore room temperature to within limits within the allotted 4-hour limiting condition for operation (LCO). The actions would consist of opening doors or installing a readily available gas-powered temporary power supply and portable fans to direct cooling air to the affected areas. Exceeding the monitored temperature limits does not mean that the equipment will fail, but only that an analysis would

be required to evaluate the impact of the higher temperature had on the operating life of the equipment. Thus, this jumper will not cause the malfunction of equipment important to safety.

2. <u>TSR-98-036 Lead Shielding Request per Procedure RP1.ID2</u>
(Unit 2) (LBIE Log No. 98-015)

This temporayr modification allowed installation and removal of temporary lead shielding on Unit 2 Lines 508, 509, and 927, located above the residual heat removal (RHR) sump in containment. The shielding will be installed in Modes 5 and 6 only and removed prior to entering Mode 4. The shielding will be installed on operable piping which creates a condition that might affect safe operation of the plant not evaluated in the FSAR Update.

<u>Safety Evaluation Summary</u>

The only Mode 5 and 6 accident analyzed in the FSAR Update is a fuel handling accident. The addition of lead shielding onto an operating residual heat removal line does not affect this accident. The blankets' tie-down arrangement is considered structurally adequate such that it will not fail during a seismic event and damage any seismically induced systems interaction targets in the vicinity. Additional weight of the blankets has been evaluated for its impact on the seismic qualifications of piping and found to be acceptable. Based on the above criteria and justification, an unreviewed safety question is not involved.

3. <u>EDG 1-2 Lube Oil Heater-Jumper No. 1-97-012 Alternate Power Supply During 1R8-Bus F Clearance</u>
Jumper 1-97-012, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-048)
Jumper 1-97-013, Rev. 0 (Units 1 & 2) - Bus G (LBIE Log No. 97-047)
Jumper 1-97-015, Rev. 0 (Units 1 & 2) - Bus H (LBIE Log No. 97-049)

This jumper permits emergency diesel generator (EDG) 1-3 lube oil heater to be energized from another power source during the Unit 1 eighth refueling outage for the Bus H clearance.

<u>Safety Evaluation Summary</u>

The Technical Specifications require one operable diesel generator in Modes 5 and 6. With the installation of this jumper, EDG 1-2 will remain operable. Energizing the lube oil heater from another source will have no impact on the accidents evaluated in the FSAR Update. In case of an electrical fault associated with the jumper, the supply breakers will clear the fault. The loss of power to the auxiliary panel has no impact on the EDG to start and load. This jumper is installed to maintain lube oil temperature above 90°F and will prevent unnecessary EDG starts to heat up the lube oil. This will permit EDG 1--2 to

remain operable and one more EDG will be either operable or available. Based on the above, this jumper will not reduce the margin of safety as defined in the basis for the Technical Specifications.

4. <u>Control Room Ventilation System Troubleshooting</u>
Jumper 97-007, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-051)

This LBIE associated with Jumper 97-007 assisted with the troubleshooting efforts associated with AR A0427712 that effected the control room ventilation system (CRVS) and the pressurizer acoustic monitors. In order to successfully troubleshoot and repair the circuit, a jumper was needed to lift certain circuits while maintaining the operability of the CRVS.

<u>Safety Evaluation Summary</u>

The evaluation screen for this condition screened "Yes" as a change to the system operation as described in the FSAR Update. The safety evaluation determined that an unreviewed safety question is not involved based on the Technical Specifications (TS) allowing the plant to operate with only one train of CRVS. Since the jumper would only effect one train, the other train would always be available to satisfy the operability requirements. The plant would enter the TS limiting condition for operation for one train inoperable during the evolution.

5. <u>Add Local Manual Control of CND -2-TCV-23</u>
Jumper 97-018 (Units 1 & 2) (LBIE Log No. 97-120)

A temporary local manual control setup was installed to bypass the normal automatic controls of Valve CND-2-TCV-23. This enabled TCV-23 to remain in service during the replacement of a faulty control element while the plant was on line. TCV-23 cannot be removed from service while the plant is operating.

<u>Safety Evaluation Summary</u>

TCV-23 is Class II (nonsafety-related) and is not included in any Technical Specifications. This jumper disabled TCV-23 to respond to a load transient bypass signal (LTBS) as described in the FSAR Update. Plant generation was reduced to below the 69 percent power level during the installation of the jumper and replacement of the control element. The LTBS cannot be initiated below this power level.

6. <u>STSR-97-137 Lead Shielding Request per Procedure RP1.ID2</u>
STSR 97-137, Rev. 0 (Unit 1) (LBIE Log No. 97-060)

This temporary modification allowed installation and removal of temporary lead shielding in Unit 1 containment for Line 256, Rupture Restraints 4-1RR and 4-2RR. The shielding will be installed in Modes 5 and 6 only and removed prior to entering Mode 4. The shielding will be installed on operable piping which creates a condition that might affect safe operation of the plant not evaluated in the FSAR Update.

Safety Evaluation Summary

The only Mode 5 and 6 accident analyzed in the FSAR Update is a fuel handling accident. The addition of lead shielding onto an operating residual heat removal line does not affect this accident. The blankets' tie-down arrangement is considered structurally adequate such that it will not fail during a seismic event and damage any seismically induced systems interaction targets in the vicinity. Additional weight of the blankets has been evaluated for its impact on the seismic qualifications of piping and found to be acceptable. Based on the above criteria and justification, an unreviewed safety question is not involved.

7. Temporary Modification to the VLPM
Jumper 98-001, Rev. 5A (Unit 1) (LBIE Log No. 98-011)

The vibration and loose parts monitor (VLPM) Channel 6 input lead is lifted by this jumper to prevent nuisance alarms that have been occurring since the Unit 1 eighth refueling outage. The field lead from Channel 6 into the VLPM has a single BNC connector at the end that is disconnected until the source of the noise on the field lead can be determined during the next outage. The implementation of the configuration change reduces the number of inputs as shown in the design documents, and therefore, the design redundancy of the signal from Steam Generator 1-1 loose part monitoring no longer exists. This design redundancy is stipulated in the FSAR Update (Section 4.4.5.4) and, though this limit is descriptive of the system, any change in that description constitutes a change in the FSAR Update text and/or tables. Thus, Question 1a) of the LBIE screen is answered "Yes." Section 4.4.5.4 of the FSAR Update further describes what occurs when the output of an individual transducer channel exceeds an adjustable setpoint. This description includes operator actions and staff actions that qualifies as a procedure as described in the FSAR Update.

Safety Evaluation Summary

The VLPM provides early detection of potential loose parts in the reactor coolant system (RCS) so remedial action may be taken before damage occurs. With one SG 1-1 VLPM channel disabled, one channel remains. The loose parts monitoring computer software has been adjusted to provide an alarm on the one remaining input. The resulting loose parts alert capability is more conservative

in this configuration because both channels must read high to initiate an alarm with two channels available. The probability of an accident or equipment malfunction caused by the lifted input lead will not contribute to an increased accident or equipment malfunction frequency because SG 1-1 loose parts monitoring capability is maintained more conservatively than described in the FSAR Update.

The VLPM is a Class II monitoring system that is not required to mitigate the consequences of any FSAR Update accident or equipment malfunction. This jumper does not raise the consequences of any evaluated accidents or equipment malfunctions in which loose parts are monitored whether the loose parts contribute to the accident or not.

VLPM operability is controlled by ECG 46.1. VLPM function is maintained. No margins of safety for this system are described in the ECG bases or the FSAR Update.

8. Temporary Jumper for Lube Oil Heater to EDG 2-1
Jumper 98-013, (Unit 2) (LBIE Log No. 98-010)

Since Bus F was out of service for maintenance, the power supply to the emergency diesel generator (EDG) lube oil Heater Panel MPF-28 was cleared. In order to maintain lube oil temperature above 90°F, the lube oil heater had to be energized from adjacent Panel MPG-31 using a jumper.

Safety Evaluation Summary

Energizing the lube oil heater from another power source (in this case, its own train) would not have an impact on the accidents evaluated in the FSAR Update. In case of an electric fault associated with this jumper, the supply breaks would clear the fault. Technical Specifications require one operable EDG in Modes 5 and 6. This was satisfied with EDGs 22 or 23. Based on the above criteria and justification, an unreviewed safety question is not involved. Also, a change to the Technical Specifications is not involved.

9. Temporary Modifications/Plant Jumpers
Jumper 98-06, Rev. 1 (Unit 1) (LBIE Log No. 97-042)

The vibration loose parts monitor (VLPM) channel 6 input lead was lifted to prevent excessive nuisance alarms that had been occurring since the Unit 1 seventh refueling outage. Although the input had been taken off scan per Procedure OP1.DC24, the alarms were still coming in to PK 11-11. The lead will remain lifted until the source of the alarms can be determined during the next outage.

Safety Evaluation Summary

This 10 CFR 50.59 safety evaluation was prepared because steam generator (SG) 1-1 VLPM redundancy, as described in the FSAR Update, was reduced from 2 to 1. Also, actions described in the FSAR Update that occur when an individual transducer channel exceeds an adjustable setpoint were modified.

The VLPM provides early detection of potential loose parts in the reactor coolant system (RCS) so remedial action may be taken before damage occurs. With one SG 1-1 VLPM channel disabled, one channel remains. The loose parts monitoring computer software has been adjusted to provide an alarm on the one remaining input. The resulting loose parts alert capability is more conservative in this configuration because both channels must read high to initiate an alarm with two channels available. The probability of an accident or equipment malfunction caused by the lifted input lead will not contribute to an increased accident or equipment malfunction frequency because SG 1-1 loose parts monitoring capability is maintained more conservatively than described in the FSAR Update.

The VLPM is a Class II monitoring system that is not required to mitigate the consequences of any FSAR Update accident or equipment malfunction. This jumper does not raise the consequences of any evaluated accidents or equipment malfunctions in which loose parts are monitored, whether the loose parts contribute to the accident or not.

VLPM operability is controlled by Equipment Control Guideline (ECG) 46.1. VLPM function is maintained. No margins of safety for this system are described in the ECG bases or the FSAR Update.

10. Determinate Defective Pressurizer Heaters
MMP M000059, Rev. 1 (Unit 2) (LBIE Log No. 97-028)

Maintenance Modification Package (MMP) M000059 allows defective pressurizer heaters to be disconnected. This allows the remaining heaters fed from the same circuit breaker to be returned to service. The initial pressurizer heater capacity was 1800 kW. This MMP allows failed heaters to be disconnected as long as total capacity of at least 1340 kW with Heater Groups 1 and 4 each having at least 276 kW and Groups 3 and 4 each having at least 345 kW is maintained.

Safety Evaluation Summary

The pressurizer heaters are nonsafety related. The purpose of the pressurizer heaters is to control pressurizer pressure during heatup and power operation and to support natural circulation of the reactor coolant system during the loss of

offsite power. 150 kW is required to support natural circulation. Technical Specifications define the minimum pressurizer heater capacity as 150 kW from two groups which can be supplied by vital emergency power. This MMP maintains this 150 kW by limiting the number of heaters that can be disconnected. Therefore, all margins of safety implicit in this Technical Specification requirement are maintained by this MMP.

11. Main Feedwater Overspeed Trip Test
Jumper PEP-04R, Rev. 5A (Units 1 & 2) (LBIE Log No. 98-036)

The overspeed trip test is normally performed using main steam. This revision of the procedure makes it acceptable to use a cross-tie to the auxiliary steam system to warm up and overspeed the turbine. The overspeed trip test is performed with the main feedwater pump uncoupled regardless of the source of the turbine steam supply.

Safety Evaluation Summary

The overspeed trip test is performed with the main feedwater pump uncoupled and out of service. The use of an alternate supply of motive steam does not create a new accident or potential malfunction. The potential failure of the jumper is bounded by a break in the auxiliary steam header.

12. Operation of the Component Cooling Water (CCW) System to Support Replacement of Temperature Control Valve (TCV)-130
TP T0-9705, Rev. 0 (Unit 1) (LBIE Log No. 97-088)

This temporary procedure was prepared to support replacement of CCW temperature control valve TCV-130. Due to a leaking return isolation valve, leak tight isolation of the line was not possible. To replace TCV-130, restriction orifice RO-239 had to be replaced with a blank plate so the bypass line would pass flow. It was necessary to establish a bleed path downstream of RO-239 to relieve pressure on the orifice so that its flange could be disassembled and the blank plate installed. New bypass isolation valves were added using a hot tap procedure that was also used to establish the bleed path.

This procedure provided instructions to establish the bleed path from the CCW system and regulate makeup flow during the activities associated with RO-209.

Safety Evaluation Summary

The vital portions of the CCW system are designed to mitigate the consequences of an accident by removing heat from the primary system and transferring it to the ocean. The probability of FSAR Update Chapter 15 accidents is not affected by CCW system operation. The consequences of

FSAR Update Chapter 15 accidents are not increased provided CCW system operation is maintained. The temporary procedure provided for manual makeup to maintain CCW surge tank level in the normal range. The bleed rate was within the makeup capability of the makeup water system. The RCS was depressurized or at very low pressure during the maintenance operation to minimize leakage of radioactive contamination. Operation of the CCW system was not significantly affected by this procedure; consequences of analyzed accidents and equipment malfunctions were not increased. No new accidents or equipment malfunctions were created. Adequate inventory was maintained in the CCW surge tank. Technical Specification 3.7.3.1 and its bases were not challenged by this procedure.

13. Installing Turbine Building Siding Near High-Voltage Lines and Equipment
TP TA-9701, Rev. 0 (Unit 1) (LBIE Log No. 97-034)

This temporary procedure was written to guide and control re-siding installation activities during the Unit 1 eighth refueling outage. Re-siding the northeast corner of the turbine building involves work near energized high-voltage sources. The Unit 2- 230-kV and 500-kV lines were energized during work near the 500-kV lines. The work was performed with a combination of suspended scaffolding and manlifts. The 230-kV and 500-kV lines are close together; additional controls were needed to ensure personnel and equipment safety.

Safety Evaluation Summary

This evaluation was performed to evaluate the implementing methods and equipment used for the re-siding work and to verify that failure of the methods and equipment would not affect safety-related equipment and safe plant operation.

Suspended scaffolding was evaluated for lifts over restricted areas, seismic interaction issues, personnel safety and operational loads to ensure structural integrity during installation and operation. Deployment and operation of the 175-ton boom crane and mobile manlifts were in compliance with all applicable procedures. Crane ground path, swing path, and station points were evaluated and documented. Cranes and manlifts were evaluated for tipping issues. Electrical observers were stationed during the work. Crane operators were trained and qualified per ANSI/SIA 92.5. Accident possibility and probability were not increased.

Installation and operation of the equipment as described above will not affect safety-related equipment. Heavy loads were not lifted over exclusion areas. Equipment malfunction possibility, probability, and consequences were not increased.

14. <u>Providing Vital 125 Vdc Power from SD 12 to SD 13 Vital Loads</u>
TP TA-9702, Rev. 0 (Unit 1) (LBIE Log No. 97-035)

This temporary procedure provided instruction to install a Class 1E jumper from dc Bus 12 to power 4-kV Bus H and its associated safeguards relay board in Modes 5 or 6. Normally, dc Bus 13 powers up these loads. However, dc Bus 13 was unavailable because its battery was being replaced. The reconfiguration of vital dc control power enabled dc power from dc Bus 12 to power diesel generator (DG) 12 and associated 4-kV Bus G, and DG 11 and associated 4-kV Bus H.

<u>Safety Evaluation Summary</u>

This temporary configuration was implemented for Modes 5, 6, or while defueled in the Unit 1 eighth refueling outage. Class 1E jumpers from dc Bus 12 to the 4-kV Bus H and it associated safeguards relay board were provided. DG 11 did not require jumpers as it is provided with a dc power transfer switch, which was selected to dc Bus 12. All applicable Technical Specifications and Outage Safety Plan requirements were met. The basis for allowing this temporary configuration was based on the following:

- Battery 12 was shown to have capacity to simultaneously operate both 4-kV buses (G and H) and start both DGs (12 and 11).

- The jumper met class IE requirements and did not introduce new failure modes.

- While operating in Modes 5, 6 or defueled, it is not necessary to postulate a single failure of the cross-train Class IE equipment.

15. <u>Replacement of Auxiliary Transformer 2-1</u>
TP TB-9721, Rev. 1 (Unit 2) (LBIE Log No. 98-020)

This temporary procedure addresses the replacement of auxiliary transformer 2-1 (UAT21). The replacement of UAT21 requires use of a crane and trailer. The process could make startup transformer (SUT) 22 and its deluge system a seismically induced systems interaction (SISI) target. During the Unit 2 eighth refueling outage, the SUT 22 provides offsite power to the vital 4-kV and 480-V ac loads.

Safety Evaluation Summary

In Modes 5 and 6, Technical Specifications require one offsite power source and one emergency diesel generator (EDG) be operable. In the event of failure of SUT 22, the EDG will provide power to the vital loads. During load lifting, the crane and its boom will not be oriented in line with SISI targets and located at a distance that minimizes potential for damage to SUT 22.

There is no inservice equipment important to safety in the area where this activity is performed. Vital 4-kV switchgear is inside the turbine building and damage to the nonvital SUT 22 will not affect the vital Bus E or DG safety function. The safety margin is not affected by this activity since the plant will be in Mode 5 or 6 and the outage safety plan addresses the requirements for power availability.

16. Moving Unit 2 Auxiliary 21 Transformer
TP TB-9721, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-017)

This temporary procedure was used to cover the drayage procedure used by Bragg Crane & Rigging Co. to move the old Auxiliary 21 Transformer out of the protected area and the new UST Auxiliary 21 Transformer from Lot 1 into the protected area.

Safety Evaluation Summary

Replacement of the Auxiliary 21 Transformer required the use of a skid system, a 200-ton crane and a transportation trailer. The work was done during Modes 5 and 6 in the Unit 2 eighth refueling outage and in the vicinity of Startup Transformer 22. The transformers were moved fully dressed and filled with oil. This temporary procedure addressed precautions taken to keep the Startup Transformer and its deluge system operational and also addressed the environmental concerns associated with an oil spill or a fire, and the impact on the Emergency Plan.

17. Implementation of DCP E-49297 Battery 13 Replacement
TP TD-9703, Rev. 0 (Unit 1) (LBIE Log No. 97-039)

This temporary procedure provided instruction to install a nonClass 1E jumper from nonvital Battery 17 to provide power to selected dc Bus 13 loads during the Unit 1 eighth refueling outage. Normally vital 125-Vdc Distribution Panel 13 loads are powered from vital Battery 13. However, due to Battery 13 replacement during the Unit 1 eighth refueling outage, selected Class 1E loads were powered via a nonvital jumper from nonvital Battery 17. The jumpers were

necessary to keep SD 13 loads operational and support outage related activities.

Safety Evaluation Summary

This temporary configuration was implemented during Modes 5, 6, or while defueled in the Unit 1 eighth refueling outage. NonClass 1E jumpers from Battery 17 to selected dc Bus 13 loads were provided. The basis for allowing this temporary configuration was based on the following:

- Nonvital Battery 17 was determined to have adequate capacity to power up the selected Class 1E jumpered loads. The circuit breakers and jumpers used in the jumper scheme were evaluated and sized and coordinated for the selected loads.

- The nonvital jumper scheme did not introduce new failure modes or create a different type of accident.

- Only those selected loads whose design classification was non-Q were able to be declared operable. The rest of the loads, even though energized by the temporary jumpers, were declared inoperable and no credit was taken to meet Technical Specification limiting condition for operation requirements.

18. Providing Vital 125 Vdc Power from SD 22 to SD 23 Vital Loads
TP TD-9802, Rev. 0 (Unit 2) (LBIE Log No. 97-184)
TP TD-9803, Rev. 0 (Unit 2) (LBIE Log No. 97-190)

This temporary procedure provided instructions for jumpering vital SD 23 loads to SD 22 for the Battery 23 replacement during the Unit 2 eighth refueling outage.

Safety Evaluation Summary

During the vital Battery 23 replacement in the Unit 2 eighth refueling outage, selected Class 1E loads that were vital to Mode 5 or 6 safety were fed from vital Battery 22. This configuration was reviewed with the Outage Safety Plan and found to not increase the probabilities or consequences of any Mode 5 or 6 accidents previously evaluated in the FSAR Update. No unreviewed safety questions were identified.

19. Energize Unit 1 12-kV Startup Bus from Auxiliary Transformer 11
TP TO-9701, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-055)

This temporary procedure (TP) energizes the Unit 1 12-kV startup bus from Auxiliary Transformer 1-1 and clears Startup Transformer 1-1 for replacement

and Startup Transformer 1-2 for maintenance. Loss of offsite power to the operable 4-kV vital buses is the only impact possible from this alignment.

Safety Evaluation Summary

Temporary Procedure (TP) TO-9701 is to be performed in Modes 5 and 6. In these modes, analyzed accidents that may be affected are fuel handling accidents, tank ruptures, and the boron dilution event. Loss of offsite power does not cause or affect mitigation of fuel handling accidents since containment isolation does not require offsite power, nor does the fuel handling building (FHB) ventilation system. Loss of offsite power has no effect on tank ruptures. Loss of offsite power does not affect the ability to secure the primary water makeup pumps - the limiting boron dilution event. A loss of offsite power is mitigated by the emergency diesels starting and assuming the vital bus loads. This temporary procedure has no effect on the ability of the Technical Specification and Outage Safety Plan required emergency power sources. Thus, there is no potential unreviewed safety question.

C. Procedure Changes

1. Core Operating Limits Report (COLR) for DCPP Unit 2, Cycle 9
COLR 2-9, Rev. 0 (Unit 2) (LBIE Log No. 98-026)

This report was performed for the initial issue of COLR 2-9. The COLR for Unit 2 Cycle 9 is the same as for Unit 2 Cycle 8 with the exception of W(z) factors that are cycle specific. Because the safety evaluation was performed by a vendor that is not Plant Staff Review Committee (PSRC)-approved, the answer to Question no.4 for the 50.59 screen on the LBIE Screen was "yes."

Safety Evaluation Summary

The safety evaluation performed for this report is the same as the one performed for the Design Change Package (DCP) N-050368, Rev. 0. That evaluation found that there are no adverse consequences to components or systems due to this core reload design. No new performance changes or demands on other components or systems are introduced by this core design.

2. Using Five-Year Average X/Qs in OffSite Dose Calculations
CAP A08, Rev. 20 (Units 1 & 2) (LBIE Log No. 96-044)

FSAR Update Section 11.3.7 assumed historical annual average X/Q values for calculating dose from normal operations for the licensing basis. Procedure CAP A-8 uses historical five-year average X/Q values for calculating dose under the same conditions.

Safety Evaluation Summary

10 CFR 50, Appendix I, states design objectives and limiting conditions for operation of for nuclear power reactor effluents. Limits to meet these conditions are implemented by Technical Specification 6.8.4.6.

FSAR Update Section 11.3 states the results of a pre-operational analysis for the estimated gaseous effluents and dose during normal operation. The analysis was performed to demonstrate that the criteria of 10 CFR 50, Appendix I, can be met. This analysis assumed annual average X/Q conditions. PG&E believes the calculated dose is the licensing basis.

Procedure CAP A-8, "Offsite Dose Calculations," implements the methodology used during normal plant operations to ensure compliance with 10 CFR 50, Appendix I, and the Technical Specification requirements. CAP A-8 uses five-year historical average X/Q values to calculate radioactive gaseous effluent dose and these values are used as one of the variables to calculate radioactive effluent and radioactive process monitor high alarm setpoints (HASP). The

HASP values are used to ensure the dose rate limits are not exceeded. The dose rate limits are a fixed value. Therefore, as X/Q values increase, the release rate decreases to maintain the dose rate limit. As X/Q decreases, the release rate limit may increase up to the dose rate limit. Thus, HASP's determined by CAP A-8 account for X/Q variations and ensure dose rate limits are not exceeded.

Five-year historical average X/Q values are more representative of average DCPP meteorology conditions than historical average X/Q values (used in the FSAR Update Section 11.3 analysis). The five-year historical average X/Q values may, in any given year, be more or less than the corresponding annual average values as stated as "estimates" in the FSAR Update analysis.

In the current revision 20 of Procedure CAP A-8, the historical five-year X/Q values are less than those in the FSAR Update Section 11.3 analysis. The values used in the analysis are listed in Table 11.3-11, " Estimates of Relative Concentration X/Q at Locations Specified in Table 11.3.-10."

An FSAR Update change to include the use of five-year historical meterological data to calculate X/Q values has been submitted.

3. <u>Offsite Dose Calculations</u>
CAP A-8, (Units 1 & 2) (LBIE Log No. 97-083)

The Offsite Dose Calculation Process (ODCP) X/Q and D/Q values are updated yearly based upon the latest five-year meteorological data. The FSAR Update also lists X/Q and D/Q values. The issue is how does the ODCP X/Q and D/Q revisions impact the FSAR Update values.

<u>Safety Evaluation Summary</u>

10 CFR 50.34a requires nuclear power plants to be designed in such a way that doses due to routine effluent releases not exceed the 10 CFR 50, Appendix I, dose design objectives. The X/Q and D/Q values listed in the FSAR Update, Section 11.3, are used for pre-operational demonstration of compliance with the 10 CFR 50.34a design criteria and, therefore, represent design bases for licensing. For purposes of demonstrating the design criteria, dose pathways and locations are assessed that are not utilized for routine effluent control. The actual dose pathways and locations used for routine effluent controls are based on the annual land use census information, as well as concurrent (latest five-year annual average) meteorological data.

Thus, the FSAR Update X/Q and D/Q values, which are used for 10 CFR 50.34a calculations, are unrelated to the X/Q and D/Q values used for routine effluent dose assessment.

4. Pilot Process Instruction Development
AD1.ID8, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-033)

This is a new procedure that provides requirements and supplemental guidance for developing instructions to control pilot processes. This procedure is in support of the Work Control Process reengineering effort.

Safety Evaluation Summary

This procedure controls how administrative instructions are written and does not directly control activities in the plant, nor does it directly control operation of equipment important to safety. This procedure does not interact with any specified margin of safety as defined in any Technical Specification.

5. Control of the Surveillance Testing Program
AD13.DC1, Rev. 4 (Units 1 & 2) (LBIE Log No. 97-207)

This revision of Procedure AD13.DC1 adds the manual vents and drains between the inner and outer containment isolation valves. The NRC recently rovided additional clarification as to the applicability of Technical Specification (TS) 4.6.1.1a in relation to which penetrations were considered "in service" during accident conditions. License Amendment 73 and 72 relocated TS Table 3.6-1, "Containment Isolation Valves," to the Diablo Canyon Power Plant Procedures that are subject to the change control provisions in the administrative controls section of the TS. Any change to the containment isolation valve list would constitute a change to the facility and thus would be subject to the provisions of 10 CFR 50.59.

Safety Evaluation Summary

The change is administrative in nature. The change should improve administrative practices without any effect on plant operations. Improved administrative practices increase the likelihood the valves will be maintained closed, thereby improving mitigation potential.

6. Control of the Surveillance Testing Program
AD13.DC1, Rev. 5 (Units 1 & 2) (LBIE Log No. 98-032)

License Amendment Request (LAR) 91-08 proposed relocation of Technical Specification (TS) Table 3.6-1, "Containment Isolation Valves," to the DCPP

procedures that are subject to the change control provisions in the administrative controls section of the TS. This revision of AD13.DC1 adds VAC-2-540 to Attachment 7.7, "Containment Isolation Valves." This change does not affect the FSAR Update.

Safety Evaluation Summary

This change is administrative in nature and should result in improved administrative practices without any effect on plant operations. The change does not result in any physical modifications and does not alter the method by which any safety-related system performs its function.

Adding VAC-2-540 to the list increases assurance containment integrity is maintained. Improved administrative practices increase the likelihood the valves will be maintained closed, thereby improving mitigation potential.

7. Outage Safety Management of Increased Risk Periods Including Hot Mid-Loop Operations
AD8.DC52, Rev. 4 (Units 1 & 2) (LBIE Log No. 98-018)

This evaluation addressed programmatic issues to incorporate hot mid-loop operations into outage nuclear safety management strategies. Specifically, it addressed a change to a policy statement regarding avoidance of mid-loop operations with fuel in the reactor vessel, and a change in the configuration of the reactor coolant system (RCS) prior to entering reduced inventory operations.

Safety Evaluation Summary

There are no FSAR Update accidents postulated for shutdown events other than a misplaced fuel assembly or fuel handling accident for which this change has no effect.

Changing the authorization process for mid-loop operation, and keeping the reactor head tensioned prior to reduced inventory conditions do not adversely affect RCS or support system hydraulics, heat transfer, pump operation, safety analyses or Technical Specification bases.

The final approval for mid-loop operation still remains with the Vice President and Plant Manager, DCPP. The Vice President and Plant Manager, DCPP is responsible for, and has control over, unit safe operation per the Technical Specifications, FSAR Update Chapter 13, and implementing Program Directives.

Allowing the reactor head to remain tensioned prior to reduced inventory operation does not affect any RCS system in use during shutdown conditions. None of the functions of the residual heat removal system or other systems

required for shutdown operations are affected by having a tensioned reactor vessel head prior to reduced inventory conditions.

8. <u>Core Offload Sequence</u>
OP B-8DS1, (Unit 2) (LBIE Log No. 98-019)

Action Request (AR) A0454011 describes an event that occurred during core offload for the Unit 2 eighth refueling outage, stemming from an indeterminate crane failure. Due to the inability to specifically identify the failed component or condition, it was decided to develop an action plan and cautiously proceed.

<u>Safety Evaluation Summary</u>

The crane functions of overload/underload/slack cable still function and are not questioned. The hoist features are separate from the crane lateral movement and overload setpoint features. The gripper and motor failure features are fail-safe and function normally. The event described has no impact on the safety features inherent in the crane design. The safety features, as described in the FSAR Update and Technical Specifications, remain operable, and the inherent safety provided by them is maintained. The Technical Specification requirements remain satisfied in this event and action plan. There is no reduction in margin of safety.

9. <u>OP C-7C:VI, "Transferring/Offloading Sulfuric Acid and Ammonium Hydroxide"</u>
OP C-7C:VI, Rev. 10 (Unit 1) (LBIE Log No. 97-140)
OP C-7C:VI, Rev. 9 (Unit 2) (LBIE Log No. 97-141)

This procedure change added measures to mitigate a chemical spill when offloading chemicals.

<u>Safety Evaluation Summary</u>

The proposed change blocks potential drainage paths when offloading sulfuric acid or ammonium hydroxide. This warranted an environmental protection plan review under the Licensing Basis Impact Evaluation (LBIE) screening criteria. Since the proposed changes would not add any new discharges, would not require a change to the Environmental Protection Plan, would not change quantities of chemicals used or stored at DCPP, nor add any new hazardous waste streams, no unreviewed environmental question exists.

10. <u>EQ Program Implementation in NTS</u>
DLAP CF3.NE8, Rev. 0A (Units 1 & 2) (LBIE Log No. 97-181)

Section 3.11.1.4 of the FSAR Update, "Class 1E Electrical Equipment Qualification List Maintenance," specified that a hard-copy output (a RAMIS

Report) of the DCPP EQ Master List information from PIMS constitutes the DCPP Class 1E Electrical Equipment Qualification List and was maintained as a controlled drawing (PG&E Drawing 050909). This Licensing Basis Impact Evaluation (LBIE) was performed to remove the administrative requirement to issued a hard copy of the EQ Master List from the FSAR Update.

The EQ Master List is now a living list that is comprised of certain fields in the PIMS component database. Therefore, there is no value in unnecessarily issuing Drawing 050909.

Safety Evaluation Summary

This administrative change does not affect the operation of the plant or accident initiation, consequences, or probability. It is an FSAR Update revision to change how the EQ Master List is handled. It was previously issued as a hard-copy drawing on a 6-month frequency. Now it is a living document (in the PIMS component database) that is revised on an ongoing basis. The end result is the Master List is always kept current versus being up to 6 months out of date. All changes to the EQ Master List are reviewed on a 6-month frequency, thereby ensuring the accuracy of the living EQ Master List.

11. Design Change Requests and Design Change Vehicles
CF4.ID1, Rev. 3 (Units 1 & 2) (LBIE Log No. 97-208)

This procedure describes the process for initiating design change requests and selecting an appropriate design change vehicle. The procedure revision introduces a new process for Class N Modifications.

Safety Evaluation Summary

This procedure revision involves Class N Modifications that are defined as being minor in nature, do not change a setpoint, do not affect the DCPP design basis, and are not considered to be design changes. Therefore, there can be no reduction in the margin of safety as defined in the basis for any Technical Specification.

12. Reactor Trip or Safety Injection
EOP E-0, Rev. 19 (Unit 1) (LBIE Log No. 97-025)

Emergency Operating Procedure (EOP) E-0 was modified to add an instruction to place two component cooling water (CCW) heat exchangers in service if the reactor coolant system (RCS) is not intact at step 21 as preparation to transfer to EOP E-1.3. An item was also added to the foldout page to transition to EOP E-1.3 if the refueling water storage tank (RWST) level is less than 33 percent.

The changes were made to reduce the operator response time to EOP E-1.3 when aligning the RCS for cold leg recirculation. The specific location of the instruction was chosen to ensure that it would be performed (i.e., not bypassed due to RNO on other steps), and to minimize potential disruptions in the flow of diagnosing and responding to the accident in progress.

Safety Evaluation Summary

The change affects equipment alignment following a loss-of-coolant accident (LOCA), which is not the cause or initiating event of an accident. There is no change to method of operation for any accident mitigation equipment. The possibility and probability of accidents or equipment malfunctions are not affected. Placing two CCW heat exchangers in service at the subject step improves operator response time to a LOCA and does not affect operator response times to non-LOCA accidents diagnosed by EOP E-0. Accident analysis assumptions are not affected if both CCW heat exchangers are placed in service and transfer to cold leg recirculation is not needed. All safety-related equipment verifications will have been performed before transfer to EOP E-1.3 when the RWST level falls to 33 percent. The consequences of an accident are not affected. The transfer does not affect Technical Specification provisions for ECCS operability and long-term core cooling. Safety margins are not affected.

13. Reactor Trip or Safety Injection
EOP E-0, Rev. 20 (Unit 1) (LBIE Log No. 97-011)
EOP E-0, Rev. 11 (Unit 2) (LBIE Log No. 97-012)

Emergency Operating Procedure (EOP) E-0 was revised such that ECA-0.0 will not be performed if any vital bus is energized. The intent is to cope with loss of ac emergency power until at least one emergency bus can be energized.

Prior to this change, one complete train of emergency core cooling system (ECCS) equipment was required to exit emergency contingency guidelines and return to the recovery and functional restoration (E-series and FR-series) guidelines. The operator was directed to remain in ECA-0.0 with one vital 4-kV bus energized because two buses were required to guarantee restoration of minimum safeguards equipment. However, with power restored to one vital bus, some degree of core cooling becomes available. Westinghouse Direct Work No. 92-033 clearly stated that minimum safeguards capacity is not required to be in the recovery and function restoration guidelines.

This change will allow the operators to fully use the emergency procedure network to mitigate concurrent accidents.

Safety Evaluation Summary

The safety evaluation was performed to evaluate the effect on ECCS equipment and to verify that a failure mode could not be created by implementing the E-series or FR-series emergency procedures with less than one full train of safeguards equipment.

The emergency contingency, recovery, and functional restoration procedures are used to mitigate the consequences of accidents. Their use cannot cause an accident.

All FSAR Update accidents were evaluated and validated for recovery with only one 4-kV bus available. In each case, it was determined that recovery was possible. The revised procedure allowed recovery options that were previously not available. Recovery was not impaired and consequences were not increased.

The change provides improved guidance for the operators when a vital bus is energized. Other procedures are adequate for accident mitigation. Use of the modified procedures does not increase the possibility, probability, or consequences of any equipment malfunction or accident. The margin of safety is not reduced because remaining equipment can be operated optimally to maintain core cooling.

14. Reactor Trip or Safety Injection
EOP E-0, Rev. 10 (Unit 2) (LBIE Log No. 97-026)

Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection" was revised to include an action to place two component cooling water (CCW) heat exchangers in service if it is determined that the reactor cooling system (RCS) is not intact and a transition to EOP E-1, "Loss of Reactor or Secondary Coolant," is required. Placing both CCW heat exchangers in service is done in anticipation of an eventual transition to EOP E-1.3. Placing this action in EOP E-0 is advantageous as it removes the operation from the timeline of cold leg recirculation alignment. Also, the foldout page was revised to instruct the operators to go to EOP E-1.3 immediately if the refueling water storage tank (RWST) level is less than 33 percent.

Safety Evaluation Summary

EOP E-0 provides diagnostic steps to provide the operators with the symptoms and appropriate actions for main steam line break (MSLB), steam generator tube rupture (SGTR), and loss-of-coolant accident (LOCA). It also provides a direct path to terminate a safety injection (SI). The accident analyses for MSLB and

SGTR, and the analysis for inadvertent SI assume specific operator response times.

The addition of a step to place both CCW heat exchangers in service following a LOCA does not impact these analyses. Accident mitigation for the LOCA is not adversely affected as the timeline for realignment for cold leg recirculation is improved. If a LOCA is diagnosed and transfer to cold leg recirculation is not needed, alignment of the second heat exchanger does not significantly delay the actions of EOP E-1; therefore, accident analysis assumptions are not altered.

15. Loss of Reactor or Secondary Coolant
EOP E-1, Rev. 14 (Unit 1) (LBIE Log No. 97-022)
EOP E-1, Rev. 8 (Unit 2) (LBIE Log No. 97-023)

This emergency operating procedure revision deletes the step that verifies that the water level in the containment recirculation sump is sufficient to support the operation of the residual heat removal (RHR) pumps in cold leg recirculation. The adequacy of the recirculation sump level to support RHR pump operation is now verified in EOP E-1.3 prior to placing the RHR pumps into service. Delaying the verification until this time allows for more inventory to collect in the sump. This reduces the potential for unnecessarily entering ECA-1.1.

Safety Evaluation Summary

The purpose of checking the level in the containment recirculation sump is to confirm that there is sufficient water available to support the operation of the RHR pumps during cold leg recirculation. Deleting this step from E-1 does not create an unreviewed safety question as this step is now performed in EOP E-1.3 just prior to placing the RHR pumps in service. Delaying this verification step to EOP E-1.3 decreases the potential of inadvertently entering ECA-1.1 due to insufficient sump level. Entering ECA-1.1 unnecessarily would delay the operator's overall response to the event. EOP E-1.3 contains guidance to enter this procedure if sump level is not adequate when the step is reached to place the RHR pumps in service. Additionally, operators are instructed to monitor for RHR pump cavitation.

16. EOP E-1.3, "Transfer to Cold Leg Recirculation"
EOP E-1.3, Rev. 6 (Unit 1) (LBIE Log No. 96-025)
EOP E-1.3, Rev. 5 (Unit 2) (LBIE Log No. 96-026)
EOP E-1.3, Rev. 6 (Unit 2) (LBIE Log No. 97-002)
EOP E-1.3, Rev. 7 (Unit 2) (LBIE Log No. 97-021)

This procedure describes the process of reconfiguring emergency core cooling pumps and valves from the injection phase of a loss-of-coolant accident (LOCA) to the recirculation phase of such an event. This process is described in detail

in the FSAR Update, so any changes to that sequence require an evaluation under 10 CFR 50.59. The proposed revisions added a few check and action steps to the process. It should also be noted that this evaluation covered all previous revisions to the procedure, which modified this sequence in various ways, but were not evaluated under 50.59.

Safety Evaluation Summary

The FSAR Update includes a statement that the above described process is completed in "approximately 10 minutes." As demonstrated analytically and through simulator runs, this sequence could be accomplished in such a time, and the addition/deletion of some of the FSAR Update-described steps does not conflict with that statement. Furthermore, the steps being added/deleted were necessary to assure proper completion of the switchover, and do not result in depletion of the refueling water storage tank (RWST) (which would require termination of injection during a LOCA) before recirculation has begun. Therefore, the modification of the FSAR Update-described sequence does not challenge the operator's ability to successfully transfer to cold-leg recirculation within the required timeframe in order to mitigate a LOCA. An unreviewed safety question is not involved.

17. Transfer to Cold Leg Recirculation
EOP E-1.3, Rev. 15 (Unit 1) (LBIE Log No. 97-020)
EOP E-1.3, Rev. 7 (Unit 2) (LBIE Log No. 97-021)

This emergency operating procedure revision moves the requirement to check the containment recirculation sump level check to just prior to starting the residual heat removal (RHR) pumps, and to move the requirement to locally close the breakers for the refueling water storage tank (RWST) suction isolation valves to the start of the procedure. Other changes increase the usability and efficiency of the procedure. As a result, the time to switchover is reduced by 45 seconds.

Safety Evaluation Summary

Restructuring the emergency operating procedure (EOP) steps reduces the time required to reach cold leg recirculation. Delaying the verification of the recirculation sump level reduces the possibility of inadvertently entering ECA-1.1 in response to inadequate sump level. Guidance has been added to ensure that RHR pump cavitation due to inadequate suction is identified, and the procedure continues to direct the operators to take the appropriate action in response to this condition. Improving the timeliness of the switchover to cold leg recirculation ensures that the design basis for the emergency core cooling system is maintained during realignment.

18. Transfer to Cold Leg Recirculation
EOP E-1.3, Rev. 14 (Unit 1) (LBIE Log No. 97-001)
EOP E-1.3, Rev. 6 (Unit 2) (LBIE Log No. 97-002)

This revision added steps to accentuate the need for timeliness and to allow certain steps to be performed in parallel. The evaluation also covered all past revisions to Emergency Operating Procedure (EOP) E-1.3 since safety evaluations were not performed for some revisions or were considered to be inadequate for others.

Safety Evaluation Summary

FSAR Update Tables 6.3-4 and 6.3-5 and Sections 6.3.1.4.4.2, 6.3.1.4.4.3, and 6.3.2.17 describe the process of transfer from the emergency core cooling system (ECCS) injection mode to the cold leg recirculation mode of operation after a loss-of-coolant accident (LOCA). Table 6.3-5 contains the basic sequence of operations to establish cold leg recirculation. Certain steps of EOP E-1.3 have been modified such that the sequence of operations is somewhat different than that described in the FSAR Update.

The following steps have been added to EOP E-1.3 that do not appear in FSAR Update Table 6.3-5:

- Step 3.d, which verifies the ASW/CCW is aligned for two ASW pumps through two heat exchangers
- Steps 5.d.3) and 7.f.3), which verify decreasing the RHR heat exchanger outlet temperatures after their respective RHR pumps have been started
- Step 6.a, which closes the CCP recirculation valves 8105 and 8106
- Steps 6.d. and 6.g.2), which throttles the RHR heat exchanger outlet valves when the RHR pumps begin supplying suction flow to the SI and CCPs

Additionally FSAR Update Table 6.3-5 identifies Valves 8701 and 8702 as being checked closed in order to provide RCS to RHR suction isolation. This is not explicitly done in EOP E-1.3, since these valves are maintained closed in Modes 1-3.

FSAR Update Section 6.3.1.4.4.2 states that the total time for the changeover from injection to recirculation is approximately 10 minutes, as shown in Table 6.3-5. The purpose of FSAR Update Table 6.3-5 is to provide a guideline for the emergency operating procedures to accomplish the transfer to cold leg recirculation in approximately 10 minutes. The importance of the 10 minutes is to ensure that there is adequate water inventory in the refueling water storage

tank (RWST) for the continuous cooling to the core/containment by the centrifugal charging pumps (CCPs), safety injection pumps (SIPs), and containment spray pumps (CSPs). The transfer to cold leg recirculation must be completed prior to running out of inventory in the RWST.

Emergency Procedure (EP) EOP-1.3, Rev. 0, contained a step that checked the position of several MOVs and a step to locally close breakers for Valves MOV 8980 and MOV 8976. It is assumed that an operator would be dispatched to close these breakers at that step, and at the point closing the valves is requested, it would have been accomplished. There is adequate time in the procedure as detailed in FSAR Update Table 6.3-4A for this to occur. It should be noted that in the current revision of EOP E-1.3, this action is moved to step 9.b.2, which is after the transfer to cold leg recirculation is completed. Neither of these two items would have an appreciable impact on the ability to complete the transfer in approximately 10 minutes. This review is considered to bound all the past revisions of EOP E-1.3 up to and including Revision 14.

The stated transfer time to cold leg recirculation of 10 minutes is an informational guideline as stated in the FSAR Update by the use of the term "approximately 10 minutes." The 10 minutes is considered a guideline since considerable margin exists in the assumptions in the table for ECCS pump flows and useable volume in the RWST between the low-alarm RHR pump trip level and the low-low alarm level (4 percent). Simulator validation with randomly selected operating crews demonstrated that the transfer to cold leg recirculation could be accomplished in the required timeframe. Results of that simulator testing are documented in AR A0416238.

In Revision 0 of EP E-1.3, dated March 11, 1985, closure of motor-operated Isolation Valves SI-8805A/B, 8976, and 8980 were moved to the nontime-critical part of the procedure, i.e., after both RHR pumps were aligned to the suctions of the CCP and SI pumps. There is no discussion in the procedure history sheets as to why this change was made, although it may have been due to a single failure analysis of an RHR pump after the first RHR pump is aligned to the suction of the SI pumps, and the resulting loss of SI flow if the 8976 valve were closed.

Several NCRs and ARs have discussed check valve testing and design basis in the past: NCRs DC0-91-TN-N026, DC0-93-TP-N028, DC0-93-TS-N042, and DC0-93-NS-N002, ARs A0351369, A0291455, and A0315425.

Operation of the Check Valves SI-8924, 8977, and 8981 in these lines can be credited. The functional description of these valves for the Inservice Testing (IST) program is contained in NPG Calculation N-124. This calculation was prepared in response to NCR DC0-93-TP-N028, to define the IST testing criteria and basis for check valves. N-124 states for each of these valves: "This valve

has a safety function for a short period of time in the closed direction to prevent the discharge of potentially highly radioactive post-LOCA containment recirculation sump water to the RWST during the switchover from the injection phase to the cold leg recirculation phase of safety injection."

The post-LOCA recirculation leakage calculated limit into the RWST per Calculation N-169 and N-170 is 11.26 gpm. These calculations were performed to evaluate the consequences of leakage of post-LOCA recirculation fluid to the RWST. Whole body and thyroid doses to the control room, exclusion area boundary, and low population zone were calculated. It was concluded that the contribution to the dose from recirculation loop leakage to the RWST is negligible for any leakage that is likely to result from check valve leakage.

As a result of the above discussions, it was acceptable to move the closure of the RWST motor-operated valves to the nontime-critical part of EOP E-1.3. It is further acceptable that certain actions be performed outside the control room due to concerns raised relative to spurious actuation. Neither the consequences nor the likelihood of an accident are increased by these changes.

19. Transfer to Hot Leg Recirculation
EOP E-1.4, Rev. 11 (Unit 1) (LBIE Log No. 97-005)

This procedure revision removes the requirement for auxiliary saltwater (ASW) train separation. A requirement was added to contact the Technical Support Center (TSC) for an evaluation of train separation and component cooling water (CCW) train separation, contingent upon TSC direction to do so.

Safety Evaluation Summary

This procedure revision provides greater flexibility in responding to an active failure while allowing train separation to mitigate a passive failure after the first 24 hours. All affected items are used to mitigate an accident and are not considered as initiators of any accident. Therefore, there are no adverse consequences of this revision.

20. Transfer to Hot Leg Recirculation
EOP E-1.4, Rev. 4 (Unit 2) (LBIE Log No. 97-006)

Emergency Operating Procedure (EOP) 1.4 was revised so that component cooling water (CCW) and auxiliary saltwater (ASW) train separations are not required following transfer to hot leg recirculation. With both systems aligned to separate trains as required by the previous EOP revision, there was a concern that a postulated of loss of Bus F power would cause loss of containment heat removal due to loss of flow in one train and loss of ASW flow to the other CCW train. The procedure change allows realignment of ASW and CCW into

separate trains to be performed after transfer to hot leg recirculation as directed by the Technical Support Center.

Safety Evaluation Summary

This 10 CFR 50.59 safety evaluation was performed because the description of ASW and CCW train separation as described in FSAR Update Section 9.2.2, Table 9.2-7, and Section 9.2.7.2 was changed to remove the requirement for train separation following transfer to hot leg recirculation. At the same time, train separation is not prohibited as a long-term recovery action if plant configuration and operating conditions warrant the action.

Separation of the CCW and ASW trains following transfer to hot leg recirculation is a long-term recovery action following an accident, and is not related to the cause of an accident or equipment malfunction. The changes affect accident mitigation by providing greater flexibility in responding to an active failure. Train separation to mitigate a passive failure after the first 24 hours is still allowed. These actions can be taken within the timeframe specified in the FSAR Update. Consequences of an evaluated accident are not increased.

21. Revision to Emergency Operating Procedure for Transfer to Hot Leg Recirculation
EOP E-1.4, Rev. 13 (Units 1 & 2) (LBIE Log No. 97-008)

This operations emergency procedure gives the necessary sequence of steps to maintain long-term core cooling following a loss-of-coolant accident (LOCA). This procedure is implemented during the first day following an accident and it limits precipitation of coolant boron onto cores surfaces that could degrade fuel rod heat transfer.

Safety Evaluation Summary

The revision was made to address the possibility that the loss of a single vital ac power source to safety-related equipment could interrupt emergency core cooling if redundant trains of cooling are physically separated (a method for passive failure protection during the long-term core recovery process). The decision to separate trains now belongs to the site emergency organization (Technical Support Center) and will be based upon the plant conditions that exist at that time. Other alternatives for passive failure protection remain available.

22. Testing and Maintenance of Battery Pack Emergency Lights Inside Power Block
MP E-67.5A, Rev. 15 (Units 1 & 2) (LBIE Log No. 97-030)

Revised maintenance procedure to combine all battery-operated light (BOL) testing into one procedure (was previously contained in Surveillance Test Procedure (STP) M-17C1/C2A/C2B).

Safety Evaluation Summary

Testing of BOLs was previous performed using Surveillance Tests (M-17C1, M-17C2A, M-17C2B) and will now be performed by a Maintenance Procedure (MP E-67.5A). This differs from commitments described in FSAR Update Appendix 9.5B, Section C.5, which states, "Test programs are laid out in detail in surveillance test procedures and are controlled by the QA Manual...." and "Procedures governing periodic inspections are laid out in the surveillance test procedures." Change from STP to MP will require a change to the Fire Protection Plan as described in the FSAR Update. This LBIE supports the use of MP for testing and inspection.

23. Loss of All Vital AC Power
EOP ECA-0.0, Rev. 11 (Unit 1) (LBIE Log No. 97-009)

Emergency Operating Procedure (EOP) ECA-0.0 was written to address the loss of all vital ac power. It has been revised such that it may now be exited when a single vital bus is energized. The previous operating philosophy required that a complete train of emergency core cooling system equipment be restored prior to exiting ECA-0.0. The change in philosophy is supported by the Westinghouse Owner's Group determination that the availability of minimum safeguards capacity is not a requirement for being in other optimal recovery guidelines and function restoration guidelines.

Safety Evaluation Summary

The total loss of vital ac, as well as the loss of two vital buses, is beyond the single failure design basis of DCPP. Modification of EOP ECA-0.0 to allow operators to return to the recovery guidelines (E and FR series procedures) when one vital bus has been restored improves the ability to cope with this beyond design basis accident. The modification of this post-accident response procedure does not increase the probability or possibility of an accident. The purpose of ECA-0.0 is to respond to multiple failures. Exiting the procedure with only one vital bus does not ensure that further malfunctions are prevented; however, the use of the full network of emergency procedures improves the overall response to the postulated plant condition.

24. Loss of All Vital AC Power
EOP ECA-0.0, Rev. 6 (Unit 2) (LBIE Log No. 97-010)

Emergency Contingency Guideline ECA-0.0 was revised such that it will not be performed if any vital bus is energized. The intent is to cope with loss of ac emergency power until at least one emergency bus can be energized.

Prior to this change, one complete train of emergency core cooling system (ECCS) equipment was required to exit emergency contingency guidelines and return to the recovery and functional restoration (E-series and FR-series) guidelines. The operator was directed to remain in ECA-0.0 with one vital 4-kV bus energized because two buses were required to guarantee restoration of minimum safeguards equipment. However, with power restored to one vital bus, some degree of core cooling becomes available. Westinghouse Direct Work No. 92-033 clearly stated that minimum safeguards capacity is not required to be in the recovery and function restoration guidelines.

This change will allow the operators to fully use the emergency procedure network to mitigate concurrent accidents.

Safety Evaluation Summary

The safety evaluation was performed to evaluate the effect on ECCS equipment and to verify that a failure mode could not be created by implementing the E-series or FR-series emergency procedures with less than one full train of safeguards equipment.

The emergency contingency, recovery, and functional restoration procedures are used to mitigate the consequences of accidents. Their use cannot cause an accident.

All FSAR Update accidents were evaluated and validated for recovery with only one 4-kV bus available. In each case, it was determined that recovery was possible. The revised procedure allowed recovery options that were previously not available. Recovery was not impaired and consequences were not increased.

The change provides improved guidance for the operators when a vital bus is energized. Other procedures are adequate for accident mitigation. Use of the modified procedures does not increase the possibility, probability, or consequences of any equipment malfunction or accident. The margin of safety is not reduced because remaining equipment can be operated optimally to maintain core cooling.

Commitments to the NRC regarding maintenance of core cooling with ac power unavailable are maintained.

25. Service Cooling Water - Alternate Cooling Supplies to SCW Heat Exchangers
OP F-1:VI, Rev. 3 (Units 1 & 2) (LBIE Log No. 97-058)

Per the FSAR Update, saltwater is the cooling medium for the service cooling water (SCW) heat exchangers. This procedure revision allows firewater to supply cooling to the SCW heat exchangers as an alternate source during outages when the circulating water system is shut down and the auxiliary header is cleared.

Safety Evaluation Summary

During the time firewater is used to cool the SCW heat exchangers, the unit secondary side is in an outage condition. The SCW system has no effect or impact on the plant safety at this time. The low operating pressure and temperature of the system minimize the probability of line failure.

At the time firewater is used as alternate cooling, there is no safety-related equipment in the area. The firewater hose reel system is a seismically qualified system that can be isolated by sectionalizing within the plant. The physical location of lines and components cooled by the system is such that the failure would not affect any safety-related Design Class 1 equipment or components.

26. Makeup Water Sources to the CCW System
OP F-2:VII, Rev. 1 (Unit 1) (LBIE Log No. 97-131)

Instructions were added to the operating procedure to use the firewater storage tank (FWST) contents to supply the component cooling water (CCW) system if needed. The use of the FWST for CCW makeup is described in FSAR Update Section 9.2.2.3.3. Prior to this revision, such usage was not addressed in an operating procedure.

Safety Evaluation Summary

The procedure change does not affect the status of any plant system prior to an accident. The FWST is used for CCW makeup only after failure of the CCW system and several other makeup sources. The specific alignment is not associated with any FSAR Update accident; it adds an additional backup source of CCW makeup and adds flexibility to the ability of operations to mitigate FSAR Update-analyzed accidents concurrent with certain failures. Accident and equipment malfunction probabilities and consequences are not increased.

27. Makeup Water Sources to the CCW System
OP F-2:VII, Rev. 1 (Unit 2) (LBIE Log No. 97-132)

The procedure was revised to include the specific steps to align the firewater storage tank as a backup source to the CCW system, as described in the FSAR Update. The revision also requires the concurrence of Chemistry and Environmental Operations prior to using the primary water storage tank as a backup source for CCW makeup.

Safety Evaluation Summary

The changes to the procedure ensure that alternate supply sources of makeup water for the CCW system are available. The availability of backup supplies increases the reliability of the CCW system. The alignment of the backup source is already described in the FSAR Update; therefore, this change does not increase the probability or consequences of an accident or malfunction. The consequences of a potential breach of the CCW system are reduced by the flexibility of the multiple backup water sources.

28. Chemistry Control Limits and Action Guidelines for the Secondary Systems
OP F-5:II, Rev. 19 (Unit 1) (LBIE Log No. 98-037)

This procedure revision proposes to use pyrrolidine, as a pH control agent, for chemical injection into the Unit 1 secondary system during operation. Currently the procedure specifies that ethanolamine (ETA) is to be used as the pH control agent. This proposed activity is being done as an Electric Power Research Institute (EPRI) Tailored Collaboration Project to evaluate actual plant performance of an alternate amine (pH control agent) in reducing corrosion product transport. Control of pH in the secondary cycle of pressurized water reactor plants is essential to minimize corrosion of secondary system components.

Safety Evaluation Summary

The use of pyrrolidine in the condensate/feedwater system and in the steam generators will enhance equipment integrity due to reduced corrosion rates at room temperature pH in these systems. It will not adversely affect any other secondary equipment. Since the use of pyrrolidine is expected to increase system pH, the elevated pH will reduce iodine volatility to some point less than presently evaluated, thus making the current FSAR Update evaluation more conservative.

No equipment is being installed or modified in the plant as a result of the use of pyrrolidine for pH control in the secondary system. Secondary water chemistry and/or the use of chemical additives or chemical controls for the secondary cycle

is not addressed in the Technical Specifications. There is no reduction in the margin of safety as defined in the bases for any Technical Specification.

29. Chemistry Control Limits and Action Guidelines for the Secondary Systems
OP F-5:II, Rev. 15 (Units 1 & 2) (LBIE Log No. 97-056)

The procedure establishes the chemistry control limits for the condensate/feedwater and steam side of the plant. It identifies guidelines for corrective actions should limits be exceeded. This procedure revision introduces a new chemical to be used for scavenging oxygen from feedwater during system operation and from steam generator water when they are in wet lay-up during cold shutdowns. The new chemical, carbohydrazide (Nalco 1250+), supplements use of hydrazine as discussed in the procedure. Limits for its use and corrective action guidelines were provided.

Safety Evaluation Summary

This procedure revision involves a chemical that is in use at other oressurized water reactor power plants for the same functions as described above. The new chemical, carbohydrazide, was tested and evaluated to assure no material compatibility or chemical reaction issues that could contribute to corrosion that may increase the probability of or consequence of in a steam generator tube rupture, rupture of a main feedwater pipe, rupture of a main steam line or potential missiles from the main turbine.

The use of carbohydrazide, with respect to equipment important to safety, was determined to be bounded by the use of hydrazine, as described in the FSAR Update. There are no Technical Specifications associated with secondary chemistry control.

The use of carbohydrazide is not described in the National Pollutant Discharge Elimination System permit but since it is a less toxic chemical than hydrazine, as determined by testing, and non-hazardous, as identified per 29 CFR 1910.1200, prior approval was obtained from the California Regional Water Quality Control Board.

30. RHR Valves 8701/8702 Interlock Jumper Installation and Removal
MP I-38-M.1, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-046)

A procedure was written to install a jumper(s) for residual heat removal (RHR) when de-energizing the solid-state protection system (SSPS) output cabinet(s). The RHR suction valves interlock relays are powered from the SSPS output cabinets. To maintain the ability to open the RHR suction valve(s) when the SSPS output cabinet(s) are be de-energized, new Procedure MP I-38-M.1 will allow the installation of a jumper(s) to lock-in the RHR suction valve(s) open

permissive. This defeats the "block opening" interlock. Jumper installation is limited to Mode 6 and defueled only.

When installing a jumper prior to de-energizing SSPS Train B with the RCS >390 psig or pressurizer vapor space >475ºF, the "block opening" interlock for Valve 8701 is defeated.

When installing jumper prior to de-energizing SSPS Train A with the RCS >390 psig, the "block opening" interlock for Valve 8702 is defeated.

Safety Evaluation Summary

This change only applies to Mode 6 with fuel removed from the reactor vessel. Therefore, because the RCS is depressurized, the change does not involve an unreviewed safety question.

31. OP J-2:I, "Main and Aux Transformer Return to Service"
OP J-2:1, Rev. 5 XPR (Unit 1) (LBIE Log No. 96-033)

This evaluation was written for an operating procedure that reflected the operation of DCPP Unit 1 without Auxiliary Transformer 1-1, which was destroyed during an attempt to energize it with a grounding device installed. The LBIE covered the change in plant configuration, as well as operation of the plant under such conditions. The plant configuration provided startup power to the units' non-vital buses, allowing operation of the circulating water pumps and reactor coolant pumps continuously on the immediate-access offsite power source.

Safety Evaluation Summary

The evaluation documented that this configuration differed from that described in the FSAR Update, but that the consequences of a complete loss of flow (CLOF) event were not changed. Although the proposed configuration results in a slight increase in frequency of a CLOF event, it does not alter its classification as a Condition III event. An unreviewed safety question is not involved.

32. OP L-4, "Normal Operation at Power"
OP L-4, Rev. 38 (Unit 1) (LBIE Log No. 97-145)
OP L-4, Rev. 25 (Unit 2) (LBIE Log No. 97-146)

This procedure change was made to document a limitation described in FSAR Update Section 15.2.6.1. The limitation is adequately bounded by adding a restriction not to restart a reactor coolant pump while the reactor is critical.

Safety Evaluation Summary

Since the proposed change completely bounds and prevents violation of the FSAR Update-described limitation, the procedure does not involve a change to the facility design, function, or method of performing the function as described in the FSAR Update. An unreviewed safety question is not involved.

33. Plant Cooldown from Minimum Load to Cold Shutdown
OP L-5, Rev. 41 (Unit 1) (LBIE Log No. 97-041)
OP L-5, Rev. 26 (Unit 2) (LBIE Log No. 97-040)

There was a discrepancy between FSAR Update Section 5.1.6.3 and OP L-5 regarding normal plant cooldown. The FSAR Update stated that steam was dumped to the main condenser. While true, steam is also dumped to the atmosphere via the 10 percent atmospheric dump valves during normal cooldown. An FSAR Update change was submitted to clarify use of the 10 percent steam dump valves.

Safety Evaluation Summary

The 10 percent atmospheric steam dump valves are normally used to control or reduce primary temperatures if the main condenser is not available. Use of the valves during normal cooldown, in the absence of any FSAR Update-analyzed accident or event, will not increase the probability of an accident or the probability of valve failure. Similarly, steam dump valve use during normal cooldown will not increase the consequences of an accident or create the possibility of a new accident or equipment malfunction. Technical Specification (TS) design margins are not affected by atmospheric dump valve use during normal cooldown.

34. Feed and Bleed of the CCW System
PEP M-246, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-182)

This procedure de-concentrates the component cooling water (CCW) system's exhausted chemicals by continuously adding makeup water to the surge line while draining CCW from the heat exchanger through temporary connections.

Safety Evaluation Summary

Even though this procedure places the CCWS in an unanalyzed configuration, the evaluation concludes that, there are adequate, dedicated personnel stationed to isolate the temporary connections and place the CCWS in a normal line-up in the event of abnormal indications. The evaluation indicates that the most likely problem encountered with this test is leakage due to failure of the

temporary connections. The CCWS has, however, been analyzed in the FSAR Update for flooding near and leakage from the CCW heat exchangers.

35. Reactor Vessel Stud Tensioning
MP M-7.2, (Units 1 & 2) (LBIE Log No. 98-030)

The proposed change involved revising the reactor vessel stud tensioning and detensioning procedures to include new stud tensioning/detensioning sequences and corresponding tensioner pressures and revising the procedure to include a larger acceptable elongation tolerance range. As part of this optimized procedure, tensioning may be accomplished with the use of either six or three hydraulic tensioners, and may even be completed with only two tensioners in the event of a tensioner failure during the procedure.

Safety Evaluation Summary

The proposed procedure change involved no changes to the material of construction for configuration of the affected system (reactor vessel closure flange). The proposed change to the tensioning procedures has no possible impact on the analyzed fuel handling accidents. While the proposed procedure change does permit the reactor vessel studs to have a larger preload stress than has been previsouly permitted by the procedure, the closure flange and studs are demonstrated to meet acceptable ASME Code stress and fatigue limits, so there is no reduction in the margin of safety of any affected components.

36. Manual Installation of Steam Generator Nozzle Dams
MP M-7.61 (Units 1 & 2) (LBIE Log No. 97-062)

This procedure is for the inspection, installation, removal, and refurbishment of the steam generator primary nozzle dams. Use of the nozzle dams is necessary to permit performance of maintenance activities in the steam generator channel heads with the water level of the reactor coolant system (RCS) above the nozzles, such as during refueling operations. Additionally, the nozzle dams minimize the potential for the loss of foreign objects into the RCS piping from the steam generator channel head.

Safety Evaluation Summary

The design pressure of the nozzle dams exceeds both the normal and anticipated accident conditions. The seismic operating basis earthquake maximum load is less than the tested pressure of the nozzle dams. Precaution and limitation steps described in Operating Procedure A-2:III also help to ensure the design and tested loads are not exceeded. Based on the above, use of the nozzle dams have been evaluated and do not represent an unreviewed safety

question, nor do they reduce the margin of safety as defined in the licensing documents.

37. Comparison of Final Feedwater Flow Nozzles to "AMAG"
PEP M-98A, Rev. 4 (Unit 2) (LBIE Log No. 97-178)

A cross-flow ultrasonic flow meter was installed on Unit 2 to replace the failed Controlatron system. The system is mounted externally to the final feedwater header in the turbine building. Data are collected and then used to establish a correction factor for the operator heat balance, STP R-2B. This revision to the procedure relaxed the frequency of data collection and increased the data precision requirement since the plant process computer is the preferred data source.

Safety Evaluation Summary

The flow meter is used to set reactor power and, therefore, the calibration of the instrument remains safety related. Use of this procedure revision will not increase the uncertainty in reactor power above that required by Regulatory Guide 1.49. The externally mounted system will not breach any pipe should the hardware fail. In addition, reactor power will not change as a result of the system failing as there is a human/machine interface required to analyze the data before use.

38. Plant Demineralizer Media
CAP O-10, Rev. 1 (Units 1 & 2) (LBIE Log No. 98-008)

Use of up to 39 cubic feet of resin in any deborating demineralizer vessel (Units 1 and 2) is evaluated for the purpose of forced oxidation or deboration.

Safety Evaluation Summary

This change allows for an additional 9 cubic feet of resin to be loaded into a deborating demineralizer vessel for a total volume of 39 cubic feet. The additional volume will optimize cleanup and minimize radwaste during forced oxygenation of the reactor coolant system. The FSAR Update and Design Criteria Memorandum (DCM) describe the vessels as having a resin volume of 30 cubic feet. The vessel design drawing allows for a total of 39 cubic feet. The need to revise the FSAR Update and DCM to reflect the use of more than 30 cubic feet on resin in the deborating demineralizer vessels was addressed.

39. Notification of the Chemistry or Radiation Protection Sections
OP O-3, Rev. 19 (Units 1 & 2) (LBIE Log No. 97-130)

This procedure is part of the Chemistry Control Program. This revision adds a requirement that concurrence be obtained from Chemistry prior to using primary water in a nonradioactive system and specifically mentions use of primary water as makeup to the component cooling water (CCW) system. Section 9.2.2.3.3 of the FSAR Update states, "If the primary water makeup to CCWS valve is to be opened, the plant operator must obtain concurrence from the Chemistry and Radiation Protection Group."

Safety Evaluation Summary

This procedure revision is an administrative change only for the purpose of bringing this procedure into agreement with the FSAR Update. The increase in administrative control does not increase the probability of an accident or the consequences of an accident. Therefore, there is no adverse impact on current plant safety.

40. Control of Flammable and Combustible Materials
OM8.ID4, Rev. 5 (Units 1 & 2) (LBIE Log No. 97-126)

This procedure revision clarifies the controls required for combustible materials when introduced into the plant and the amount that may be introduced without being regarded as "bulk storage."

Safety Evaluation Summary

The guidance provided by this revision ensures the introduction of combustible materials will not impact DCPP's ability to achieve and maintain safe shutdown as described in the FSAR Update. There are no adverse consequences due to this revision.

41. General Authorities and Responsibilities of Operating Personnel
OP1.DC10, Rev. 4 (Units 1 & 2) (LBIE Log No. 97-118)

This procedure change added the Work Control Shift Foreman (SFM) position and related responsibilities to the shift operating personnel. FSAR Update Section 13.1.2.2.2.4 describes the shift operating personnel, including the Unit Shift Foreman. A "Work Control" Shift Foreman position was being implemented to perform some of the administrative functions of the SFM. Since an FSAR Update change was being made to support this change, a safety evaluation was performed.

Safety Evaluation Summary

Since there is no change being proposed for the duties and responsibilities of the Unit Shift Foreman (SFM) (as described in the FSAR Update), and since adding a Work Control SFM will serve to enhance the crew's ability to respond to an emergency, the proposed change will not increase the probability or consequences of an accident previously described in the FSAR Update. An unreviewed safety question is not involved.

42. Revision to Plant Administrative Procedure on the Authorities and Responsibilities of Operating Personnel
DLAP OP1.DC10, Rev. 3 (Units 1 & 2) (LBIE Log No. 96-043)

This Operations Department Administrative Procedure establishes authorities and responsibilities of plant Operators, Shift Supervisors, Shift foremen, and Shift Technical Advisors in terms of procedure usage, response to instrumentation, actions in emergencies, and other related matters.

Safety Evaluation Summary

This procedure was revised to allow the unit Shift Foreman to delegate some administrative duties to permit better control room supervision. FSAR Update Section 13.1.2.2.2.4 describes the Shift Foreman's responsibilities.

43. Control of Plant Equipment Not Required by the Technical Specifications
OP1.DC16, Rev. 2 (Units 1 & 2) (LBIE Log No. 97-102)

In letter DCL-95-222, dated October 4, 1995, PG&E submitted License Amendment Request (LAR) 97-07 that proposed to relocate several Technical Specifications (TSs) to Equipment Control Guidelines (ECGs) and to reference the ECGs in the FSAR Update. During review of LAR 97-05, the NRC staff requested PG&E to add the following wording to FSAR Update Section 16.1: "ECGs containing relocated TSs are incorporated into the FSAR Update, by reference, in Table 16.1-1. For ECGs listed in Table 16.1-1, if the equipment cannot be returned to service as required by the ECG, then a review in accordance with 10 CFR 50.59 is required."

PG&E made these changes to the FSAR Update, and incorporated them into plant procedures by revising OP1.DC16, Section 5.5, "Noncompliance with Equipment Control Guidelines," to add:

Attachment 7.2, "Equipment Control Guidelines - Technical Specifications Relocated in Accordance With NRC's Final Policy Statement on Technical Specification Improvements." The ECGs listed in Attachment 7.2 are the same ECGs listed in FSAR Update Table 16.1-1.

A requirement that if an ECG listed in Attachment 7.2 is not complied with, then prior to exceeding the completion time of any required action, a 10 CFR 50.59 evaluation must be approved by the PSRC justifying the acceptability of exceeding the completion time.

Safety Evaluation Summary

The revision to OP1.DC16 places increased administrative controls on ECGs that are relocated from TSs, but does not change the ECG requirements themselves.

44. Setting of the Centrifugal Charging Pump 2-1 Miniflow Orifice Flow Rate
PROC PEP M-223 (Units 1 & 2) (LBIE Log No. 96-046)

Erosion of the centrifugal charging pump (CCP) 2-1 recirculation line restricting orifice resulted in increased flow as measured at power. This procedure measures the flow through the CCP 2-1 recirculation orifice during the CCP full flow performance test in Mode 6. The as-found flow is throttled using a manual valve downstream of the orifice to ensure that CCP 2-1 recirculation orifice design resistance is restored, and to ensure that CCP 2-1 will pass emergency core cooling system (ECCS) flow balance (STP V-15) acceptance criteria. Throttling of the manual valve is not the designed method of preventing excess recirculation flow; therefore, PEP M-223 results in a change to the method of performing the CCP recirculation flow limiting function as described in the FSAR Update.

Safety Evaluation Summary

Throttling the manual valve in the CCP recirculation line to lower recirculation flow will prevent the CCP from exceeding the allowable total pump flow while maintaining the required pump minimum flow. Although the manual valve's design function is to isolate the recirculation line, it will perform as a throttle valve. Restoring the recirculation line design flow resistance will ensure that CCP 2-1 is available to perform its accident mitigation function, and thus maintain the ECCS flow balance.

45. Control Room Vent
AR PK15-06, Rev. 14 (Unit 1) (LBIE Log No. 97-143)

This procedure revision addressed the response to losing both subtrains of control room ventilation system (CRVS) cooling, as described in the FSAR Update. Previously, there was no guidance provided for this event, since there are four equally redundant subtrains available to perform CRVS functions.

However, to prevent the possibility of overlooking the FSAR Update described response, these actions were added to the procedure.

Safety Evaluation Summary

Since the proposed procedure revision adds the detail as described in the FSAR Update, there is no change to the facility or operation as described in the FSAR Update. The safety evaluation was performed since the FSAR Update describes response to loss of all CRVS in moderate detail. An unreviewed safety question is not involved.

46. Routine Surveillance Test of PDP 2-3
STP P-PDP-23, Rev. 5 (Units 1 & 2) (LBIE Log No. 97-209)

Operator actions to secure the positive displacement pump (PDP) upon safety injection (SI) actuation were added to allow the surveillance test procedure (STP) to be run with a potential nonconforming condition found in the inadvertent safety injection analysis assumptions.

Safety Evaluation Summary

The procedure revision allowed the PDP to be run with added compensatory measures to mitigate a nonconservative error found in the inadvertent SI analysis. Dedicated operators were required to be stationed in the control room and at the breaker cubical to secure the PDP upon a SI actuation. The inadvertent SI analysis has time critical operator actions in emergency operating procedures (EOPs) to ensure that the pressurizer safety valves will operate within their design and licensing basis. The safety evaluation was required to be performed in accordance with Generic Letter 91-18, Revision 1, which requires a 50.59 review to be performed for any interim compensatory action taken to address a degraded or nonconforming condition. Operator actions were analyzed, in accordance with Information Notice 97-78, to ensure that the actions were adequate to maintain the reactor coolant system within the limits to equipment. The operator actions were determined to be within the training and capability of the operators, and the time allowed in the analysis, and not to affect the performance of operator actions required for EOPs. Potential misoperations were analyzed and determined not to affect plant response to other transients.

47. Environmental Radiological Monitoring Procedure
RP1.ID11, Rev. 33 (Units 1 & 2) (LBIE Log No. 97-193)

In response to a finding from Nuclear Quality Services Audit 962610007, this procedure was revised to include training of personnel responsible for quality-related Radiological Environmental Monitoring Program (REMP) activities, allow

the sampling frequencies specified in Table 1 to be extended by 25 percent, and include how training of REMP personnel is to be documented.

Safety Evaluation Summary

The proposed changes to the FSAR Update only involve analysis of environmental samples for the REMP, or editorial changes that do not impact the intent of the FSAR Update. These changes are not accident related, and they do not affect accident analysis or safety-related equipment.

48. Spent Fuel Cooling System
DCM S-13, Rev. 3.2 (LBIE Log No. 96-045)

One sentence in the Design Criteria Memorandum (DCM) was replaced with a paragraph that added clarification and updated detail about the pool water design temperatures assumed in the spent fuel cooling system criticality analysis. A design memo was referenced that showed that spent fuel pool temperatures could drop to as low as 32°F, well below the minimum ultimate heat sink of the plant. Previously, 68°F was considered the design basis.

Safety Evaluation Summary

A decrease in spent fuel pool temperature has no contribution to the FSAR Update accidents assumed for the spent fuel pool. The contribution to the consequences of an accident is not increased because the resultant K_{eff} remains below the Technical Specification limit of 0.95.

D. Tests and Experiments

1. Continued Operation in Mode 3 with MSSV Inoperable
(Units 1 & 2) (LBIE Log No. 96-028)

This safety evaluation was in response to the August 10, 1996, DCPP unit trips in which main steam safety valves (MSSVs) lifted well beneath set pressure. The safety evaluation covered (in advance) the period of time during which six inoperable MSSVs were tested and reset while the units were in Mode 3. The assumption was that, should a high pressure transient occur during this period, the MSSVs would operate at a lower pressure than set pressure. This was a change in the performance of the MSSVs from that described in the FSAR Update. It was also a condition that might affect safe operation of the plant but was not anticipated or evaluated in the FSAR Update. The safety evaluation also supported a prompt operability assessment.

Safety Evaluation Summary

A lower MSSV opening pressure is a benefit for all FSAR Update accident analyses associated with overpressure protection or minimum heat removal. A review of the accident analysis profile shows that the only FSAR Update accident analyses potentially negatively impacted by the lower pressure are those associated with steam generator tube rupture (SGTR) or overcooling. In the case of SGTR, it was shown that the various effects of Mode 3 operation compared to Mode 1 caused a large enough benefit in SGTR dose and overfill calculations to offset the negative effects of a lower than anticipated lift pressure. In terms of overcooling, the low opening pressure of the MSSVs were bounded by steamline break analyses from a shutdown condition.

2. RCS Flow Calorimetric Beginning of Cycle to End of Cycle
(Units 1 & 2) (LBIE Log No. 97-003)

This procedure describes the performance of a primary to secondary calorimetric and determines the corresponding reactor coolant flowrate. This safety analysis covers the performance of this test at the end of cycle (EOC) operation versus the beginning of cycle (BOC) operation. This was generated as a non-conformance corrective action to document previous EOC testing, and does not describe the current practice of testing at BOC.

Safety Evaluation Summary

The performance of Surveillance Test Procedure (STP) R-26 was moved to EOC to reduce the bias effect of hot leg streaming on the RCS flow measurement. The movement of the STP to EOC and the consequent verification of flow at the beginning of the next cycle using plant indication resulted in larger flow

measurement random error and larger RCS loss of flow-low setpoint random error. The larger errors were determined by calculation and have been addressed in STP R-26 by the application of a flow penalty,which is used to bias the RCS measured flow in the conservative direction. This safety evaluation determined that there is no adverse effect on the DCPP LOCA and non-LOCA safety analyses as a consequence of the flow penalty method. In addition, the RCS flow elbow taps are documented to be reliable for the verification of RCS flow after restart from refueling.

The performance of STP R-26 at EOC provided the requisite level of safety and protection as prescribe in the Technical Specifications.

3. <u>Inservice Testing (IST) Program Plan (Plan) 2nd 10-Year Interval, Revision 12</u> (Units 1 & 2) (LBIE Log No. 97-050)

This revision included the following changes:

- Removed manual stroke test of RHR-8701 and -8702 from the Inservice Test (IST) Plan P
- Removed several component cooling water (CCW) Header "C" relief valves from IST Plan P
- Removed partial stroke test requirement for emergency core cooling system (ECCS) check valves from IST Plan P

<u>Safety Evaluation Summary</u>

The revision deleted manual stroke test of the residual heat removal (RHR) suction from the reactor coolant system (RCS) hot leg valves RHR-8701 and -8702. Manual operation of these valves is not required because accidents analyzed in the FSAR Update would make the containment building inaccessible and therefore these valves could not be operated post accident.

Several CCW Header 'C' relief valves were deleted from the IST Plan P. These valves do not protect vital components, and they only function as relief valves when the subsystem being protected is out of service.

The revision also included removal of the partial stroke test after disassembly requirement from the IST Plan P for several ECCS check valves. NUREG-1482 recommends the partial stroke if practical. The evaluation shows that removal of these items does not create an unreviewed safety question.

4. <u>Containment Fan Cooler Unit (CFCU) Time Delay Relays Replacement Test</u>
Preventive Mainteance Testing (PMT) 23.24-23.28 (Unit 2) (LBIE Log No. 97-129)

This item is related to Design Change Package (DCP) E-50344, which replaces the Unit 2 containment fan cooler units (CFCU) time delay timers with more accurate digital type Agastat DSC timers. This evaluation was performed to support the performance of this PMT with the plant in Mode 1.

<u>Safety Evaluation Summary</u>

Testing of the CFCU from the solid-state protection system (SSPS) using a slave relay is not described in the FSAR Update. Also, testing of the CFCU using a simulated auto-transfer signal is not described in the FSAR Update. Connecting the toggle switch across the slave relay contacts does not prevent normal operation of the SSPS slave relay or the auto-transfer relay. Because no engineered safety feature components are disabled during the performance of this test, there is no adverse effect on safety.

5. <u>Commitment Change - Revision of Corrective Action Regarding Main Steam Safety Valve Testing for Notice of Deviation (Inspection Report Nos. 50-275/96-12; 50-323/96-12)</u>
Letter DCL-97-073 (Units 1 & 2) (LBIE Log No. 97-075)

PG&E eliminated an NRC commitment on main steam safety valve (MSSV) testing.In a Safety Evaluation Report dated December 26, 1995, the NRC referenced PG&E's letter of November 1, 1995, in which PG&E described an augmented testing program for the main steam safety valves (MSSVs). PG&E stated in the letter that during the seventh refueling outages for Units 1 and 2, PG&E would obtain valve signature profiles on live steam and with the AVK test equipment. PG&E would also obtain the magnitudes of the AVK test equipment bias. PG&E's intent in collecting this data was to develop valve-specific correction factors (more recently called valve-specific mean seat areas, or MSAs) which would be used to increase the accuracy of setpoint adjustment during MSSV testing. The data to be collected and the derivation of valve specific MSAs for subsequent use in setting MSSVs would be used to establish a means for administrative control over the testing and accurate setting of MSSVs.

The commitment was eliminated because PG&E discontinued the use of the AVK test equipment and methodology (TE&M) in favor of Trevitest TE&M.

Safety Evaluation Summary

The collection of data would have been for the development of valve-specific MSAs for use with AVK TE&M. As the use of valve-specific MSAs with the AVK equipment had demonstrated the inability to assure setting of MSSVs within their Technical Specification limits, the data would not be obtained. The Trevitest equipment and the Trevitest standard MSA had been demonstrated throughout the industry to provide more accurate valve settings. Thus, the Trevitest equipment was selected for use during future MSSV testing rather than AVK. Therefore, testing and resetting of MSSVs with setpoints meeting the Technical Specification limits did not constitute an unreviewed safety question.

6. Auxiliary Saltwater (ASW) Flow Test Procedure
Surveillance Test Procedure (STP) M-26, Rev. 21 (Unit 1) (LBIE Log No. 97-195)
STP M-26, Rev. 2 (Unit 2) (LBIE Log No. 97-196)

This procedure change revised the frequency of flow testing of the ASW system from monthly to quarterly. This was a change to the frequency as stated in PG&E's responses to Generic Letters 89-13 and 91-13.

Safety Evaluation Summary

The implementation of equipment control guideline (ECG) 17.2 for continuous chlorination, operating experience and inspection results demonstrate that the ASW system is not susceptible to flow degradation due to biofouling, siltation, or coating failure over a quarterly inspection interval.

Continuous chlorination has eliminated growth of biofouling organisms in the ASW system. The design and operation of the ASW system prevent siltation of out of service or inservice piping. Inspection of the ASW pipe and heat exchangers has demonstrated that there is no degradation of the pipe liner material that could block flow.

Based on the above, the increase in the surveillance interval will not create an unreviewed safety question.

7. 4-kV Vital Bus Undervoltage Relay Calibration
Surveillance Test Procedure (STP) M-75, Rev. 18 (Units 1 & 2) (LBIE Log No. 97-165)

Prior to Revision 18, STP M-75 was performed on a deenergized bus with the plant in Mode 5 or 6. Revision 18 provides the necessary precautions and instructions for performing undervoltage relay calibration with the bus energized and the plant in Mode 1. These changes cause the associated diesel generator

and bus auxiliary transformer or startup transformer feeders to be inoperable at various times during the test. With extended fuel cycles, this change was necessary to comply with the 18-month calibration frequency required by Technical Specification Table 4.3-2, items 7a and 7b.

Safety Evaluation Summary

Performance of STP M-75 in Mode 1 is an abnormal condition compared to past practice and therefore represents a new plant configuration. Performance of STP M-75 in Mode 1 does not make the 4-kV bus inoperable. Periods of inoperability for the diesel generator and auxiliary or startup feeders are within the time constraints of Technical Specification action statements. Therefore, Technical Specification requirements and margins of safety are maintained. Performance of the test will not initiate any accidents or plant transients. All FSAR Update Chapter 15 accidents have been evaluated for occurrence with minimum safeguards equipment available. Therefore, there is no unreviewed safety question.

8. Main Steam Safety Valve Testing (STP M-77B)
STP M-77B, (Units 1 & 2) (LBIE Log No. 97-119)

Main steam safety valve (MSSV) discs were replaced with an improved material (Inconel X-750). This necessitated a change to the test procedures allowing the elimination of Mode 1 testing (20 to 30 days after an outage). The testing ensured no sticking following restart with the 422SS MSSV discs. A new STP (STP M-77B) was developed to document the Inconel X-750 test and validation plan and was also designed to minimize test cycling of the valves.

Safety Evaluation Summary

The new material with pre-oxidizing reduces the galling potential and therefore ultimately the sticking of the disc and nozzle within the MSSV. The test plan is an iterative process that ensures continuous valve performance. The reduction of testing and valve cycles serves to improve overall valve performance. Based on these considerations, the probability or consequence of accidents is not affected. Additionally, the improved performance and reduced maintenance cycling reduce the probability and consequence of an equipment failure. The alteration of the disc material and reduction of testing does not introduce new accidents.

9. Boric Acid Inventory
STP R-20, Rev. 18 (Units 1 & 2) (LBIE Log No. 97-194)

This change involved a procedure revision to Surveillance Test Procedure R-20 to implement Plant Staff Review Committee (PSRC) Technical Specification

Interpretation 97-05, which increased the minimum refueling water storage tank (RWST) level from 400,000 gal. to 443,455 gal. Also the level indicator reading error penalty was lowered from 5 percent to 2 percent.

Safety Evaluation Summary

PSRC Technical Specification Interpretation 97-05 increases the minimum RWST level to 443,455 gal. This required level increase is due to revisions to the containment recirculation sump level calculations. The level indication post-accident monitoring (PAM) calculation lowered the error penalty from 5 percent to 2 percent for RWST level switches.

10. Plant Process Computer and Manual Operator Heat Balance
Surveillance Test Procedure (STP) R-2B1, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-065)
STP R-2B1, Rev. 1 (Unit 2) (LBIE Log No. 97-066)

The proposed activity is to change the standard test procedure requirement to a check, and reset if required, of the nozzle fouling factor (NFF) for curtailments below 85 percent power. The newly proposed check determines if the existing NFF may continue to be used for the return to 100 percent power. Should the NFF not meet certain criteria, it shall be reset to unity before proceeding to 100 percent power and a new NFF will need to be calculated.

Safety Evaluation Summary

The procedural change for re-establishing or verifying the validity of the NFF does not increase the probability, or change any of the consequences, of an accident previously analyzed in the FSAR Update. This verification of the NFF only aids in assuring that the reactor is operating at the allowable thermal output as licensed.

11. Containment Fan Cooler Unit (CFCU) Timers Setting Verification
TP TB-9627, Rev. 0 (Units 1 & 2) (LBIE Log No. 96-041)

Testing of the containment fan cooler unit (CFCU) from the solid-state protection system (SSPS) using a "simulated" slave relay contact is not described in the FSAR Update. This test procedure describes use of a toggle switch for this test.

Safety Evaluation Summary

Using a simulated signal to start only the CFCU instead of the slave relay requires less equipment to be removed from service to perform this test. All ESF equipment remains in service to mitigate an accident. No engineered safety feature component is disabled for the performance of this test, so there is no

increase in the probability of an accident. Therefore, an unreviewed safety question is not involved, and a change to the DCPP Technical specifications is not involved.

12. Component Cooling Water (CCW) to Spent Fuel Pool Heat Exchanger Flow Test
TP TB-9703, Rev. 0 (Unit 1) (LBIE Log No. 97-061)

The purpose of this test is to provide data to evaluate the desirability of proposed future modifications that would increase the heat transfer capacity of the existing spent fuel pool heat exchanger. The test involves increasing the CCW flow from a design flow rate of 3000 gpm to approximately 5000 gpm for no more than five minutes.

Safety Evaluation Summary

This test will not impact the ability to remove decay heat from the reactor core or the spent fuel pool. The test can not impact the operation of the CCW system to any significant extent, nor can it impact the reactivity controls and shutdown margin. No physical modifications will be made to the plant and the design and/or function of the system has not been changed. Based on the above, there is no unreviewed safety question.

13. Cable Spreading Room Pressure Boundary Integrity Verification
TP TB-9711, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-142)

Test Procedure (TP) TB-9711 involves a pressure test on the cable spreading room boundary to determine the amount of leakage present. This test consists of isolating the heating, ventilating, and air conditioning (HVAC) to the room to simulate the conditions present during a CO_2 discharge, and then pressurizing the room to 0.04 in. - 0.06 in. water and cooling by using a fan. The amount of air leaking out of the room will be determined by the installed test equipment.

Safety Evaluation Summary

Isolating the room HVAC will result in a modest rise in ambient temperature above the initial ambient temperature of 80°F. Test conditions require that the cable spreading room be maintained below a maximum limit of 108°F. This temperature limit was conservatively chosen based upon the 120°F operating limit for the Eagle-21 system. Additionally, the maximum allowed test temperature is well below the operating limit of the reactor protection system and electrical equipment in the room. Equipment Control Guideline 23.1 limits the temperature in the cable spreading room to 119°F. In order to maintain the control room habitable during a postulated accident, the test procedure contains instructions to terminate the test in the event of an accident, and to open a door

to relieve the pressure. At all times the control room pressurization system will be capable of maintaining a pressure higher than the pressure in the cable spreading room, which will ensure that any leakage is to the outside of the control room.

14. <u>Motor-Operated Valve (MOV) Flow Test - Charging Injection Valves in Mode 5</u>
TP TB-98-2, Rev. 0 (Unit 2) (LBIE Log No. 98-031)

Surveillance Test Procedure (STP) V-15 (ECCS Flow Balance Test) is currently performed in Mode 6 with the reactor vessel head removed. Although dynamic testing of the charging injection Motor Operated Valves (MOVs) 8801A/B & 8803A/B has been previously performed at DCPP, it has been done in Mode 6 under the STP V-15 test conditions. This procedure is written to perform dynamic testing of the valves in Mode 5.

<u>Safety Evaluation Summary</u>

The valves being tested are designed to operate under the conditions established for the test. The charging system is operated within its design parameters during the test, as are the individual components within the system. No material changes are made to the system or components being tested. Low temperature overpressure protection is enabled during testing to address potential overpressurization. Residual heat removal (RHR) system operation is consistent with current Mode 5 or 6 operation of the system. By using the operable centrifugal charging pump , the required boration flowpath is maintained throughout the test. RHR flow continues to provide reactor coolant system cooling during the test.

Since no material changes are made to the system, and system and component operation is within design parameters and Technical Specification limitations, the test does not involve an unreviewed safety question.

E. Equipment Control Guidelines

1. Administrative Changes to ECGs
   ECG 0.0, Rev. 3 (LBIE Log No. 97-103)
   ECG 7.7, Rev. 1 (LBIE Log No. No. 97-109)
   ECG 17.3, Rev. 1 (LBIE Log No. No. 97-110)
   ECG 23.1, Rev. 1 (LBIE Log No. 97-111)
   ECG 39.6, Rev. 1 (LBIE Log No. 97-112)
   ECG 99.9, Rev. 1 (LBIE Log No. 97-113)
   ECG 4.3, Rev. 1 (LBIE Log No. 97-104)
   ECG 7.3, Rev. 2 (LBIE Log No. 97-105)
   ECG 7.4, Rev. 1 (LBIE Log No. 97-106)

   In PG&E's Letter DCL-95-222, dated October 4, 1995, PG&E submitted License Amendment Request (LAR) 97-07 that proposed to relocate several Technical Specifications (TSs) to Equipment Control Guidelines (ECGs) and to reference the ECGs in the FSAR Update. During review of LAR 97-05, the NRC staff requested PG&E to add the following wording to FSAR Update Section 16.1: "ECGs containing relocated TSs are incorporated into the FSAR Update, by reference, in Table 16.1-1. For ECGs listed in Table 16.1-1, if the equipment cannot be returned to service as required by the ECG, then a review in accordance with 10 CFR 50.59 is required."

   PG&E made these changes to the FSAR Update, and incorporated them into the existing ECGs listed in FSAR Update Table 16.1-1 by adding the following note to each ECG: "Prior to exceeding the Completion Time of any Required Action, a 10 CFR 50.59 evaluation must be approved by the PSRC justifying the acceptability of exceeding the Completion Time."

   Safety Evaluation Summary

   The revision to ECGs relocated from TSs places increased administrative controls on obtaining approval to exceed completion times for required actions but does not change the ECG requirements themselves. By requiring Plant Staff Review Committee approval of a 10 CFR 50.59 safety evaluation for an extension of a completion time for a required action, assurance is provided that the extension will not involve an unreviewed safety question.

2. Equipment Control Guidelines Applicability
   ECG 0.0 & 0.0, Rev. 2 (Units 1 & 2) (LBIE Log No. 97-029)

   This issue concerns opposite wording between Equipment Control Guidelines (ECG) 0.4 and Technical Specification (TS) 3.0.4 regarding allowance of mode transitions. The less restrictive wording of ECG 0.4 could have permitted mode transitions which would not have been permitted under the more restrictive

wording of TS 3.0.4. However, a review identified no actual instances where the plant has been in an ECG shutdown action statement. The ECG 0.4 wording problem will be corrected by a revision to the ECG.

Safety Evaluation Summary

Because this change is a revision to an ECG, Procedure OP1.DC16 requires a 50.59 safety evaluation. The ECG revision involved revising ECG 0.4 to more restrictive language regarding allowance of mode transitions to agree with TS 3.0.4. The probabilities of occurrence of an accident or consequences are not increased. The probability of occurrence of a malfunction of equipment important to safety previously evaluated is not increased. A new type of accident is not created. The margin of safety as defined in the Technical Specifications is not reduced.

3. Post Accident Sampling System
ECG 11.1, Rev. 5 (Unit 1) (LBIE Log No. 97-004)

This Equipment Control Guideline (ECG) revision changes the Mode Applicability requirements for monitoring reactor coolant for dissolved hydrogen from "with fuel in containment" to "Modes 1 through 4." In addition, this revision clarifies the Technical Specification bases to reference the Unit 1 PG&E letter to NRC and reference Surveillance Test Procedure (STP) G-14, Rev. 1, as the original acceptance criteria document for post-accident sampling.

Safety Evaluation Summary

This ECG revision is an administrative change only, and does not involve any physical changes to the post-accident sampling system (PASS) or dissolved hydrogen monitors. It only affects the mode applicability requirements of a plant monitoring instrument. The dissolved hydrogen monitors will continue to function post-loss-of-coolant accident in accordance with NUREG-0737 and Regulatory Guide 1.97 when indication of dissolved hydrogen content in the reactor coolant system (RCS) is required for accident monitoring purposes.

The capability to obtain and analyze RCS samples under accident conditions will be maintained as a result of this ECG revision. This change does not affect the operation of any safety-related systems or equipment and does not introduce any new failure modes for any equipment in the PASS or any safety-related system component or equipment. This change does not affect Technical Specification 6.8.4.e, "Post accident Sampling." Therefore, there is no reduction in the margin of safety.

4. <u>Post Accident Sampling System</u>
ECG 11.1, Rev. 6 (Units 1 & 2) (LBIE Log No. 97-170)

The revision to Equipment Control Guideline (ECG) 11.1 added a note to the actions to include the reporting requirement in Facility Operating License DPR-80, Section 2.G. In accordance with the provisions of 10 CFR 50.72, the NRC must be notified within 24 hours when the post accident sampling system and its alternate sampling methods are not available, with written follow-up in accordance with the provisions of 10 CFR 50.73.

<u>Safety Evaluation Summary</u>

A safety evaluation was performed because revision of an ECG requires a safety evaluation. This revision is an administrative change to make the reporting requirement more visible to the Operations staff. The reporting requirement was removed from Surveillance Test Procedure G-4 and placed in ECG 11.1 and Procedure XI1.ID2.

5. <u>Equipment Control Guideline: CCW Surge Tank Pressurization System</u>
ECG 14.1, Rev. 2 (Units 1 & 2) (LBIE Log No. 97-074)

ECG 14.1 was developed and implemented in 1996 to place administrative controls on the newly installed component cooling water (CCW) surge tank pressurization system. In 1997, PG&E elected to convert the ECG to a technical specification and submitted License Amendment Request (LAR) 97-05 to accomplish this (Reference DCL-97-074, dated May 22, 1997). In developing the LAR, a more conservative allowed outage time (AOT) was defined, and an additional surveillance requirement was identified. Revision 2 to ECG 14.1 was issued to have the ECG requirements coincide with the proposed LAR.

<u>Safety Evaluation Summary</u>

Revisions 0 and 1 to ECG 14.1 were approved to establish controls to assure that the CCW pressurization system and the CCW system are capable of performing their required functions. Revision 2 to ECG 14.1 reduces the AOT for the surge tank pressurization system from 7 days to 12 hours based on a probabilistic risk assessment. The CCW pressurization system is designed to mitigate an accident and is not an accident initiator. ECG 14.1 establishes administrative requirements and the proposed changes, including the reduction in the AOT, will not affect the method of operation of the CCW system or the method by which the CCW system performs its function.

6. Fire Suppression Systems/Fire Suppression Water Systems
ECG 18.1, Rev. 4 (Units 1 & 2) (LBIE Log No. 97-199)

This Equipment Control Guideline (ECG) revision extends the urveillance frequency for testing fire suppression valves as specified in the ECG Surveillance Requirement 18.1.8 (valves not testable during plant operation) from 18 months to 24 months to be consistent with 24-month fuel cycles.

Safety Evaluation Summary

This change does not alter the way any important to safety structure, system, or component functions, and does not change the manner in which the plant is operated. Increasing the surveillance interval of the fire water valves not testable during plant operation will not alter the operation of the fire water system or the intent or method by which the surveillance is presently conducted. The operability of the fire water valves is not affected by the surveillance interval change as these valves are maintained in the sealed open position to ensure a water source is always available.

This change does not result in a physical modification to either the valves or any important-to-safety system, structure, or component. The fire water valves and fire water system are not addressed in any Technical Specifications or associated with any margin of safety. Since the increased surveillance frequency does not impact the operation of the fire water system, this change does not involve a reduction in margin of safety as defined in the Technical Specifications.

7. Fire Hose Stations
ECG 18.2, Rev. 5 (Units 1 & 2) (LBIE Log No. 97-200)

Revision to Equipment Control Guideline (ECG) 18.2 changes the surveillance requirement frequency of fire hose inspections for hose stations not accessible during plant operation from 18 months to at least once per refueling interval (24 months).

Safety Evaluation Summary

With the inspection interval extended 6 months, the hoses and valves remain qualified to perform their intended functions. As a result, inadvertent introduction of fire water into containment is not considered a credible event due to hose aging and subsequent failure.

8. Fire Hose Stations
ECG 18.2, Rev. 4 (Units 1 & 2) (LBIE Log No. 97-173)

Equipment Control Guideline (ECG) 18.2 Surveillance Requirement 18.2.6 required that hose hydrostatic tests be conducted every 3 years at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater. This change would revise the hydrostatic test pressures to conform with the requirements of National Fire Protection Association (NFPA) Standard 1962, "Care, Use and Service Testing of Fire Hose Including Couplings and Nozzles." NFPA 1962 requires that the hoses be tested to 150 psig, consistent with the current requirements.

Safety Evaluation Summary

Fire hoses are not accident initiators, therefore the probabilities of occurrence and the consequences of an accident are not increased. The design of any equipment important to safety, the method by which any equipment important to safety performs its required function, and the operation of equipment important to safety are not affected. Therefore, no accident consequences are increased, no new accidents are created, no new types of equipment malfunctions created and there is no reduction in the margin of safety as defined in the basis for any Technical Specifications.

9. Fire Hose Stations
ECG 18.2, Rev. 3 (Units 1 & 2) (LBIE Log No. 96-007)

The revision to Equipment Control Guideline (ECG) 18.2 on Fire Hose Stations added operability requirements for two existing fire hose stations located in the intake structure. The fire hose stations are now credited as part of the fire protection for 10 CFR 50, Appendix R circuits to the auxiliary saltwater pumps.

Safety Evaluation Summary

The revisions made to the ECG provide guidance in controlling the operation of the fire hose stations located in the intake structure. Accidents analyzed in the FSAR Update are unaffected and operability of equipment important to safety is not impacted. Therefore, the change does not involve an unreviewed safety question.

10. Fire Detection Systems
ECG 18.3, Rev. 3 (Units 1 & 2) (LBIE Log No. 96-008)

This revision to Equipment Control Guideline (ECG) 18.3 on Fire Detection Systems changed a note regarding the heat sensors that actuate the carbon dioxide flooding systems in the intake structure. The note changed a reference to a new section that was added to ECG 18.5 (CO_2 Systems). The CO_2 System and its heat sensor are now credited as part of the fire protection for 10 CFR 50. Appendix R circuits to the auxiliary saltwater pumps.

Safety Evaluation Summary

The changes made to the ECG provide administrative controls for fire protection features credited to protect one train of redundant safe shutdown circuits. By including these fire protection features into ECGs, adequate compensatory measures are provided should these systems be impaired. The addition of fire suppression and detection systems into existing ECGs does not affect the function of equipment important to safety. By including these fire protection features into the ECGs, stricter surveillance requirements are implemented and compensatory measures are implemented should these systems become impaired. Based on the above criteria and justification, an unreviewed safety question is not involved.

11. Spray and/or Sprinkler Systems
ECG 18.4, Rev. 2 (Units 1 & 2) (LBIE Log No. 97-192)

This Equipment Control Guideline (ECG) deletes ECG Surveillance Requirement (SR) 18.4.5. The current requirements specify cycling each valve in the flow path that is not testable during plant operation. The only plant valves in these systems that fall into this category are located inside the containment. These valves are already covered by ECG 18.1.8. Thus ECG 18.4.5 is a duplicate and is not required.

Safety Evaluation Summary

This change does not affect the way any important to safety structure, system, or component functions, nor does it change the manner in which the plant is operated. The elimination of the surveillance requirement of ECG 18.4.5 for the fire water valves does not have any impact on plant equipment because it is a duplicate of the requirements of ECG 18.1.8. The operability of the fire water valves is not affected by eliminating this surveillance requirement because the only valves not testable during plant operation are located inside the containment and are covered by ECG 18.1.8.

The fire water valves and fire water system are not addressed in any Technical Specification or associated with any margin of safety. This change does not impact the operation of the fire water system or the fire water valves. Therefore, it does not involve a reduction in the margin of safety as defined in the Technical Specifications.

12. CO_2 Systems
ECG 18.5, Rev. 3 (Units 1 & 2) (LBIE Log No. 97-201)

This Equipment Control Guideline (ECG) revision extends the frequency for ECG Surveillance Requirement (SR) 18.5.2.5 from 18 months to at least once per refueling interval (24 months). Performance of this SR during refueling outages is specified due to the personnel safety associated with entering the circulating water pump (CWP) motor cubicle to perform this SR during power operation.

Safety Evaluation Summary

The high pressure CO_2 system is designed to mitigate a fire and is not an accident initiator. The system is provided to suppress oil fires internal to the CWP motor housing. This change does not alter the way any important-to-safety structure, system, or component functions, nor does it change the manner in which the plant is operated. Increasing the surveillance interval for the high pressure CO_2 detection system does not result in any physical modifications to either the detection system or any important to safety structure, system, or component. This change does not alter the intent or method by which the surveillance is presently conducted.

The high pressure CO_2 detection system is not addressed in any Technical Specification or associated with any margin of safety. The increased surveillance frequency does not impact the operation of the detection or suppression system. Therefore, this change does not involve a reduction in margin of safety as defined in the basis for any Technical Specification.

13. CO_2 Systems
ECG 18.5, Rev. 2 (Units 1 & 2) (LBIE Log No. 96-009)

This revision Equipment Control Guideline (ECG) 18.5 on carbon dioxide systems added operability requirements for the high pressure CO_2 system at the intake structure. The CO_2 System is now credited as part of the fire protection for 10 CFR 50, Appendix R circuits to the auxiliary saltwater pumps.

Safety Evaluation Summary

The high pressure CO_2 system in the intake structure is credited to suppress the combustible materials located in the circulating water pump (CWP) motor housings. This suppression system protects at least one train of auxiliary saltwater (ASW) pump and exhaust fan circuits against the highest fire hazards in the area. As a backup suppression system, a sprinkler nozzle is also installed above one train of ASW pump and exhaust fan circuits to ensure that one train will remain available for safe shutdown in the event of a fire. Either the CO_2

system or the sprinkler head, in conjunction with local smoke detection, manual fire suppression, and "No Storage of Combustible Material" zone, will provide adequate protection such that one train of ASW pump and exhaust fan circuits will be available for safe shutdown in the event of a fire in Fire Area IS-1/Fire Zone 30-A-5 (Reference DCPP Fire Hazards Appendix R Evaluation (FHARE) 110 and FSAR Update Appendix 9.5A).

14. <u>Multiple Procedure Change</u>
ECG 18.6 - Recission, ECG 18.1, Rev. 3, and ECG 18.1, Rev. 4, (Units 1 & 2) (LBIE Log No. 97-174)

Equipment Control Guideline (ECG) 18.6 was rescinded because the Halon fire suppression system was removed from the solid state protection system (SSPS) room. ECG 18.1 was revised to delete the reference to Halon. The detection panel alarms, and ventilation damper controls associated with the Halon suppression system were not removed. Surveillance testing of this part of the system was unchanged. The testing requirements were relocated to ECG 18.3.

The Halon fire suppression system was removed from the SSPS room because the Halon fire suppression system is no longer necessary to provide automatic fire suppression to the SSPS room. Current analysis assumes the loss of an SSPS train as the result of a fire. Manual actions and redundant safe shutdown components are credited for mitigating the effects of a fire in this area. Halon is no longer commercially available due to environmental concerns over release of hydrofluorocarbons into the atmosphere.

Physical changes were made by DCP M-049295/050295, Rev. 0.

<u>Safety Evaluation Summary</u>

This 10 CFR 50.59 safety evaluation was prepared because OP1.DC16 requires a safety evaluation for all ECG revisions.

In a meeting on January 30, 1997, the NRC concurred with PG&E's approach to evaluate and remove the Halon system based on DCPP license conditions and on performance of a 50.59 safety evaluation.

The safety evaluation performed for this ECG change is based on the evaluation performed for Design Change Packages (DCPs) M-049295 and M-050295 and approved by the DCPP Plant Staff Review Committee (PSRC) on August 13, 1996.

The Halon fire suppression system is designed to mitigate the consequences of a fire. Removal of the SSPS room Halon suppression system does not change combustible loading or ignition controls in the SSPS room. The design basis

accident is a fire, and removal of the SSPS room will not increase the probability of a fire in the SSPS room or any other fire area.

Consequences of a fire in the SSPS room are not increased by removal of the Halon system because current analysis assumes loss of a SSPS train as a result of a fire. The safe shutdown analysis credits operators with manually tripping the reactor from the control room. DCPP Fire Hazards Appendix R Evaluation (FHARE) 112 credits manual actions and redundant safe shutdown components for mitigating the effects of a fire in this area, and for providing the capability to achieve and maintain safe shutdown.

Possibility, probability, and consequences of other accidents or equipment malfunctions are not increased. Once the reactor is tripped, equipment in the SSPS is not required to maintain safe shutdown. If there is a fire in the SSPS room, FHARE 112 credits operator action to trip the reactor from the control room.

15. Fire Rated Assemblies
ECG 18.7, Rev. 2 (Units 1 & 2) (LBIE Log No. 97-202)

This Equipment Control Guideline (ECG) revision modifies Surveillance Requirement (SR) 18.7.1 to extend the frequency for inspecting fire rated assemblies inside containment from every 18 months to every 24 months to be consistent with 24 month refueling cycles. Performance of this surveillance is necessary due to the rated enclosures being located in the containment and as low as is reasonably achievable (ALARA) considerations.

Safety Evaluation Summary

Fire rated assemblies are designed to mitigate the consequences of fires and do not cause an accident. This change does not alter the way any important-to-safety structure, system, or component functions, nor does it change the manner in which the plant is operated. Increasing the surveillance interval for the containment fire related enclosures does not result in any physical modifications to either the barrier or any important to safety structure, system, or component. This change does not alter the intent, scope, or method by which the surveillance is presently conducted.

The fire rated assemblies are not addressed in any Technical Specification (TS) or associated with any margin of safety. The increased surveillance frequency does not impact the operation of any fire protection system or component, and does not affect the operation of equipment protected by the enclosure. Therefore, this change does not involve a reduction in margin of safety as defined in the TS.

16. Radioactive Liquid Effluent Monitoring Instrumentation
ECG 39.3, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-148)

This Equipment Control Guideline (ECG) changes the channel check frequency for the Oily Water Separator (OWS) Flow Recorder (FR)-251, as required by Surveillance Requirement (SR) 39.3.1, from "24 Hours" to "daily." This change is needed to eliminate some unnecessary OWS manual pump downs that are done only to perform the FR-251 channel check.

Safety Evaluation Summary

Changing FR-251 channel check frequency from 24 hours to daily does not change the way that FR-251 performs its required function. FR-251 does not initiate an accident, but is intended to provide a flow measurement in the event radioactive material is released due to some other condition in the plant. The daily channel check will provide, over time, the same number of required periodic channel checks as the current requirement.

The maintenance history of FR-251 shows that this instrument is very reliable. Since corrective maintenance was last performed on the instrument in 1992, it has passed all of its quarterly functional surveillance tests satisfactorily. Consequently, FR-251 is not expected to fail in between channel checks, even if they are performed as much as 48 hours apart over a two day period. There are no Technical Specification (TS) requirements or TS bases for the OWS. Therefore, there is no reduction in the margin of safety.

17. ATWS Mitigation System Actuation Circuitry
ECG 4.1, Rev. 2XPR (Units 1 & 2) (LBIE Log No. 97-188)

This Equipment Control Guideline (ECG) revision changes the applicability from Mode 1 above 40 percent turbine power to Mode 1 above 40 percent reactor power. This change brings ECG 4.1 into compliance with the licensing basis for the anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) system. The licensing basis is contained in the safety evaluation for the AMSAC system. The safety evaluation assumes AMSAC is armed when reactor power is above 40 percent. The current AMSAC arming setpoint (C-20) of 40 percent turbine power is equivalent to approximately 46 percent reactor thermal power (RTP). The current AMSAC setpoint is within the design limits of the system, but is not within the licensing basis which assumes 40 percent RTP.

Safety Evaluation Summary

AMSAC is designed to mitigate the consequences of an ATWS event and is not an accident initiator. An ATWS event is not a design basis event analyzed in

FSAR Update Chapter 15, but AMSAC is a licensing requirement specified by 10 CFR 50.62 and is documented in the FSAR Update. This change does not affect the design or configuration of AMSAC, but changes the alarming setpoint to agree with the AMSAC safety evaluation. There is no change in the way AMSAC operates other than the power level at which it becomes armed. This change restores the assurance that the fuel cladding will perform its required function. This change reduces the potential for voiding in the reactor vessel in the event of an ATWS, thereby assuring that the fuel remains cooled.

AMSAC is not addressed in the Technical Specifications, but this change lessens the consequences of an ATWS event and increases the margin of safety.

18. <u>Steam Generator Level and Pressure Instruments (Appendix R) - Surveillance Requirement 4.2.2</u>
ECG 4.2, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-159)

This Equipment Control Guideline (ECG) change revises the steam generator (SG) level and pressure instruments surveillance requirements (SRs) to change the surveillance frequency from at least once every 18 months to at least once per refueling interval. The SG instrumentation covered by this ECG provides alternate monitoring and indication capability in the event of a fire in the control room or the cable spreading room.

<u>Safety Evaluation Summary</u>

This change does not alter the way any important to safety structure, system, or component (SSC) functions, nor does it change the manner in which the plant is operated. There are no physical modifications to either the SG level and pressure instruments or any important-to-safety SSC. Increasing the surveillance interval does not alter the operation of the instruments, the intent or the method by which the surveillance is conducted, or the scope or intent of the associated surveillance test procedures. It does not adversely affect safety function performance, or alter the intent or method by which surveillance tests are performed.

The instruments will continue to effectively perform their design function for the longer operating cycles, and there is no time dependency associated with the encountered component failures. There is inherent substantial redundancy and other periodic checks that help ensure sufficient availability of these instruments to perform their design functions. These instruments are not required to be operable by the Technical Specifications. Therefore, this change does not involve a reduction in the margin of safety.

19. Technical Support Center ERDS
ECG 52.2, Rev. 2 (Units 1 & 2) (LBIE Log No. 96-031)
ECG 52.3, Rev. 1 (Units 1 & 2) (LBIE Log No. 96-032)

Design Change Packages (DCPs) J-49246 and J-50426 establish the emergency response facility data system (ERFDS) within the plant process computer (PPC). The PPC performs the functions of data acquisition, display, recording and recall as required by DCPP commitments to NUREG-0696 and NUREG-0737, Supplement 1. This.change is made to upgrade the data systems with equipment that performs the required functions, with high reliability.

Safety Evaluation Summary

A PPC subsystem, ERFDS, is a monitoring system isolated from the plant input instrumentation. These systems do not control plant equipment and are electrically isolated from data input uses. The PPC/ERFDS provide data to the control room, technical support center, and emergency operations facility during accident response and is used as input in decision-making following accidents. The PPC/ERFDS cannot create an accident or cause a malfunction of equipment important to safety. There is no decrease in the margin of safety.

20. Reactor Coolant System (RCS) Instrumentation (Appendix R)
ECG 7.1, Rev. 2 (Units 1 & 2) (LBIE Log No. 97-150)

This Equipment Control Guideline (ECG) change revises the reactor coolant system (RCS) instrumentation (Appendix R) ECG Surveillance Requirement (SR) 7.1.2 to change the surveillance frequency from at least once every 18 months to at least once per refueling interval. The pressurizer pressure and pressurizer level instrument channels covered by this ECG are located on the dedicated shutdown panel (DSP). These devices provide alternate monitoring and indication capability in the event of a fire in the control room or the cable spreading room.

Safety Evaluation Summary

This change does not alter the way any important-to-safety structure, system, or component (SSC) functions and does not change the manner in which the plant is operated. There are no physical modifications to the instrument channels, or to any SSC. Increasing the surveillance interval does not alter the operation of these instrument channels, the intent or the method by which the associated surveillances are conducted, or the scope or intent of the associated surveillance test procedures. It does not affect safety function performance, or alter the intent or method by which surveillance tests are performed.

The instrument channels will continue to effectively perform their design function for the longer operating cycles. There are no indications that any of the identified RCS instrumentation failures are cycle-length dependent. There is no safety analysis impact since this change does not affect any safety limit, protection system setpoint, or limiting condition for operation.

21. <u>Power-Operated Relief Valve (PORV) Emergency Close at the Hot Shutdown Panel (HSP) (10 CFR 50, Appendix R) - Surveillance Requirement 7.2.1</u>
ECG 7.2, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-151)

This Equipment Control Guideline (ECG) change revises the PORV emergency close at the HSP (Appendix R) ECG surveillance requirement (SR) to change the surveillance frequency from at least once per 18 months to at least once per refueling interval. The PORV emergency close control circuits covered by this ECG provide an alternate method of closing the PORVs in the event of a fire in the control room and/or the cable spreading room.

<u>Safety Evaluation Summary</u>

This change does not alter the way any important to safety structure, system, or component (SSC) functions and does not change the manner in which the plant is operated. There are no physical modifications to the control circuitry, or to any SSC. Increasing the surveillance interval does not alter the operation of these control circuits, the intent or the method by which the associated surveillances are conducted, or the scope or intent of the associated surveillance test procedures. It does not affect safety function performance, or alter the intent or method by which surveillance tests are performed.

The PORV emergency close control circuits will continue to effectively perform their design function for the longer operating cycles. No failures have been encountered during the functional testing of this circuitry, and there is no evidence that the performance of these control circuits is time dependent or that the longer surveillance interval will adversely affect the performance of these switches.

22. <u>Safety Valves - Shutdown</u>
ECG 7.3, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-079)

Revision 1 to Equipment Control Guideline (ECG) 7.3 updates the ECG Bases to document the current method for testing pressurizer safety valves. The valves are tested in accordance with Westinghouse report WCAP 12910, Rev. 1A, "Pressurizer Safety Valve Set Pressure Shift," dated March 1993.

Safety Evaluation Summary

Revision 1 to ECG 7.3 updates the ECG Bases to reflect the current approved method for pressurizer relief valve testing. The revision does not modify the ECG itself. The ECG 7.3 requirements were relocated from Technical Specifications as approved by License Amendments 98/97. Therefore the ECG Bases revision does not involve an unreviewed safety question.

23. Positive Displacement Pump (PDP)
ECG 8.1, Rev. 3 (Units 1 & 2) (LBIE Log No. 97-171)

This Equipment Control Guideline (ECG) revises the allowed outage time (AOT)from 7 days to 14 days each calendar year to perform overhaul work on the positive displacement pump (PDP). A 7-day AOT does not allow sufficient time to perform an overhaul, without obtaining an AOT extension.

Safety Evaluation Summary

The increased AOT from 7 days to 14 days once per year to overhaul the pump does not change the operating methods or practices for the PDP. The increased AOT does not result in any changes to hardware or equipment associated with the PDP. It will continue to operate and be operated as it is currently. The PDP is not an accident initiator, but is designed to allow the plant to achieve safe shutdown if a fire were to occur in the centrifugal charging pump (CCP) room such that both CCPs were disabled. The increased AOT to allow the performance of maintenance will increase the overall reliability of the PDP.

The PDP is not Technical Specification required equipment and is not credited in any FSAR Update Chapter 15 design basis accident analysis as mitigation equipment. Therefore, the increased AOT will not result in a reduction in the margin of safety as defined in the basis for any Technical Specification.

24. Centrifugal Charging Pump (CCP) Backup Firewater Cooling
ECG 8.3, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-009)

This change creates a new Equipment Control Guideline (ECG) to provide administrative controls for backup firewater cooling to CCPs. An analysis that supports the DCPP Security Plan takes credit for the CCP backup firewater cooling system to meet vehicle barrier security requirements.

Safety Evaluation Summary

The CCP backup firewater cooling system is a design feature that allows the CCPs to provide cooling to the reactor coolant pump (RCP) seals in the event of complete loss of auxiliary saltwater (ASW) system. It is a mitigation system and

not an accident initiator. However, it is not credited in the FSAR Update for mitigating any accident.

This new ECG places existing administrative controls for the CCP backup firewater cooling system, plus a new monthly surveillance requirement (visual inventory), under the controls of the ECG program specified by procedure OP1.DC16. The only equipment important to safety affected by these controls are the CCPs.

Creation of the ECG and its associated surveillance requirements (SRs) provides assurance that consequences of loss of both ASW trains due to vehicle damage of the trains will not increase the probability or the consequences of a small break loss-of-coolant accident by assuming that this backup method of cooling the CCPs, and consequently the RCP seals, is available. None of these controls are included in Technical Specifications. Therefore, there is no reduction in margin of safety as defined in the basis for any Technical Specification.

25. <u>Safety Injection – Accumulator Pressure and Water Level Instrumentation</u>
ECG 9.1, Rev. 1 (Units 1 & 2) (LBIE Log No. 96-040)

This Equipment Control Guideline (ECG) revision makes a distinction between surveillance requirements for narrow range (NR) and wide range (WR) safety injection (SI) accumulator water level channels. Previously, ECG Surveillance Requirement (SR) 9.1.1 required a channel functional test (CFT) on a 31-day frequency for pressure and NR level channels, but did not apply to WR level. However, WR level is part of Regulatory Guide 1.97 post accident monitoring (PAM) requirements. Other PAM instruments receive a channel calibration on a refueling outage frequency (18 months in Rev. 1; 24 months in Rev. 2). SR 9.1.2 was clarified to add WR level channel calibration on an 18-month frequency.

<u>Safety Evaluation Summary</u>

The SI accumulator WR level channels are for PAM indications only. They do not provide input to any engineered safeguards features (ESF) function required for accident mitigation. Neither can the channels initiate any accident. The channels are used following an accident to provide a qualitative indication of whether an accumulator has injected. The indication itself has no effect on the consequences or management of an accident.

Clarification of SI accumulator WR level surveillance requirements has no impact on the frequency or consequences of any accident or equipment malfunction as evaluated in the FSAR Update. The WR level channels are not part of any Technical-Specification-required function and are not used to

establish operating level in the accumulators; there is no impact on any Technical Specification safety margin.

26. Snubbers
ECG 99.1 (Units 1 & 2) (LBIE Log No. 97-085)

This Equipment Control Guideline (ECG) change was performed to alter the snubber functional testing frequency from 18 to 24 months during shutdown to coincide with the 24-month fuel cycle.

Safety Evaluation Summary

This change in surveillance frequency was determined to be in compliance with the ASME/ANSI OM Part 4, OMa-1988 addenda to the OM-1987 Edition, which states: "Testing shall take place at least every refueling outage using a sample of snubbers." In 1993 an ASME/ANSI OM-4 task group completed a review of the surveillance frequency, in light of reactor facilities extending fuel cycles to 24 months and concluded the code is applicable for the 24-month cycle. A review of the test results from the most recent nine refueling outages at DCPP showed a test failure rate of less than 1 percent and in all those cases the piping had remained operable with the failed snubber. This low testing sample failure rate and the ANSI OM-4 study showed that the proposed frequency will maintain a high confidence level in snubber operability.

27. Snubbers
ECG 99.1, Rev. 3 (Units 1 & 2) (LBIE Log No. 97-187)

This Equipment Control Guideline (ECG) revises the surveillance frequency for functional testing and service life monitoring of snubbers, as specified in Surveillance Requirements (SRs) 99.1.3 & 99.1.7, from 18 months to 24 months to be consistent with 24-month fuel cycles.

Safety Evaluation Summary

Snubbers are designed to limit pipe movement during design basis seismic events and are not accident initiators. This change does not affect the way in which the snubbers operate. Also, this change complies with the ASME/ANSI OM Part 4 code requirements.

A review of snubber operational, maintenance, and surveillance testing history has demonstrated that the snubbers are reliable and can be expected to perform their required function when tested on a 24-month interval. The snubber test failure rate has been less than 1 percent over the past nine refueling outages and in each case of snubber test failure, the piping analysis has shown that the piping remained operable. This low sample test failure rate combined with

additional tests performed for service life monitoring provides a high confidence level in snubber operability using the sampling test methods prescribed in ASME/ANSI OM Part 4. Therefore, this change does not involve a reduction in margin of safety as defined for any Technical Specification.

F. FSAR Update Changes

1. NIS Detector Description
FSAR Update 4.4.5.3 (LBIE Log No. 98-002)

FSAR Update Section 4.4.5.3 describes the positioning of the nuclear instrumentation system (NIS) detectors around the reactor core. This section states, "The two positions opposite the other two flat positions of the core are spare instrumentation wells." These instrumentation wells in fact house the post-accident neutron flux monitor detectors.

Safety Evaluation Summary

This FSAR Update change corrects the description of the NIS detector well arrangement. No physical modifications or design changes are required. There is no accident evaluated in the FSAR Update that is affected by changing the description of the NIS detector placement. Equipment operation and function remains the same. No technical specifications changes are required. The margin of safety is not affected by changing the NIS detector placement description in the FSAR Update.

2. Change Testing Methodology for P-8 Blocking Function
FSAR Update 7.2.2.2.1.7(2) (LBIE Log No. 98-001)

The testing description in FSAR Update Section 7.2.2.2.1.7 was changed to reflect the manner in which the P-8 blocking function is tested. This was required when it was discovered that the P-8 block is not tested by the SSPS semi-automatic tester. A commitment was made in PG&E letter DCL-97-172 to test the P-8 function using other means, a description of which was added to FSAR Update Section 7.2.2.2.1.7.

Safety Evaluation Summary

The FSAR Update change clarifies the test method described in the FSAR Update. The additional testing to verify the P-8 block function is performed when the solid-state protection system (SSPS) train is removed from service. No accidents previously evaluated are affected by clarifying the SSPS test methodology. Since testing is performed with the SSPS train out of service the probability of a malfunction of equipment important to safety is not increased. No physical modifications are performed. No Technical Specification changes are required. The additional testing satisfies the requirements of Technical Specification 4.3-1, item 22.

3. Subcooled Margin Monitor
AR A0427405 (LBIE Log No. 97-067)

This change corrects information in FSAR Update Table 7.5-5, "Information Required on the Subcooling Meter," concerning:

- The uncertainty of the input parameters (reactor coolant system (RCS) coolant temperature and pressure)
- The uncertainty in the digital readout meter
- The range of RCS pressures that the digital readout meter uncertainty is applicable

Safety Evaluation Summary

The uncertainty (input signals and digital readout) information was updated to reflect the calculations revised as part of the Extended Fuel Cycle project. These calculations used DCPP specific calibration data to model equipment performance for both 18- and 24-month fuel cycles. The digital readout meter uncertainty applicability limitation was added based on a 1986 Westinghouse commitment to the NRC on subcooled margin monitor (SMM) accuracy requirements.

The proposed changes to the FSAR Update Table 7.5-5 accuracy statements provide an envelope for the expected control room indication uncertainty. The proposed changes will provide realistic control room indication accuracy statements in compliance with the requirements of Regulatory Guide (RG)1.70, Revision 1. An evaluation of the control room indication accuracy has been performed as follows:

The text of the FSAR Update does not indicate that range and accuracy information included in Table 7.5-5 is based on an operational requirement of DCPP. However, the PG&E has evaluated the proposed changes with respect to the design requirements established in DCPP design basis documents. This evaluation identified no adverse impact on the design bases of DCPP.

4. Substitution of Alternate Fire Protection Features for Unqualified 3-Hour Fire Barrier
FSAR Update Appendices 9.5A and 9.5H (LBIE Log No. 96-006)

The qualification of the Pyrocrete and 3M material used as a 3-hour fire barrier was determined to be questionable. 10 CFR 50, Appendix R, Section III.G.2.b, allows crediting other means of meeting the required separation of redundant circuits credited for safe shutdown in the event of a fire. Specifically it allows

crediting a 20-foot horizontal separation with no intervening combustibles or fire hazards, coupled with installation of fire detectors and an automatic fire suppression system in the fire area. All of these alternate requirements are met for the auxiliary saltwater (ASW) pump and exhaust fan circuits in Fire Area IS-1/Fire Zone 30-A-5.. These include local smoke detectors, overhead sprinkler heads, and local heat-activated carbon dioxide suppression systems. To ensure the operability of these detectors and suppression systems that are now being credited to meet Appendix R requirements in lieu of the 3-hour fire barrier, they have been brought under formal administrative control by adding them to the appropriate equipment control guideline

Safety Evaluation Summary

Meeting these alternate requirements ensures that at least one of the redundant ASW trains in each unit will remain operable in the event of a fire at any location in the fire area. There is no decrease in safety because in the event of a fire, the NRC has determined that the alternative measures being instituted provide equivalent protection against a fire to the fire protection measure being replaced.

5. Post-LOCA Hydrogen Generation
FSAR Update Appendix 6.2C (LBIE Log No. 97-212)

This change reflects a revised analysis performed by Westinghouse using more representative, yet conservative, assumptions for hydrogen concentration and amounts of aluminum and zinc in the containment. Specifically, the post-LOCA hydrogen concentration was increased from 35 to 60 cc/kg to envelope an anticipated change in procedures to increase the maximum concentration from 40 to 50 cc/kg to account for measurement uncertainty. Also, the allowed aluminum content was reduced from 4,076 to 3,576 lbm and the allowed zinc increased from 44,305 to 48,884 lbm.

Safety Evaluation Summary

Appendix 6.2C of the FSAR Update describes post-LOCA hydrogen generation in the containment. There is a licensing commitment to operate the hydrogen recombiners in such a manner as to keep the hydrogen concentration below 3.5 percent by volume. The revised analysis performed by Westinghouse demonstrates that the hydrogen limit continues to be met. Placing this revised analysis in the FSAR Update is within the licensing basis and there is no unreviewed safety question.

6. Timing for Main Feedwater Pump Trip After MSLB
FSAR Update Appendix 6.2C (LBIE Log No. 97-220)

This change clarifies Appendix 6.2C by adding a sentence to the discussion regarding the time available for the main feedwater pump to trip following a main steam line break. The sentence added is, "The analysis uses a feedwater flow curve that indicates feedwater pump trip at 2 to 2.5 seconds, but evaluation in Reference 14 of the total conservatism in the feedwater flow curve shows that up to 5 seconds time to receive the isolation signal is acceptable." Reference 14 is a letter from Westinghouse to PG&E documenting that the analysis in WCAP-13908 bounds an isolation signal time as long as 5 seconds.

Safety Evaluation Summary

The added information is a clarification of an accident analysis assumption statement. The analysis assumption, is not changed or any of the accident results or conclusions. Therefore, no unreviewed safety question is involved.

7. Installation of 3-Hour Rated Fire Damper
FSAR Update Appendix 9.5A (LBIE Log No. 97-127)

The fire hazards analysis for Fire Areas S-7 and TB-12 in Fire Zone 23-C in FSAR Update Appendix 9.5A was revised to reflect the existence of a 3-hour rated fire damper in the heating, ventilation, and air conditioning (HVAC) ductwork connecting these two area. The previous description indicated that the duct was undampered. The 3-hour rated damper was installed in 1983, but the change was not reflected in the FSAR Update.

Safety Evaluation Summary

A safety evaluation was not written for the change at the time because Unit 2 had not yet received its operating license. The existence of the 3-hour barrier in the duct serves to prevent the spread of a fire in one area to another area, and hence serves to reduce the consequences of a fire.

8. Fire Barrier Descriptions
FSAR Update Appendix 9.5A (LBIE Log No. 98-092)

Errors in fire barrier descriptions have been corrected to reflect current plant configurations. No design changes, procedure changes, or calculation changes are associated with this FSAR Update change.

Safety Evaluation Summary

The changes are being made to conform the FSAR Update with the current plant configuration. No new fire hazards are being introduced and no designs, calculations, or procedures are being revised. The radiological consequences of a fire or accident are not affected and no physical changes to the plant are involved. Based on the evaluation, it is concluded that no unreviewed safety question is involved.

9. Fire Protection - Reactor Coolant Pump (RCP) Lube Oil Collection System
FSAR Update Appendix 9.5C (LBIE Log No. 97-097)

Appendix 9.5C of the FSAR Update was revised to refer to a deviation, rather than an exemption, from the requirements of Appendix R for the reactor coolant pump (RCP) oil collection system. The appendix was also revised to change the oil flashpoint temperature 480°F to 425°F.

Safety Evaluation Summary

The deviation from 10 CFR 50, Appendix R, involves use of a common lube oil collection tank in lieu of dedicated collection tanks for each reactor coolant pump (RCP). Changing the terminology from "exemption" to "deviation" is an administrative change. Reducing the flashpoint temperature to 425°F is evaluated in DCPP fire hazards Appendix R evaluation (FHARE) 115, where it is concluded that the lower flashpoint temperature does not affect the basis for the deviation or the safe shutdown analysis. Therefore, it is concluded that no unreviewed safety question is involved.

10. Fire Protection Program Administration
FSAR Update Appendix 9.5H (LBIE Log No. 97-138)

FSAR Update Appendix 9.5H was revised to update organization and responsibilities descriptions, update the list of procedures, delete operating and surveillance requirements that are verbatim to existing Equipment Control Guidelines (ECGs), and add a list of ECGs that are related to rire protection systems, 10 CFR 50, Appendix R, and ECG bases. This change was made to reflect current DCPP fire protection system controls.

Safety Evaluation Summary

The FSAR Update change was made to reflect current administration of the DCPP fire protection program. There was no effect on plant operation. No accident (fire or design basis accident) or equipment malfunction probabilities or consequences were affected.

11. Tornado Failure Analysis Table
FSAR Update Chapter 3, Table 3.3-3 (LBIE Log No. 97-175)

This FSAR Update change corrected the tornado failure analysis table, which stated that the automatic make-up valves would fail open on a loss of instrument air resulting from a tornado. The actual system response of the make-up valves is to fail closed on a loss of air.

Safety Evaluation Summary

This evaluation bases its conclusion on the fact that automatic makeup is not required for component cooling water (CCW) to continue operating during a postulated tornado because other system leakage is not required to be postulated during the tornado. The normal position of the make-up valves is closed. If make-up water were desired after a postulated tornado had damaged the instrument air lines, the make-up valve can be manually bypassed to provide make-up.

12. Reactor Coolant System
FSAR Update Chapter 5 (LBIE Log No. 97-016)

FSAR Update Section 5.4.3 was changed from, "The storage racks are then removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to reactor closure removal and refueling cavity flooding" to "The storage racks are then removed from the refueling cavity for maintenance and inspection prior to reactor closure removal and refueling cavity flooding." This change was made because closure studs are normally stored outside containment during refueling.

Safety Evaluation Summary

The purpose of Section 5.4.3 is to explain that closure studs are protected from exposure to borated refueling cavity water by removing them from the refueling cavity before the cavity is flooded with borated water. Changing the storage location does not increase the chances of exposure to borated refueling cavity water. Floor loading and missiles outside of containment, in general, have already been evaluated.

13. Electrical Power and Emergency Lighting FSAR Discrepancies
FSAR Update Chapters 8 and 9 (LBIE Log No. 97-019)

This is a general revision to the electrical sections of the FSAR Update. The general revision provides clarification of as-built configurations and consistency

with existing procedures. There are no physical changes associated with this revision.

Safety Evaluation Summary

There was an inconsistency in the voltage, frequency, diesel generator speed, and nomenclature used to describe the requirements when the diesel generator energizes the bus. These changes do not increase the probability of an accident, since they have been changed to be consistent with the Technical Specification, design basis, and accident analysis. The remaining items do not increase the probability of an accident because they are changes to systems and components that do not affect the operability of ESF loads or are editorial in nature.

The changes do not adversely change the operation or maintenance of any equipment that could result in the malfunction of any equipment important to safety. The changes do not result in any physical change or procedural change that could result in the malfunction of any equipment important to safety.

14. 12-kV Cable Spreading Room Transite Panels," Pyrocrete Enclosure Doors, and Fire Hazards Analysis
FHARE 17 and 55, FSAR Update Appendix 9.5A (LBIE Log No. 98-005)

Because the original safe shutdown analysis conservatively assumed a loss of offsite power concurrent with a fire, most of the pyrocrete barriers and one plaster barrier were installed in certain fire areas to protect circuits associated with diesel generators and diesel fuel oil pump operation to ensure that an onsite power source was available for safe shutdown. Based on the revised safe shutdown methodology in Calculation M-680, the safe shutdown analysis for the affected fire areas were reviewed to determine if the fire barrier will still need to protect emergency diesel generator and/or diesel fuel oil pump circuits. As a result of the review, it was determined that offsite power would be available for shutdown in the event of a fire in the affected areas. As a result of not crediting the pyrocrete barriers, two fire hazards Appendix R evaluations (FHAREs) associated with pyrocrete enclosures are no longer required. This change was previously discussed with the NRC on September 17, 1997, and it was determined that this change could be evaluated via the safety evaluation process and would not require prior NRC approval for changing the assumptions related to offsite power.

Safety Evaluation Summary

The changes made to the FSAR Update and revisions to FHAREs reflect the results of the safe shutdown analysis. These changes incorporate the corrective actions completed for NCR N0001887. Not crediting the fire-rated enclosures in

some fire areas complies with the requirements of Appendix R, Section III.G. for separation of circuits. These changes will not increase fire hazards, ignition sources, or operation of fire protection equipment and safety-related equipment. Based upon the above criteria and justification, an unreviewed safety question is not involved. Also a change to the DCPP Technical Specifications is not involved.

15. System Performance Evaluation
FSAR Update Section 6.2.3.3.5 (LBIE Log No. 97-122)

The changes made are removing an incorrect reference and corrections to text in Section 6.2.3.3.5 that describes values used to determine the unsprayed volume of "approximately 17 percent" inside containment. Although the value of the unsprayed volume has not changed, the input parameters used to determine the unsprayed volume have been corrected to account for occupied volume above the deck, credit for sprayed refueling cavity volume, and mixing above the spray ring headers. This is a document change only, there is no physical change to the plant and no affect on procedures, plant operations, or accident analysis.

Safety Evaluation Summary

This change is to the method of establishing one of the parameters used to evaluate the performance of a safety system to mitigate an accident. The corrections to the input parameters did not change the value of the unsprayed volume, and therefore the analysis of the iodine removal capability of the containment spray is unaffected. So the consequences of accidents remain unchanged.

16. Electrical Bus Configuration During Modes 5 and 6
FSAR Update FSAR Update Chapter 8, Section 8.3.1 (LBIE Log No. 97-176)

This section describes allowed 4-kV, 120-Vac and 125-Vdc bus configurations during Modes 5 and 6. The revision was to clarify the allowed configurations.

This change has been superseded by another change and its evaluation.

Safety Evaluation Summary

The proposed revision clarifies configurations allowed by existing procedures. As noted, this change has been superseded.

17. Low-Pressure Turbine Exhaust Hood Spray Operation
FSAR Update Section 10.2.1.3 (LBIE Log No. 97-161)

The operation description in the FSAR Update for low-pressure exhaust hood spray was changed to better reflect the recommendations and parameters included in the vendor manual, for proper spray operation during plant operation.

Safety Evaluation Summary

This change was performed for clarification, because the description and parameters included in the annunciator response procedure for the operation and parameters of the low pressure turbine exhaust spray did not match the description and parameters contained in the FSAR Update.

This change did not create or increase the frequency of an accident different from those previously evaluated in the FSAR Update. This system is not addressed in the Technical Specifications nor Equipment Control Guidelines. This system does not affect or have inputs to or from any safety related system, systems important to safety, or any protection system for the turbine and reactor.

18. Clarification of the Leak Rate Requirements for Main Steam System Isolation Valves
FSAR Update Section 10.3 (LBIE Log No. 97-064)

Local leak rate testing of valves that isolate main steam system containment penetrations is not required by 10 CFR 50, Appendix J, Section II.H, or by FSAR Update Table 6.2-39, "Containment Piping Penetrations and Valving." However, some wording in FSAR Update Section 10.3 could have been interpreted to imply that such testing is performed. This wording was revised to remove that potential for misinterpretation and make it clear that such testing is not required.

Safety Evaluation Summary

Such testing is not needed because the main steam system inside containment is a seismically analyzed closed system whose pressure boundary integrity is verified during the containment integrated leak rate test and is not damaged during a loss-of-coolant accident (LOCA), so that post-LOCA containment atmosphere cannot enter that system and escape from the containment via that system's penetrations. Following a steam generator tube rupture accident, the contribution to offsite doses through the subject leakage paths is negligible compared to that through the stuck open 10 percent atmospheric steam dump valve assumed in the FSAR Update accident analysis, and hence need not be considered to be a contributor to offsite dose for this accident. Since leakage through these flow paths is not a contributor to offsite dose, there is no effect on

the consequences of an accident as a result of not local leak rate testing them, as allowed by 10 CFR 50, Appendix J, and FSAR Update Table 6.2-39.

19. Increase Steam Generator Blowdown
FSAR Update Section 10.4.8 (LBIE Log No. 97-218)

An engineering evaluation shows that the blowdown to the steam generator blowdown tank (SGBD) can be increased during plant startup and other plant evolutions. The system is designed for 150 gpm continuous blowdown, but can be increased to 320 gpm during plant startups following plant shutdowns in excess of 72 hours.

Safety Evaluation Summary

The SGBD sysem components have been shown by calculation to be able to accommodate the increased blowdown flow. As SGBD piping is 6 inches or smaller, pipe breaks are in the category of minor secondary system pipe breaks. Even with the increased flow, the consequences of pipe failures are bounded by high-energy line breaks associated with either main steam or main feedwater line ruptures. The flow increase has also been shown to be acceptable from an erosion/corrosion standpoint. Therefore, it is concluded that the change does not involve an unreviewed safety question.

20. Gaseous Radwaste System Parameter Change
FSAR Update Section 11.3 and Table 11.3-1 (LBIE Log No. 96-024)

This FSAR Update revision made minor changes to the design and operating parameters of the gaseous radwaste system components described in the referenced table.

Safety Evaluation Summary

This change has no safety impact because there is no change to the equipment and these values were not used in any analysis or licensing basis. The gaseous radwaste system equipment involved with this change is not involved with the gaseous sadwaste accident analyzed in FSAR Update Chapter 15.

21. Mobile Radwaste Processing System
FSAR Update Section 11.5.4.4 (LBIE Log No. 98-074)

This change corrects the FSAR Update by eliminating reference to interlocks and control signals for the mobile radwaste processing system (MRPS), features that the MRPS does not have.

Safety Evaluation Summary

The MRPS is manually operated in accordance with procedures to prevent overflow and spills. Any spill would be contained in the bermed pad and sump for return to the radwaste system. Radwaste overflow accidents are not evaluated in the FSAR Update and there is no need for interlocks and automatic control signals. Therefore it is concluded that this change does not involve an unreviewed safety question.

22. Additional Operations Directors
FSAR Update Section 13.1.2.2.2.1 (LBIE Log No. 98-016)

The change involves appointment of an additional Operations director for periods of high workload.

Safety Evaluation Summary

Appointment of an additional director is an administrative change that does not affect the accidents evaluated in the FSAR Update. Therefore, it is concluded that this change does not involve an unreviewed safety question.

23. Technical Specifications and Equipment Control Guidelines
FSAR Update Section 16.1 (LBIE Log No. 97-101)

In letter DCL-95-222, dated October 4, 1995, PG&E submitted License Amendment Request (LAR) 97-07 that proposed to relocate several Technical Specifications (TSs) to Equipment Control Guidelines (ECGs) and to reference the ECGs in the FSAR Update. During review of LAR 97-05, the NRC staff requested PG&E to add the following wording to FSAR Update Section 16.1:

"ECGs containing relocated TSs are incorporated into the FSAR Update, by reference, in Table 16.1-1.

"For ECGs listed in Table 16.1-1, if the equipment cannot be returned to service as required by the ECG, then a review in accordance with 10 CFR 50.59 is required."

Safety Evaluation Summary

These changes place increased administrative controls on ECGs that are relocated from TSs, but do not change the ECG requirements themselves.

24. Equipment Control Guidelines
FSAR Update Section 16.1 (LBIE Log No. 98-061)

Nine Technical Specifications have been relocated to Equipment Control Guidelines (ECGs) and added to FSAR Update Table 16.1-1 pursuant to License Amendments 120 and 118 dated February 3, 1998.

Safety Evaluation Summary

Adding the nine ECGs to FSAR Update Table 16.1-1 is an administrative change that has no safety or licensing basis implications. Therefore, it is concluded that no unreviewed safety question is involved.

25. Delete Precipitation Gauge Tipping Bucket Accuracy
(LBIE Log No. 97-017)
FSAR Update Section 2.3.3.4 (LBIE Log No. 98-061)

FSAR Update Section 2.3.3.4 specified the accuracy of the precipitation guage tipping bucket. This accuracy specification is excessive detail and is not consistent with other FSAR Update sections. Design Change Package (DCP) J-49101 installed a new tipping bucket with similar but different accuracy specification. This FSAR Update change deletes all reference to tipping bucket accuracy.

Safety Evaluation Summary

Accuracy of the precipitation gauge tipping bucket does not increase the probability of occurrence of accidents. The tipping bucket accuracy is not used in modeling of any accident evaluated in the FSAR Update. The Emergency Plan is not affected by the tipping bucket accuracy. The accuracy specification in the FSAR Update does not affect other equipment important to safety. The accuracy of the tipping bucket is not used in calculating the margin of safety for any technical specification.

26. Wind and Tornado Loadings
FSAR Update Section 3.3 and Associated Tables (LBIE Log No. 97-096)

This change revises information in Section 3.3 of the FSAR Update concerning safety-related equipment potentially exposed to the effects of a tornado. The changes do not represent any physical changes to the plant or procedures. An evaluation consisting of plant walkdowns and an engineering analysis of the effects of tornado wind and/or tornado missiles was performed. The significant conclusions from this evaluation were incorporated into the FSAR Update.

Safety Evaluation Summary

The engineering evaluation and walkdowns identified safety-related equipment and component vulnerabilities to tornado and missile effects. No circumstances were found that violated the tornado licensing basis. Therefore, there is no unreviewed safety question associated with this change.

27. Wind and Tornado Loadings, (Change No. K-3.3(7))
FSAR Update Section 3.3, Table 3.3-2 (LBIE Log No. 97-073)

This change revises the discussion of the wind and tornado design of the major structures, given in FSAR Update Section 3.3, to address certain problems identified during the preparation of Design Criteria Memorandum (DCM) T-9, "Wind, Tornado, and Tsunami." The majority of the changes either correct typographical errors or are editorial. However, certain changes correct minor discrepancies in the text of the FSAR Update:

The discussion of wind loading on the turbine building in Section 3.3.1.2 was clarified to indicate that loads were developed in accordance with U.S. Navy Design Manual DM-2, not the Uniform Building Code, as was implied by the text.

Corrected a misleading statement in Section 3.3.2.1.1 to indicate that PG&E does not have a commitment to a specific design basis tornado wind speed. This statement was originally added during the preparation of the FSAR Update based on a quote from Supplement 7 to the Safety Evaluation Report, which reflected the NRC's attempt to develop a conservative estimate of the tornado wind speed for DCPP, not the wind speeds used by PG&E.

The discussion of atmospheric pressure drop values in Section 3.3.2.1.1 was expanded to include those applicable to both large and small structures.

The discussion of calculational methods used for the determination of tornado missile forces on structures in Section 3.3.2.2.2 was expanded to include those applicable to both concrete and steel structures.

Safety Evaluation Summary

These changes make the FSAR Update consistent with the information contained in the applicable DCM and the design calculations associated with wind and tornado loading. There are no physical modifications to the structures or any changes in the reported wind and tornado resistance of the structures. Therefore, the level of wind and tornado protection afforded to safety-related equipment is not changed. Hence, the ability to safely shut down the plant in the event of severe winds or tornadoes is not affected.

28. Supplemental Meteorological System
FSAR Update Section 3.3.2 (LBIE Log No. 98-077)

This change was made to describe an upgrade to the supplemental meteorological measurement system. The system was upgraded from a single-phase, non-personal computer (PC) based system to a phased array, PC-based (IBM clone) system.

Safety Evaluation Summary

The previous system had Doppler acoustic sounders installed at three locations. The new system is more accurate, effiicent, and reliable than the old system. It was determined that installing Doppler sounders at two locations would provide information equivalent to the three previous locations. Therefore, it was decided to drop the Los Osos site. As this system is only a backup for accident response assessments and is equivalent to the old system, no unreviewed safety questions exist.

29. Component Cooling Water (CCW) System Surge Tank
FSAR Update Section 3.3.2.3.2.2 (LBIE Log No. 97-068)

This change clarifies the FSAR Update description of the redundant CCW surge tank high level alarms to match actual plant configuration. Contrary to the previous description, there is only one high level alarm on the CCW surge tank.

Safety Evaluation Summary

The installed high level alarm on the surge tank is Class II and utilizes a switch and transmitter that are safety-related for pressure boundary integrity only. The high level alarm has no active safety function and is not required to mitigate an accident or prevent an off site dose release. Therefore redundant high level alarms are not required.

30. Installation of the Component Cooling Water (CCW) Surge Tank Pressurization System
FSAR Update Section 3.3.2.3.2.2 and Table 3.3-3 (LBIE Log No. 97-154)

This change was made to reflect reflect the installation of the component cooling water (CCW) surge tank pressurization system. This change also corrected a statement in the tornado failure analysis table, which incorrectly stated that a maximum of 5000 gal. would spill from the surge tank in the event of tornado damage. The actual volume is 8100 gal. Additionally, this change clarified where the discharged 8100 gal would be directed.

Safety Evaluation Summary

The physical implementation of the CCW surge tank pressurization systems was covered by the evaluations performed for Design Change Packages (DCPs) M-049284 and M-050284. This evaluation concludes that revising the FSAR Update to reflect previously evaluated approved system changes does not result in a unreviewed safety question. This evaluation also concludes that the extra volume of CCW discharged as a result of a tornado can be managed by the inside or outside drainage systems without impairing the safety functions of the CCW system or other systems.

31. Detailed Results of the Tornado Evaluation of Turbine Building
FSAR Update Section 3.3.2.3.2.8 (LBIE Log No. 97-018)

This change updates FSAR Update Section 3.3.2.3.2.8 to reflect the in-situ configuration of the tornado missile barriers in the turbine building for the emergency diesel generator air intakes. Previously the FSAR Update indicated that the original air intake louvers were immediately behind the external missile barriers, while, in reality, these louvers were removed during the installation of the missile barriers in 1975. A similar misstatement in Section 4.3.5.1.5 of Design Criteria Memorandum (DCM) T-9 is also corrected.

Safety Evaluation Summary

This change makes the FSAR Update and DCM consistent with the in-situ configuration. There are no physical modifications to the missile barriers or any changes in the reported wind and tornado resistance of the barriers. Therefore, the level of wind and tornado protection afforded to the safety related diesel generators is not changed. Hence, the ability to safely shut down the plant in the event of tornadoes is not affected.

32. High-Energy Line Break (HELB) Compartment Pressurization Time Response
FSAR Update Section 3.6 Figures (LBIE Log No. 98-098)

Some of the high-energy line break (HELB) compartment pressurization time response figures in FSAR Update Section 3.6 have been revised to correct deficiencies.

Safety Evaluation Summary

The figures being revised document the consequences of non-mechanistic HELBs. The post-HELB compartment pressures and temperatures are decreased for some compartments and increased for others, but do not exceed the environmental qualification values for equipment required to mitigate the

HELB. Therefore, it is concluded that the changes do not involve an unreviewed safety question.

33. Design of Containment Structure
FSAR Update Section 3.8 and Associated Figures (LBIE Log No. 96-020)

This change updates FSAR Update Section 3.8.1 and several of the associated figures to reflect the latest analyses of the containment shell and liner plate. Containment pressure and temperature transient curves, element forces, and stresses are revised. In addition, an error in the reference to the code applicable to the design of the liner plate is corrected.

Safety Evaluation Summary

This change makes the FSAR Update consistent with the current loadings and analysis results. There are no physical modifications to the containment structure. All loads and stress levels are still well within the acceptance criteria established in the FSAR Update. Therefore, the level of accident protection provided by the containment structure is not changed. Hence, containment integrity and the ability to safely shutdown the plant in the event of a design basis accident is not affected.

34. Lighting and Communication in Containment Personnel Hatches
FSAR Update Section 3.8.1.1.3.3 (LBIE Log No. 98-068)

This change deletes the word "emergency" from the sentence describing the lighting and communications systems in the containment personnel hatches. The lighting and communications systems in the personnel hatches are "normal" systems operating from external "normal" supplies.

Safety Evaluation Summary

A review of 10 CFR 50, Appendix R, other parts of the FSAR Update, NRC Safety Evaluation Reports, pertinent NRC corespondence, and Occupational Safety and Health Administration requirements revealed no requirements or commitments for emergency lighting and communications in the personnel hatches. Therefore, the change is within the licensing basis and no unreviewed safety question exists.

35. Mid-Loop Operation and Use of Steam Generator Nozzle Dams
FSAR Update Section 5.1.6.5 (LBIE Log No. 97-215)

A section was added to describe reactor coolant system mid-loop operation and use of steam generator nozzle dams. The addition is based on information

contained in Westinghouse Technical Report, "RCS Pressurization Analysis for Diablo Canyon Shutdown Scenarios," dated April 1997.

Safety Evaluation Summary

The safety issue evaluations contained in the LBIE reference documents conclude that mid-loop operation with core decay heat no greater than 15.3 MWt will not have an adverse effect on safe operation at DCPP. Since an operating charging pump or gravity feed of the reactor coolant system from the refueling water storage tank, if required due to loss of station power, can provide sufficient inventory to maintain core cooling, the consequences of a loss of residual heat removal cooling event are considered acceptable from the standpoint of safety to operating personnel and to the general public for both high and low decay heat situations. The material added to the FSAR Update is consistent with NRC requirements and PG&E commitments to the NRC. Therefore, no unreviewed safety question exists.

36. Clarification of Water Systems Used During Refueling
FSAR Update Section 5.5.6.2.2.4 (LBIE Log No. 97-124)

Discussions related to filling the reactor cavity during refueling operations were clarified in the FSAR Update to note that several systems other than the residual heat removal (RHR) system can be used for filling and that the RHR inlet isolation valves are not closed if there is fuel in the core.

Safety Evaluation Summary

The changes made are for clarification purposes only. Uncontrolled boron dilution and dilution during refueling are not affected by this change. Therefore, no unreviewed safety question is involved.

37. Changes of Generic Discussions of Valves to Make Them DCPP Specific
FSAR Update Section 6.2 and Table 6.3-1 (LBIE Log No. 96-013)

The generic discussion of emergency core cooling system (ECCS) and containment isolation valves in FSAR Update Sections 6.2 and 6.3 was not completely accurate for DCPP. Not all these valves are double packed and fitted with stem leakoffs to the equipment drain system. What constitutes a full set of packing is not defined at DCPP. Nor are packless valves always used where possible. Furthermore, at DCPP these valves are normally purposefully not backseated when opened, to help minimize stresses on the valves imposed by thermal transients. Finally, not all body-to-bonnet valve gaskets are asbestos since use of asbestos in the plant has been curtailed. The discussion of these considerations in the FSAR Update was revised to make it specific to DCPP.

Safety Evaluation Summary

These aspects of valve design and operation are all related to controlling their post-accident radioactive leakoff to be less than that assumed in the DCPP offsite dose analysis. At DCPP, assurance that post-accident radioactive leak rate to the environment would be less than that assumed in the offsite dose analysis is demonstrated by performance of surveillance test procedures that measure such leakage from the subject valves. In this manner, it is confirmed by test that the consequences of an accident would not be increased above that previously evaluated.

38. Insulation for CFCU Motor Leads and Connections
FSAR Update Section 6.2.2.3.3.2 (LBIE Log No. 98-055)

This change revises the FSAR Update with regard to the rating of the insulation for the containment fan cooler unit (CFCU) motor internal leads and terminal box-motor interconnections. The FSAR Update previously stated that the insulation rating met or exceeded the rating of the motor (2300 V). The insulation rating, however, only needs to meet or exceed the rating of the service voltage, which is 460 V. The FSAR Update was therefore changed to reflect this reduced voltage requirement by reference to a Westinghouse evaluation contained in a PG&E environmental qualification (EQ) file.

Safety Evaluation Summary

The CFCU motor internal leads and terminal box-motor interconnections only need to exceed the service conditions of the DCPP 480 V system (460 V +/- 10 percent at the motor terminals per Design Criteria Memorandum (DCM) S-64). The Westinghouse evaluation (WCAP-7829) is a part of EQ file 1H-05 and confirms that the use of 600 V cables are adequate to ensure the ability of the CFCUs to fulfill their post-accident function. Therefore, this change does not involve an unreviewed safety question.

39. Evaluation of Insulation Loss and Recirculation Sump Availability
FSAR Update Section 6.2.3.3.8 (LBIE Log No. 97-164)

The containment recirculation sump debris analysis was inconsistent with statements made in the FSAR Update. Specifically, the FSAR Update stated that all fiberglass insulation debris is assumed to be transported to the sump, and the sump screen is designed to continue functioning without impeding water flow when it is 95 percent blocked. A new insulation debris methodology was developed, based on NUREG/CR-2791, which assumes less than 100 percent of the insulation debris reaches the sump, and utilizes a better criterion (differential pressure drop across the outer screen) for assessing the impact of sump screen

blockage. Other conservative assumptions were made which made the new analysis, on the whole, more conservative than the previous analysis.

Safety Evaluation Summary

The containment recirculation sump debris analysis demonstrates that the screen remains operable with old and new methodologies under the worst-case loss-of-coolant accident conditions when both fiberglass and paint debris are deposited on the sump screen. This analysis shows that the sump screen and debris provide low enough flow resistance to ensure adequate flow to the residual heat removal (RHR) pumps, and to ensure that RHR pump net positive suction head is maintained during recirculation.

40. Changeover from Injection Mode to Recirculation After Loss of Primary Coolant
FSAR Update Section 6.3, Appendix 6.3A, and Associated Tables
(LBIE Log No. 97-024)

The FSAR Update was revised to reflect changed assumptions in residual heat removal pump and containment spray pump flows, and containment pressure during the changeover from injection to recirculation following a loss-of-coolant accident (LOCA). Included is a discussion of the single active failure during the changeover. The changes are due to a more accurate analysis using more conservative assumptions. There were no changes in the configuration of DCPP. These changes affected the time available for operator manual actions.

Safety Evaluation Summary

This safety evaluation defines the acceptance criterion for the evaluation of manual switchover to cold leg recirculation as the refueling water storage tank (RWST) volume margin. As stated in Supplement 9 to the Safety Evaluation Report (SSER), there is an implied margin of 32,500 gallons remaining in the RWST at the completion of the switchover to cold leg recirculation. Other design bases for switchover include maintaining sufficient net positive suction head (NPSH) for the emergency core cooling systems (ECCS) and completion of operator switchover actions in about 10 minutes.

The manner in which the ECCS and containment spray system are operated and sequenced during the changeover from the injection mode to the recirculation mode does not change the probability or consequences of any accident previously evaluated in the FSAR Update. Although the new analysis increased the flow rates for the residual heat removal (RHR) and containment spray (CS) pumps, NPSH is not affected and the total time available for operator switchover continues to be greater than 10 minutes. The analysis shows that there are no increases in probability or consequences of a malfunction of equipment because

there is no loss of NPSH which could interrupt continuous cooling flow to the core.

The revised analysis assumptions do not introduce a possible new malfunction of equipment because the flows remain within pump, piping, and instrument design bases. The changes do not introduce any new common mode failures. The safety evaluation specifically includes the analysis of a single failure of an RHR pump to trip automatically. The single failure had not been previously defined nor evaluated. The new switchover time available is still greater than the actual time to accomplish switchover, even under the worst case single failure assumption.

The margin of safety, as defined in the ECCS Technical Specifications (TS) Bases, 10 CFR 50.46, and 10 CFR 50, Appendix K, is not reduced. The available RWST inventory meets the safety analysis and the TS, even with the new, more accurate, instrument uncertainties. The RWST volume margin remaining at the completion of switchover is 37,450 gallons and is greater than the SSER 9 implied margin. Therefore, the margin of safety has not been reduced.

41. Pump Net Positive Suction Head
FSAR Update Section 6.3.2.14 and Table 6.3-11 (LBIE Log No. 96-017)

Revise FSAR Update Section 6.3.2.14 and Table 6.3-11 to capture the minimum sump water elevation static head above the sump flow elevation, as allowed by Regulatory Guide (RG) 1.1, and to increase the residual heat removal (RHR) pump maximum flow rate from 4,500 gpm to 4,900 gpm for the worst case assumption in the post loss-of-coolant accident (LOCA) alignment.

Safety Evaluation Summary

To increase RHR pump maximum flow rate from 4,500 gpm to 4,900 gpm, the required net positive suction head (NPSH) has been increased from 19.5 feet to 25 feet. By taking credit for the minimum water level above the sump floor, the available NPSH is 28 feet which is more than the required NPSHR of 25 feet. Increased RHR pump flow will not impact the post-LOCA emergency core cooling system (ECCS) performance because: (1) there will be more flow to the core to provide cooling, thus increasing conservatism, (2) it will not boost centrifugal charging pumps (CCPs) nor safety injection pumps (SIPs) beyond their runout limits since the boosted pressure from the RHR pump decreases as flow increases, (3) RHR pumps have been actually tested by the vendor to a flow beyond the 4,900 gpm limit, (4) the increased brake horsepower is within the motor capability, and (5) NPSH margin exists. Therefore, it is acceptable to increase the RHR pump maximum flow rate from 4,500 gpm to 4,900 gpm.

42. Chemistry Environment for Environmental Qualification
FSAR Update Section 6.3.2.4 (LBIE Log No. 98-100)

This change removes the detailed description of the chemistry environment used for environmental qualification in Section 6.3.2.4 and references Section 3.11, where the environmental program is described with references to pertinent documents that contain the details of the chemistry environment used in the testing.

Safety Evaluation Summary

This FSAR Update revision only changes the location of the information about the chemistry of the spray solution used for environmental testing. Therefore, no unreviewed safety question is involved.

43. Digital Feedwater Control System Steam Flow Arbitrator
FSAR Update Section 7.2.2.3.5 (LBIE Log No. 98-048)

This change adds a description of the digital feedwater control system (DFWCS) steam flow arbitrator signal validation function to FSAR Update Section 7.2.2.3.5 based on a Westinghouse safety evaluation. The change shows how the DFWCS meets IEEE Standard 279.

Safety Evaluation Summary

The FSAR Update only discussed the median signal selector function of the DFWCS. The steam flow arbitration function is design basis information that should have been included in the FSAR Update when the DFWCS design change was made. Based on the Westinghouse evaluation contained in NSAL 96-04, the addition of this information was determined to not involve an unreviewed safety question.

44. Addition of Automatic Start on Degraded 4.16-kV Vital Bus Voltage
FSAR Update Section 7.4.1.2.3 (LBIE Log No. 97-153)

The FSAR Update description of automatic diesel generator (DG) start was corrected to add automatic start on degraded 4.16-kV vital bus voltage as well as on loss of offsite power, loss of 4.16-kV vital bus voltage and safety injection (SI). In addition, the description of manual DG controls was corrected to remove the incorrect statement that manual controls for DG starting and control were provided at the vital switchgear. No such controls are located on the 4.16-kV vital switchgear.

Safety Evaluation Summary

The FSAR Update change was made to correct information in the FSAR Update so that the system design and operation were correctly described. No changes were made to the facility or system operation. There was no impact on any accident or equipment analysis evaluation.

45. Discrepancy Between FSAR Update and Calculation IH-100 Rev 10/Plant Information Management System (PIMS) CDB
FSAR Update Section 7.5 and Table 7.5-6 (LBIE Log No. 97-091)

FSAR Update Section 7.5 and Table 7.5-6 contained incorrect instrument ranges for the containment recirculation sump water level (narrow range) and Containment Pressure (wide range). This evaluation addressed changing the range for the narrow range containment sump water level from "88.5 to 97 ft" to "88.5 to 96.6 ft" and changing the range for the wide range containment pressure from "0 to 200 psig" to "-5 to 200 psig."

Safety Evaluation Summary

The range for the narrow range containment recirculation sump level was changed from "88.5 to 97 ft" to "88.5 to 96.6 ft." This range should have been changed to "88.5 to 96.6 ft" as part of Design Change Package (DCP) J-41715, which identified the need to change the range statements in the FSAR Update. However, in revising the FSAR Update, the range was rounded from 96.6 ft. to 97 ft.

The range for the wide range containment pressure was changed from "0 to 200 psig" to "-5 to 200 psig." This range change is in accordance with the original license commitment (see Supplement 14 to the Safety Evaluation Report) to have continuous indication of containment pressure over a range of -5 psig to three times the design pressure of containment for concrete or four times the design pressure for steel. The installed equipment has a range of -5 to 200 psig as was originally installed. Therefore, the FSAR Update is being revised to reflect the correct design of the wide range containment pressure.

46. Thermal and Hydroelectric Plants Underfrequency Setpoints
FSAR Update Section 8.2.2.2 (LBIE Log No. 98-110)

PG&E has revised the hydroelectric generating plant underfrequency setpoints based on the guidelines of the Western Systems Coordinating Council. A discussion of these guidelines and the broad capability of hydroelectric units to operate during underfrequency events has been added to the FSAR Update.

Safety Evaluation Summary

The purpose of the underfrequency setpoints is to provide security and protection of the interconnected bulk power network by arresting frequency decline during periods of insufficient generation. PG&E's load shedding program is intended to automatically shed about 50 percent of the load within its control area. The previously hydroelectric underfrequency setpoint was designed so that hydroelectric generation would be the last units on the grid to trip on underfrequency. This remains true for the revised setpoints. Therefore, changing the hydroelectric underfrequency setpoint does not affect accidents or equipment malfunctions evaluated in the FSAR Update. Based on these considerations, it was concluded that no unreviewed safety question is involved.

47. Diesel Generator Capability
FSAR Update Section 8.3 (LBIE Log No. 97-082)

This change was made to clarify transient voltage and frequency dip and recovery times during load sequencing to discuss previously established commitments with respect to emergency diesel generator (EDG) performance during load sequencing, to recognize that the KWS relays are not credited for performing a safety function, and to appliy Regulatory Guide (RG) 1.9, Rev. 2, "Regulatory Position C4," as it pertains to voltage and frequency dip and recovery during load sequencing,

Safety Evaluation Summary

DCPP meets the frequency and voltage dip requirements of RG 1.9, Revision 0, as demonstrated by analysis in DCPP Calculation 215-DC, Revision 2, for nominal load block time intervals. DCPP meets the frequency and voltage recovery requirements of RG 1.9, Revision 0, as demonstrated by analysis in Calculation 215-DC, Rev. 2 for nominal load block time intervals. DCPP meets commitments to the NRC in PG&E Letter DCL 85-132 for demonstrating that the objectives of RG 1.9 are met for worst case load block time intervals, as demonstrated by analysis in Calculation 215-DC, Revision 2, and preoperational testing.

48. Diesel Generator Frequency Dip and Recovery
FSAR Update Section 8.3.1 (LBIE Log No. 98-034)

FSAR Update Section 8.3.1.1.13, "Diesel Generator Units," has been revised to clarify transient voltage and frequency decrease and recovery times during load sequencing. Specifically, the revisions (1) apply the criteria of Regulatory Guide (RG) 1.9, Revision 2, "Regulatory Position C4," as it pertains to voltage and frequency dip and recovery during load sequencing, (2) documents previously established commitments to demonstrate emergency diesel generator (EDG)

performance during load sequencing with worst-case sequence timing intervals, and (3) explicitly states that the KWS relays are not credited for performing a safety function.

Safety Evaluation Summary

The evaluation performed shows that the EDGs (1) meet the frequency and voltage decrease requirements of RG 1.9, Revision 0, for nominal load block time intervals, (2) meet the frequency and voltage recovery requirements of RG 1.9, Revision 2, for nominal load block time intervals, and (3) meet commitments to the NRC for demonstrating that the objectives of RG 1.9 are met for worst-case load block time intervals. Additionally, analysis is referenced that demonstrates acceptable frequency recovery without crediting the KWS relays. Based on the results of the evaluation, it is concluded that no unreviewed safety question is involved.

49. Diesel Generator Starting Air Requirements
FSAR Update Section 8.3.1.1.13.2 (LBIE Log No. 97-053)

FSAR Update Section 8.3.1.1.13.2 is revised to clarify the statement "three (3) consecutive 15-second cranking cycles" as it relates to nominal sizing criteria for the air start receivers. This FSAR Update change revises the text to identify the "three (3) consecutive 15 second cranking cycles" as sizing criteria used by the vendor to size the air start receivers .

Safety Evaluation Summary

The emergency diesel generators (EDGs)) are not accident initiators. This change is not the result of any physical modification to the EDGs or related systems. The capability of the DEGs to perform their design function will not be adversely impacted by revision to the FSAR Update text to clarify the starting air receiver sizing criteria. The probability of occurrence of an accident, malfunction of equipment important to safety, radiological consequences of accidents evaluated, different type of accidents, or margin of safety previously evaluated in the FSAR Update will not be adversely impacted. Therefore, revising the FSAR Update text to clarify the EDG starting air receiver sizing criteria will not result in an unreviewed safety question.

50. FSAR Update Change Chapter 8.3.1.1.13.6
FSAR Update Section 8.3.1.1.13.6 (LBIE Log No. 97-162)

This section describes diesel generator engine trips. The change deletes the 4160-V bus differential from the list of trips.

Safety Evaluation Summary

The proposed revision modifies an existing section of the FSAR Update to correct an error. The 4160-V bus differential only trips the 4-kV breaker to the diesel generator. It does not trip the diesel generator. There is no safety significance to this change since there is already a failure that would disable the diesel from providing power to its load. The correction describes the as-built response to this failure.

51. Modes 5 and 6 Electrical Alignment
FSAR Update Section 8.3.1.4 (LBIE Log No. 97-057)

The change addressed the removal from FSAR Update Section 8.3.1.4, Independence of Redundant Systems [Class 1E Electrical Systems], of material describing contingency configurations for Mode 5 and 6 operation. This material was not incorrect, but it was an inappropriate level of detail, and it described sample configurations that may not be utilized.

Safety Evaluation Summary

There was no identified safety impact of removing the material. The described configurations are still allowed.

52. Deletion of Requirement for Cables Terminating on Separate Terminal Blocks
FSAR Update Section 8.3.1.4 (LBIE Log No. 98-003)

This FSAR Update change deletes the requirement that redundant cables terminate on separate terminal blocks. It properly characterizes that they "typically" are terminated on separate terminal bolts.

Safety Evaluation Summary

The design basis is that mutually redundant circuits be separated by 5 inches or a separation barrier. When mutually redundant circuits are terminated on the same terminal block, barriers are an acceptable means of providing separation and are provided up to the terminal block as required by Design Change Memorandum (DCM) T-19. The terminal block provides sufficient clearance and leakage distance to meet the requirements of a separation barrier. The barriers assure that the probability or consequence of an accident will not be increased.

53. Use of ICEA P-54-440 for Derating of Cable Installed in Cable Tray
FSAR Update Section 8.3.1.4.3 (LBIE Log No. 96-029)

The FSAR Update has been updated to reflect the use of Standard ICEA P-54-440, which is the industry recognized method for derating cables installed within cable tray. ICEA P-54-440 derates cable ampacity based upon percent fill.

Safety Evaluation Summary

Cable ampacity is not the source of an accident and does not impact the consequences of a malfunction of equipment. Cable derating is not related to creation of accidents or to the margin of safety. The probability of occurrence of a malfunction of equipment or the possibility of a malfunction of equipment important to safety will be decreased by the use of standard ICEA P-54-440.

54. Reactor Vessel Stud Detensioning
FSAR Update Section 9.1.4.2.1.13 (LBIE Log No. 98-014)

The FSAR Update is revised to eliminate reference to the number of reactor vessel stud tensioners required for tensioning and detensioning activities.

Safety Evaluation Summary

An engineering evaluation of reactor vessel stud tensioning and detensioning procedures has been performed and revised procedural guidance developed and implemented. As part of the revised procedure, detensioning may be accomplished with either six or three hydraulic tensioners, or with only two if one fails. The revised procedure has no effect on accidents analyzed in the FSAR Update. All ASME Code stress and fatigue limits will be met and there will be no effect on any other safety-related equipment. Therefore, it is concluded that the proposed change does not involve an unreviewed safety question.

55. Auxiliary Systems/Fuel Storage and Handling
FSAR Update Section 9.1.4.2.1.4 (LBIE Log No. 97-155)

The FSAR Update section was revised to remove a statement that the fuel handling building (FHB) crane is normally stored in the hot shop. The statement was not supported by any commitment or basis. The FHB crane is seismically qualified for its entire runway. Parking is not restricted. Removing the statement from the FSAR Update eliminated unnecessary movement of the FHB moveable seal walls, and allowed operations flexibility to store the crane to better support normal operations.

Safety Evaluation Summary

The only FHB crane accident evaluated in the FSAR Update is a load drop over irradiated fuel. The crane is seismically qualified for its entire runway. Travel over the spent fuel pool (SFP) is controlled by Technical Specification 3/4.9.7 and by plant procedures. The crane is not stored over the SFP. No new accidents are created by storing the crane in the FHB while not in use. Allowing the crane to remain in the FHB while not in use will not affect any accident analysis. Equipment required for safe shutdown is not affected because crane storage is limited to FHB areas that do not contain equipment important to safety.

56. Relief Valve Criteria
FSAR Update Section 9.2.2.2.9 (LBIE Log No. 97-052)

The change addressed was the removal from FSAR Update Section 9.2.2.2.9, component cooling water (CCW) system valves, of the paragraph which contains the following information: "The relief valve [RV-52] on the component cooling water piping downstream of the excess letdown heat exchanger is sized for a tube break in the heat exchanger .... The relief capacity of this valve is such that the design pressure of this portion of the CCWS will not be exceeded."

Existing DCPP calculation for this valve demonstrating capacity is an equilibrium calculation. Equilibrium assumptions like perfect mixing are not as conservative as a three-dimensional transient analysis would be. It would be difficult to perform an accurate two-phase, three-dimensional analysis, and it is believed that the results may indicate higher than design pressures.

Safety Evaluation Summary

The justification for removing the statement is that there is no need to design against a tube rupture in this heat exchanger. A tube rupture here is a non-credible event. These tube walls are relatively thick, the fracture analysis indicates that failures will be more likely to be axial cracks and double-ended rupture will not occur, and inservice time for these heat exchangers (HXs) is very low (typically these HXs are only used when mechanical troubles exist in the normal letdown HX path. Westinghouse discusses the non-credibility of a tube break in letter PGE-97-530. Not assuming a double ended rupture in these 5/8-in. tubes is consistent with NRC Standard Review Plan 3.6.1, Appendix B, which does not require assuming circumferential breaks in pipes of less than 1-in. diameter.

DCPP's excess letdown heat exchanger is similar to most Westinghouse designs and no other Westinghouse FSAR Update has been discovered with this design statement. It does not appear in RESAR-3 or the updated FSARs of

Turkey Point, Vogtle, Shearon Harris, Salem, Sequoia, or Commanche Peak. Therefore, removal of this statement makes the DCPP FSAR Update consistent with sister Westinghouse plants, and enables PG&E to avoid reliance on a calculation believed to be non-conservative.

Licensing material such as the Standard Review Plan and (Supplement to Safety Evaluation Reports (SSERs) were reviewed. There was no evidence discovered that would lead to the conclusion that the NRC relied upon this function of RV-52 to grant the DCPP Operating License.

57. <u>Delete Reference to Flow Switch 22</u>
FSAR Update Section 9.2.3.3 (LBIE Log No. 97-114)

All references to flow switches/alarms (FS-22) are deleted from Section 9.2.3.3. FS-22 is located in the main makeup water header from the raw water reservoirs into the fuel handling building/auxiliary building. It has been concluded that break flow would not be enough to actuate the FS-22. No fieldwork is planned - the switches are to remain installed but not maintained.

<u>Safety Evaluation Summary</u>

The probability of occurrence of an accident (flooding) does not increase by deleting the flow switches/alarms in Section 9.2.3.3. The auxiliary building sump's high level alarm is adequate to detect and mitigate flooding, therefore break detection and isolation can still be accomplished as before.

58. <u>Liquid Sampling System FSAR Update Discrepancies</u>
FSAR Update Section 9.3 and Table 9.3-2 (LBIE Log No. 97-072)

These changes are made to revise sample transit time, make an editorial change in the description of sample flow, and revise the number of sample heat exchanger and sample heat exchanger design information.

<u>Safety Evaluation Summary</u>

There is no safety impact because this change does not involve any physical change to the plant. The changes are being made to update the design data in the FSAR Update and to better reflect actual plant conditions. Therefore, there is no increase in probability or consequences of an accident and no reduction in margin of safety.

59. Testing of Floor and Equipment Drainage Systems
FSAR Update Section 9.3.3.4 (LBIE Log No. 98-099)

This change clarifies the monitoring and testing performed on the floor and equipment drainage system prior to and during plant operation. It states that the systems were tested and inspected prior to plant operation and are periodically monitored during plant operation.

Safety Evaluation Summary

The evaluation performed for this clarification concluded that no unreviewed safety question is involved.

60. Undampered Ventilation Duct Penetrations
FSAR Update Section 9.5.1 and Appendices 9.5A and 9.5B
(LBIE Log No. 97-139)

DCPP fire hazards Appendix R evaluation (FHARE) 33 evaluates the acceptability of having ventilation duct penetrations that do not have fire dampers and do meet the 3-hour rating definition for a fire barrier. Some of the undampered duct penetrations have been previously described to the NRC in the Unit 2 10 CFR 50, Appendix R, report. See Supplement 31 to the Safety Evaluation Report for approval of deviations.

FHARE 33 was revised to incorporate the current safe shutdown analysis (DCPP Calculation M-928) and to delete a ventilation duct penetration previously evaluated as undampered but where a fire damper was actually installed. With the addition of the sixth diesel generator (DG 2-3), the physical layout of the plant was changed along with the safe shutdown analysis due to new safe shutdown circuits. A new layout and fire area was added for DG 2-3. References and combustible loading description were revised to be consistent with the FSAR Update and the combustible loading calculation.

Safety Evaluation Summary

FHARE 33 evaluates an as-built condition against the effects of a postulated fire. No new fire hazards were introduced. Normal function of safety-related equipment was not affected. Fire protection features were not changed. Probability of a fire or of other accidents was not changed. The primary changes to FHARE 33 involved addition of a new fire area for the DG 2-3 room and incorporation of the safe shutdown analysis in Cal. M-928. The changes do not affect ability to achieve and maintain safe shutdown. FHARE 33 does not affect non-fire accidents evaluated in the FSAR Update.

61. Computer-Based Fire Alarm System
FSAR Update Section 9.5B (LBIE Log No. 98-029)

In response to NRC Open Item 275/87-27-02, DCPP committed to install a computer based fire alarm system capable of providing the features specified in National Fire Protection Association (NFPA) 72D. This new fire alarm system has back up power through an uninterruptible power supply or batteries that meets the requirements of NFPA 72D. As a result of this change, FSAR Update Section 9.5B is revised to clarify the sources of back up power for the fire detection and alarm panels

Safety Evaluation Summary

This change only clarifies the sources of back up power for the fire detection and alarm panels. There is no increased probability of an accident, no increased accident consequences, no increase in the probability of occurrence of malfunction of equipment and no increase in consequences due to equipment malfunction. No new type of accident is created and there is no reduction in the margin of safety as defined in the Technical Specifications.

62. Drainage to the Equipment Drain Tank and Auxiliary Building Sump
FSAR Update Sections 11.2.2.2 and 11.2.2.3 (LBIE Log No. 98-075)

This change corrects drainage inputs from equipment in the auxiliary building that are collected in the miscellaneous equipment drain tank and corrects sources of potentially contaminated auxiliary building floor drain wastes that are collected in the auxiliary building sump.

Safety Evaluation Summary

The changes to the FSAR Update involve the specific routing of wastes to closed and open drains to bring the document into conformance with the plant design. The total amount or processing of liquid waste is unaffected. None of the changes decrease confinement and most increase confinement. Potential accidents analyzed in the FSAR Update are unaffected. Therefore, it is concluded that no unreviewed safety question is involved.

63. General Reference to the DCPP Q-List
FSAR Update Sections 3.1 and 3.2 (LBIE Log No. 96-019)

The substance of these changes represents an enhancement of the presentation of the design bases and classification details for DCPP structures, systems, and components (SSCs). No physical or de facto changes were made to the plant; no analyses or analysis assumptions or inputs were revised; no SSC

classifications were changed; DCPP compliance with the general design criteria (GDC) remains unchanged; control of the DCPP design basis is maintained.

Safety Evaluation Summary

Previously-evaluated accidents and malfunctions probabilities and dose consequences are unaffected, no new or different types of such events are created or become credible, and no interface with the technical specifications or their bases results from these changes.

64. Factors of Safety for Structure Gaps and Raceway Flexibility
FSAR Update Sections 3.8.1.5.3 and 3.10.2.18 and Table 3.8-5B (LBIE Log No. 98-108)

The FSAR Update changes evaluated are necessary to account for the dimensions of existing gaps between interior and exterior surfaces of the containment structure determined during walkdowns. The specific changes involve revision of the FSAR Update to account for revised factor of safety calculations that consider relative seismic displacements. Also, the FSAR Update section on electric cable raceways is revised to show that the effects of differential displacements on raceways spanning between structures can be accommodated through either use of flexible joints or through the flexibility of the raceway and its supports.

Safety Evaluation Summary

The existing gaps in the containment annulus and between the containment structure and the auxiliary building have been compared with calculated seismic displacements at several elevations. Factors of safety against contact have been determined and found to be adequate. Evaluations have also been performed to ensure that electric raceways between structures can withstand structural shifts due to seismic effects without damage. Based on a detailed consideration of each of the 10 CFR 50.59 questions, it has been determined that the seismic gap issues do not involve an unreviewed safety question.

65. Reactor Coolant System FSAR Update Discrepancies
FSAR Update Sections 5.1, 5.2, 5.4, and 5.5 (LBIE Log No. 97-069)

The changes are made to clarify the content of the FSAR Update and better describe the system design and operation. Also, changes are made to correctly describe the actual inspection performed. No physical changes are being made to the plant due to this FSAR Update change. The changes are made to more precisely describe system and the inspection of various components. No commitment or required inspection is being removed.

Safety Evaluation Summary

There is no increase in probability or consequence of an accident and no reduction in margin of safety. Therefore, it is concluded that no unreviewed safety question is involved.

66. Residual Heat Removal System FSAR Update Discrepancies
FSAR Update Sections 5.2, 5.5, 5.6, Tables 5.2-10 and 5.2-22 (LBIE Log No. 97-070)

The changes are made to precisely describe the content of the FSAR Update, to reflect the actual setpoint of the low-pressure alarm (pressurizer relief valve interlock), and to remove some unnecessary information to reflect the as-installed condition.

Safety Evaluation Summary

No safety impact exists because this FSAR Update change notice does not involve any physical change to the plant. The changes are being made to more precisely describe system and the inspection of various component. No commitment or safety function of the affected equipment is being changed. Therefore, there is no increase in probability or consequences of an accident or any reduction in margin of safety.

67. Reactor Vessel Fluence Calculations
FSAR Update Sections 5.2.4.4.4 and 5.2.4.4.5 (LBIE Log No. 97-177)

FSAR Update Sections 5.2.4.4.4 and 5.2.4.4.5 are revised to reflect the current methodology used by industry for performing reactor vessel fluence calculations. The methodology change incorporates more modern and accurate methods, new neutron transport computer programs and cross section database, has been reviewed and approved by NRC (WCAP -14040-NP-A), and is consistent with Draft Regulatory Guide DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

Safety Evaluation Summary

This change to the methodology for calculating reactor vessel fluence has been benchmarked, and reviewed and approved by NRC (WCAP-14040-NP-A). While the updated fluence methodology could affect calculation of reactor pressure vessel (RPV) fracture toughness, inadequate fracture toughness leading to a postulated failure of the RPV is outside the plant design basis. Fracture toughness requirements are ensured through the federal regulations (10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50, Appendix G, in combination with monitoring programs required by 10 CFR 50, Appendix H). This change in

fluence methodology has no impact on the existing fracture toughness limits, or the margin prescribed to these limits (which account for uncertainties in vessel fluence measurements and methodology), or the methods for calculating RPV fracture toughness (ASME B&PV Code Sections III and XI, Appendix G, and 10 CFR 50, Appendix G).

68. Reactor Coolant System FSAR Update Discrepancies
FSAR Update Sections 5.5, 5.6, Tables 5.2-9, 5.2-16, and 5.5-16 (LBIE Log No. 97-071)

The changes are made to: (1) more precisely describe the reactor coolant system (RCS), (2) delete redundant information, and (3) reflect the actual installed equipment.

Safety Evaluation Summary

No safety impact because this FSAR Update change notice does not involve any physical modification to the plant and the design and/or function of the RCS has not been changed.

69. Valve Leakage Criteria
FSAR Update Sections 6.2 and 6.3, Tables 6.3-1 and 6.3-3 (LBIE Log No. 97-133)

The FSAR Update is updated to clarify that the valve leakage criteria listed in Section 6.2.4.2.2 and Table 6.3-1 are those used for initial valve purchase, and not used for maintenance/in-service testing, and that the specific valves requiring a specific leak rate are covered by Technical Specification required surveillance testing programs.

Table 6.3-3 was updated to reflect that valves may have corrosion resistant bolting in addition to the listed low alloy bolting.

Safety Evaluation Summary

This change to the FSAR Update is a clarification to the text/table which does not affect the operation of the facility. The affected emergency core cooling system (ECCS)/containment isolation valves cannot cause any evaluated accident. The consequences of an accident are not increased as the valves in question perform their safety function unchanged in any manner. All valves in which leakage is a safety requirement remain tested per Technical Specification requirements.

70. Sodium-Hydroxide Additive Tank (SAT) Low-Level Alarm Function
FSAR Update Sections 6.2.3.4.2.1 & 6.2.3.5.3 (LBIE Log No. 98-004)

The change to FSAR Update Section 6.2.3.5.3 is a change to an incorrect description of the function of the SAT low-level alarm described in the FSAR Update. The SAT level instrumentation is not safety- related and is set to alarm just above the minimum required Technical Specification level. Operator action is required to investigate the cause of the low level alarm and to increase SAT inventory to within its normal operating band. No operator action is required when the SAT inventory is depleted during a loss-of-coolant accident (LOCA), and therefore no alarm is provided for this function.

Safety Evaluation Summary

The low level alarm provides early warning to operators to ensure that minimum SAT inventory is available pursuant to Technical Specification requirements.

During an accident, when the SAT is depleted, it would have performed its safety function of providing pH control for iodine removal by the containment spray. Under certain LOCA scenarios, it is possible for the SAT to be depleted before emergency core cooling system (ECCS) injection phase is completed (refueling water storage tank (RWST) at low-low level). This would allow nitrogen from the SAT to be ingested into the containment spray system (CSS) pumps near the end of the ECCS injection phase until the CSS pumps are shut off. An evaluation was performed indicating that the amount of nitrogen ingestion does not affect the CSS pumps from performing their containment spray function and transferring the RWST contents into the containment. The CSS pumps are not required for accident mitigation after completion of the injection phase of ECCS and the CSS pumps are shut off.

71. Clarification of Load Rejection Capabilities
FSAR Update Sections 7.7.1.8 and 10.4.4.1 and Table 1.3-1 (LBIE Log No. 97-149)

Prior to this change, the context of the referenced FSAR Update sections stated that the reactor would not trip following a 100 percent load loss. The DCPP turbine bypass system was designed to accommodate a load rejection above 50 percent power without a reactor or turbine trip. However, due to the large number of systems that must operate precisely in a fully coordinated manner, a manual or automatic reactor trip may follow a large load rejection. In fact, there has been only one occasion where a full load rejection has not resulted in a reactor trip.

Safety Evaluation Summary

The FSAR Update change was made to better describe the plant response following a large load rejection event. It was not made because of any plant equipment modification or a change in the way the plant was operated. No new or different accidents or malfunctions were introduced. There was no change that could affect the cause of a load rejection event. Therefore, the probability of a load rejection event, either with or without a reactor trip, is not affected.

The FSAR Update change clarifies the response of the plant to most load rejection events. The plant was designed to accommodate a full load rejection event without a reactor or turbine trip, and this design was not changed. FSAR Update Section 15.2.7 evaluated full load rejection events both with and without reactor trip. Consequences of a load rejection event are not increased whether or not the reactor trips.

The change implies a potential decrease in load rejection events without reactor trip, and a corresponding increase in load rejection events with reactor trip. Such an increased reactor trip frequency could affect the plant's cyclic or transient design. Technical Specification Table 5.7-1 reactor trip system cyclic limits were not approached or exceeded when the current reactor trip rates were extrapolated for the remaining life of each DCPP unit. There is no effect on probability of an evaluated equipment malfunction.

72. Sale of MBPP and Establishment of the Independent System Operator (ISO)
FSAR Update Sections 8.1, 8.2 and 8.3 (LBIE Log No. 97-191)

Changes to FSAR Update Sections 8.1, 8.2 and 8.3 were made to reflect 230-kV system operation without the Morro Bay Power Plant (MBPP) and the addition of capacitor banks at DCPP. The changes also address the transfer of the 230-kV and 500-kV transmission system control from PG&E to the ISO.

This change also addresses the voltage improvements in the plant buses achieved through the installation of new startup transformers with automatic load tap changing feature. The new startup transformers are designed to maintain a preset voltage at the plant buses regardless of the 230-kV system voltage variations.

Safety Evaluation Summary

DCPP meets the design basis requirements for offsite power availability according to the commitments to 1971 general design criteria (GDC) 17, Regulatory Guides 1.6 and 1.32, and IEEE 308.

The offsite power remains reliable since the ISO is required to operate the grid in a reliable manner and maintain sufficient capacity and voltage to assure that safety loads are operable when powered from offsite power. The ISO has commited to comply with the requirements of Operating Instruction O-23. Operating Instruction O-23 provides minimum 230-kV voltage requirements to maintain DCPP operability with and without shunt capacitors and no MBPP generation. This change does not result in a change of operation, maintenance, physical change, or procedural change that would affect the probability or consequence of an accident.

73. <u>HVAC System Changes</u>
FSAR Update Sections 9.4 and 12.2 (LBIE Log No. 96-012)

Miscellaneous changes were made to the wording of the FSAR Update description and discussion of the DCPP heating, ventilating, and air conditioning (HVAC) systems.

<u>Safety Evaluation Summary</u>

A few of the changes were to more accurately reflect the actual configuration, function, or operation of HVAC systems, but have no affect on, or relation to, the safe operation of the plant or the system's ability to mitigate an accident or the probability of equipment malfunction. The nature of these wording changes to increase the accuracy of the FSAR Update is such that none of them has any effect on the probability of an accident occurring, on the consequences of an accident, on the probability or consequences of equipment malfunction, or on margin of safety.

74. <u>FHB Ventilation Flow and Control Room Heat Load</u>
FSAR Update Sections 9.4.4 and 9.4.5 (LBIE Log No. 98-094)

The changes involve updating the flow quantities of the fuel handling building (FHB) ventilation system and the control room (CR) heat load, as well as numerous editorial corrections.

<u>Safety Evaluation Summary</u>

The FHB ventilation system is designed to help mitigate a fuel handling accident in the FHB. The total as-built exhaust flows are the same for both units, but the branch flows are different. The Chapter 15 accident analysis is based on total flow, so branch flow has no impact on analysis results.

The design heat load for the control room was used to size the air conditioning equipment. Ample margin exists so that changes in calculated heat load, including accident conditions, can be accommodated. In both cases, it is

concluded that the updating of FHB ventilation flow and CR heat load do not involve an unreviewed safety question.

75. Clarification of Procedure Enhancement Documentation
FSAR Update Table 17.1-1 (LBIE Log No. 98-027)

A clarification is added to the exceptions related to quality assure program requirements to explain the administrative control used to provide procedure enhancements feedback to the procedure sponsor.

Safety Evaluation Summary

The change deals with clarification of an administrative control regarding tracking of procedural enhancements and has no bearing on accidents evaluated in the FSAR Update. Therefore, it is concluded that no unreviewed safety question is involved.

76. Clarifications to the List of Active Valves
FSAR Update Table 3.9-9, Rev. 11A (LBIE Log No. 97-147)

The list of active valves in FSAR Update Table 3.9-9 contains valves that have a design basis active safety function to support accident mitigation and achieve safe shutdown, and also contains valves that have a nonsafety-related licensing basis active function to support achieving cold shutdown following a Hosgri earthquake. The distinction between these two classifications of valves has been made in other design basis documents outside the FSAR Update, but has not been made in this FSAR Update table. In addition, as iterated in the Diablo Canyon Supplement 7 to the Safety Evaluation Report (SSER) and SSER 22, DCPP is a "hot shutdown" plant, meaning that following an accident, "safe shutdown" is considered to be Mode 3. However, DCPP is required to be capable of achieving cold shutdown following a Hosgri earthquake or 10 CFR 50,Appendix R, fire with no concurrent accident. Again, while this distinction between the design basis safe shutdown and licensing basis shutdown definitions has been made in other design and licensing basis documents, it is not readily apparent in the FSAR Update. To clarify these two distinctions in the FSAR Update, two notes were added to the list of active valves in Table 3.9-9 that explain the distinctions.

Safety Evaluation Summary

While this constitutes a change to the FSAR Update, thus requiring a 50.59 safety evaluation, it is for clarification only, and does not constitute a change to the DCPP design or licensing basis. There is no change in the Code or quality classification, quality assurance, maintenance, or surveillance testing of any of the listed valves resulting from adding this additional clarifying information. The

operability, dependability, and behavior of the valves is unchanged, and hence the consequences of an accident are unchanged.

77. CCW Train Separation Valve Designation
FSAR Update Table 3.9-9 (LBIE Log No. 97-063)

The component cooling water (CCW) system is normally operated with its two safety-related trains cross-connected. Since a passive failure of one of the trains is postulated after 24 hours after a loss-of-coolant accident (LOCA) has occurred, the system is designed to allow separation of the trains using manually operated valves. The valves used to accomplish this train separation are considered to be active valves and are listed in the FSAR Update Table 3.9-9 list of active valves. The configuration of the system is such that the trains could be separated in several ways, using different combinations of manual valves. Table 3.9-9 was revised to list the valves actually used by procedure to perform the train separation.

Safety Evaluation Summary

All the valves involved have the same Code classifications and level of maintenance and testing, and are equivalently capable of achieving the train separation. Hence there is no change in the potential consequences of an accident because there is no change in the ability to achieve CCW train separation using the equivalent set of valves indicated in the Emergency Operating Procedure when compared with using the set originally listed in the FSAR Update.

78. Reactor Coolant Pressure Boundary (RCPB) Leakage Detection Systems
FSAR Update Table 5.2-16 (LBIE Log No. 98-089)

This change revises Table 5.2-16 to correct typographical errors, clarify ranges of instruments, restate the way containment condensation liquid detectors respond, and updates the approximate time needed to detect a 1-gpm leak for each detector.

Safety Evaluation Summary

The proposed changes in FSAR Update Table 5.2-16 do not affect the function of the reactor coolant pressure boundary (RCPB) leakage detection systems, do not degrade the ability of the plant to detect a reactor coolant system (RCS) leak, and do not impact any assumptions made in evaluating the radiological consequences of accidents. There is no change in equipment reliability and no impact on fission product barriers. The changes simply bring the FSAR Update into agreement with the plant design licensed by the NRC and therefore do not create the possibility of an unreviewed safety question.

79. Administrative Control of Containment Isolation Valves 8823, 8824, 8843, and 8885A/B
FSAR Update Table 6.2-39 (LBIE Log No. 97-213)

This change updates Table 6.2-39 to show that the administratively controlled Containment Isolation Valves 8823, 8824, 8843, and 8885A/B do not have active control room position indication while they are administratively cleared, i.e., the control room indicating lights do not function.

Safety Evaluation Summary

The changes are FSAR Update clarifications that do not affect operation of the plant or the results of any accident analyses. Therefore, no unreviewed safety question exists.

80. Post-Accident Monitoring Indicators
FSAR Update Table 7.5-4 (LBIE Log No. 98-088)

This change corrects a typographical error and deletes "indicator" from the plant vent monitor as low as reasonably achievable (ALARA) since only a recorder exists for this variable.

Safety Evaluation Summary

The plant vent monitor ALARA is a Regulatory Guide 1.97, Category 3, variable for which only a recorder is needed. This change is a document change only and involves no physical work The change is within the licensing basis and does not involve an unreviewed safety question.

81. Emergency Diesel Generator (EDG) Stop Button Loads for Battery 13
FSAR Update Table 8.3-11 (LBIE Log No. 98-091)

The power supply to the emergency diesel generator (EDG) emergency stop buttons was deleted and the buttons were wired directly to the shutdown lockout relays. The buttons are therefore no longer a load for Battery 13 and the FSAR Update was revised accordingly.

Safety Evaluation Summary

Modification of the stop button circuit is a design improvement that enhances reliability. The licensing basis is unattended; hence, no unreviewed safety question is involved.

82. Control Room Indicator Accuracy and Range Statements
FSAR Update Tables 7.5-1 Through 7.5-5 (LBIE Log No. 97-038)

There were several instances of inconsistent information concerning control room indication accuracy and range statements in FSAR Update Tables 7.5-1 through 7.5-5 as compared to the design calculations, supporting documents and as-built design. These changes were reviewed by all affected DCPP departments.

Safety Evaluation Summary

Many of the accuracy statements in FSAR Update Tables 7.5-1 to 7.5-5 are more conservative than what has been computed in design and supporting calculations. The proposed changes to the FSAR Update are intended to report the current and the correct status of the plant. The collective coordination related to this FSAR Update change request with the appropriate departments and groups ensures the consideration of control room indication accuracy in the plant activities are consistent with the plant design parameters. Therefore, there is no increase in the probability or consequences and no reduction in margin of safety.

83. Boric Acid Heat Tracing and Tank Heater Loads
FSAR Update Tables 8.3-3, 8.3-5, 8.3-6, and 8.3-7 (LBIE Log No. 97-214)

FSAR Update tables were revised to account for a reduction of electric heat tracing and tank heater loads of the boric acid system. The heater loads were previously assumed to be operating at 100 percent rated capacity. Since the heaters are temperature controlled and operate intermittently, their loading was reduced to 50 percent.

Safety Evaluation Summary

Reducing the heat tracing and tank heater loads of the boric acid system, which is not safety related, makes the FSAR Update consistent with plant operation. The margin of safety is not reduced and it concluded that this change does not involve an unreviewed safety question.

G. Other Changes

1. <u>Reclassification of Containment Spray in the Recirculation Mode of Emergency Core Cooling to Nonsafety-Related</u>
(Units 1 & 2) (LBIE Log No. 97-206)

This evaluation justified the reclassification of the containment spray system to nonsafety-related during the recirculation mode of a loss-of-coolant accident (LOCA), thereby eliminating the requirement that the containment spray system be functional during recirculation.

<u>Safety Evaluation Summary</u>

The safety evaluation concluded that the reclassification is justified since containment spray is only required to be in service during the injection phase of an accident, and not during the recirculation phase of an accident.

Note, however, that during the NRC architect-engineer inspection conducted in August and September 1997, the inspection team indicated that PG&E's decision to declassify the containment spray function during the recirculation phase of a LOCA was a potential unreviewed safety question (USQ).

In order to resolve this disagreement regarding the USQ, PG&E has submitted LAR 98-03 to change Technical Specification 3/4..6.2.1, "Containment Spray System," to clarify that containment spray is not required to be actuated during recirculation, but may be actuated at the discretion of the Technical Support Center.

2. <u>Outage Safety Plan Schedule Change 1R8-05: Backseating of RCP 1-4 in Mode 5</u>
(Units 1 & 2) (LBIE Log No. 97-115)

Reactor coolant pump (RCP) 14 will be backseated to work on the seal package. The Outage Safety Plan and schedule requires the RCPs to be coupled when the reactor coolant system (RCS) is intact. The reason is that, upon loss of residual heat removal (RHR), the RCS will eventually pressurize, lfting the RCP off the backseat, which would result in a small cold leg opening. A cold leg opening would lead to an inventory loss, which could eventually lead to core uncovery.

<u>Safety Evaluation Summary</u>

The Outage Safety Plan is not described in the FSAR Update. Additionally, maintenance of the RCP seals is not covered in the FSAR Update. This condition/configuration will not affect the safe operation of the plant. A

contingency is provided to maintain the backseated condition of RCP 1-4 in order to maintain RCS integrity to the low temperature overpressure protection (LTOP) pressure-operated relief valve (PORV) lift point. This will allow for core coolng to occur per normal shutdown operation scenarios. This contingency consists of one 20-ton hydraulic jack between the pump and motor shafts. This contingency can be installed within a very short duration relative to the time to boil.

3. Use of RM-87 as an Alternate to RM-29 in Emergency Plan
(Units 1 & 2) (LBIE Log No. 96-039)

This licensing basis impact evaluation (LBIE) was prepared for Plant Staff Review Committee approval to formally allow RM-14/87 to be recognized and used as an alternative to RM-29 in the Emergency Plan for monitoring the plant vent effluent variable. The emergency action level classification chart in Emergency Procedure (EP) G-1 states that an unusual event must be declared if both the safety parameter display system and RM-29 lost all display capabilities. Since RM-14/87 has the same instrument classifications and covers a wider instrument range with better accuracy, it should be considered an alternate to RM-29, therefore, preventing unnecessary emergency declarations if RM-29 and SPDS were not available.

Safety Evaluation Summary

The LBIE covered a 50.59 review and a 10 CFR 50.54(q). Since no change to the facility or operation was proposed, and the RM-14/87 monitors met the requirements of the existing RM-29, the 50.59 did not reveal any unreviewed safety question issues. The 50.54(q) determined that the clarification in EP G-1 and Surveillance Test Procedure (STP) G-16 to also use RM-14/87 still meets the intent of NUREG-0654, 10 CFR 50.47(b)(4) and 10 CFR 50, Appendix E because it does not alter the emergency classification or condition but adds an additional method of performing the function.

4. Undampered Ventilation Opening in the Unit-2 Auxiliary Feedwater Pump Rooms
FHARE 10, Rev. 3 (Unit 2) (LBIE Log No. 98-071)

This fire hazards Appendix R evaluation (FHARE) revision addresses a previously unevaluated seismic support strut penetration through a ventilation damper between the two auxiliary feedwater pump rooms. This Appendix R fire barrier is rated as a 1-hour barrier. The subject penetration is sealed with an untested 4-in. thick configuration consisting of fire resistant materials of calcium silicate board and silicone foam sealant around the 3-1/2-in. diameter schedule 80 steel pipe strut.

Safety Evaluation Summary

The maximum equivalent fire duration in either fire area/zone on each side of the barrier is 20 minutes, a low fire severity; there are no combustible materials in the vicinity of the strut on either side of the barrier. Detection and automatic suppression features exist on both sides of the barrier. In the unlikely event of a fire affecting the fire area/zone on both sides of the barrier, the consequences would not be different from those evaluated and found acceptable in the FSAR Update.

5. Requirements for Non-Class IE Containment Overcurrent Protection
QE 10011535 (Units 1 & 2) (LBIE Log No. 97-098)

Supplementary Safety Evaluation 8 (SSER 8), dated November 15, 1978, required that primary and backup non-Class 1E penetration overcurrent protection be capable of remaining operable during an operating basis earthquake (OBE). The safety evaluation is to demonstrate that the existing documentation and analyses are sufficient to meet the intent of the SSER 8 requirements to protect the penetrations in the event of an OBE.

Safety Evaluation Summary

There is no licensing basis accident in Chapter 15 of the FSAR Update that postulates an accident during or after a seismic event. Seismic qualification of the reactor coolant pressure boundary precludes a loss-of-coolant accident occurring as a result of a seismic event. Redundant overcurrent protection assures that containment integrity is maintained and the single failure criterion is met during an accident. A failure modes and effects analysis demonstrates that there are no credible failure modes that would result in a failure to protect the penetrations for a fault inside containment after an OBE. The radiological consequences of analyzed events requiring containment integrity are not increased. There are no new accidents or increased consequences of malfunctions of equipment important to safety. There is no impact on the Technical Specifications or their Bases.

6. Unqualified Penetration Seals in the ASW Pump Room Barriers
FHARE 114 (LBIE Log No. 97-044)

Fire hazards Appendix R evaluation (FHARE) 114 evaluates the ability of the unqualified penetration seals in the auxiliary saltwater (ASW) pump room barriers to protect one train of safe shutdown circuits from the effects of a fire. FSAR Update Appendix 9.5A has been revised accordingly to address the non-rated seals.

Safety Evaluation Summary

The non-rated seals are installed in a configuration that is not supported by a qualified fire test. Therefore, a fire rating cannot be assigned to these configurations. In accordance with the fire hazards analysis in Appendix 9.5A of the FSAR Update, the ASW pump vaults are separated from other fire areas by a 3-hour-rated fire barrier. FHARE 114 concludes that the effectiveness of the barrier is not reduced by the use of the non-rated seals. Therefore, no unreviewed safety question is involved.

7. CCW Heat Exchanger Rooms - Fire Area Boundary
FHARE 120, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-042)

This fire hazards Appendix R evaluation (FHARE) evaluates the acceptability of combining Fire Areas 14-E (component cooling water (CCW) Heat Exchanger Room) and 14-A (Main Turbine Building) in Unit 1 and combining the equivalent Unit 2 areas 19-E and 19-A. This would result in the existing barrier between the two areas no longer being controlled as an Appendix R barrier.

Safety Evaluation Summary

A review of the safe shutdown capabilities associated with these fire areas shows that there are no safe shutdown features in the CCW Heat Exchanger Room that are redundant to those in the Main Turbine Building, and vice versa. The ability to achieve and maintain safe shutdown will not be affected by this combining of areas. By combining these two areas in each unit, there is no intention to change the existing combustible loading in either area (which is already low), nor to change the existing fire detection and suppression features in either area, nor to reduce the effectiveness of the existing barrier between the two present areas.

8. Pipe Penetration Seals Through Plaster Walls in the Unit 1 AFW Pump Rooms
FHARE 121, Rev. 0 (Unit 1) (LBIE Log No. 98-073)

Numerous (~45) penetrations exist in three of the Appendix R fire barriers through plaster walls of the auxiliary feedwater pump rooms. These penetrations are located in the 1- and 2-hour rated sections of the fire barriers. The details of these penetrations are very similar to a tested configuration; but, not being identical, they are considered as untested, requiring a fire hazards Appendix R evaluation (FHARE).

Safety Evaluation Summary

The combustible loading in the related fire areas results in a maximum equivalent fire duration of much less than 1 hour, the largest being 19 minutes,

The similarities of the design of these penetrations to the tested 3-hour configuration provides a level of protection easily commensurate with these low fire severities. In addition, detection and suppression equipment exists to respond to postulated fires. The FHARE evaluation of safe shutdown capabilities confirms that safe shutdown can be achieved and maintained in the event of a postulated fire.

9. Unsealed Penetrations With Fusible Link Chain Penetrants Through Fire Barriers
FHARE 123, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-039)

Small diameter (typically 3/4-in. or less), unsealed penetrations exist at various locations in DCPP, 3-hour rated, Appendix R fire barriers. These holes have fusible-linked chains passing through them associated with the fire-caused closure of roll-up fire doors. Free movement of the chain through the hole is required for the fire door actuation to occur. The wall thickness at the location of these penetrations is 10-in. minimum. They are typically high on a wall, near the top of the associated fire door.

Safety Evaluation Summary

These penetrations were provided to meet the requirements of NFPA-80, "Standard for Fire Doors and Fire Windows." For DCPP, Figure B-48 illustrates such installations. The small diameter of the hole and substantial thickness of the wall will limit the quantity of combustion products that will pass through the fire barrier while preventing flames from passing through the opening. This limited quantity of heated gasses is not expected to be great enough to raise the general area temperature or affect the operation of equipment in the unexposed compartment.

10. Unsealed Penetrations Through Barrier 119
FHARE 124, Rev. 0 (Unit 1) (LBIE Log No. 98-040)

Two unsealed penetrations exist in the CCW Pump 1-3 room floor, a 3-hour rated Appendix R fire barrier. The steel sleeves in the penetrations extend 4-in. above the floor level. The fire hazards Appendix R evaluation (FHARE) evaluates this condition between Fire Zone 3-J-3 above and fire Zone 3-C below.

Safety Evaluation Summary

Supplemental Safety Evaluation Report (SSER) 23 previously accepted the use of curbing around floor openings as contributing to the prevention of fire spread from flammable liquids. These two sleeved penetration configurations are consistent with the justification/reasoning given in SSER 23: the 4-in. high

sleeves will prevent the flow of combustible liquid (lube oil) from the CCW pump area to the zone below. The total combustible loading in both the upper and lower areas is low (i.e., equivalent fire severity of less than or equal to 15 minutes). Fire detection and suppression systems exist in both zones. There is no redundant safe shutdown equipment in the two fire zones associated with these penetrations. A fire originating in the lower area is no more likely to compromise safe shutdown functions in the CCW pump room above, especially since a fire in the lower zone can only subject one of the penetrations to a fire since there is a wall between the two penetrations below the floor.

11. <u>Lesser-Rated Plaster Blockouts and Penetration Seal Configurations</u>
FHARE 125, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-041)

The fire barrier between each Unit's turbine-driven auxiliary feedwater pump room and the liquid holdup tank (LHUT) room is an Appendix R, 3-hour rated barrier. An approximately 2½ by 5 foot opening exists in each of these barriers; the opening is sealed with a 2-hour rated gypsum plaster seal assembly.

<u>Safety Evaluation Summary</u>

The LHUT area has manual suppression equipment and a combustible loading with an equivalent fire duration of only ½ minute; each pump room has area-wide detection and suppression equipment and a fire loading with an equivalent fire duration of less than 20 minutes. In the unlikely event a fire breached the subject boundary, safe shut down would not be compromised since no redundant safe shutdown components exist in the adjacent areas.

12. <u>HVAC Ducts Through Modified Unrated Hatches</u>
FHARE 126 (Units 1 & 2) (LBIE Log No. 98-101)

The 3-hour rated, Appendix R floor of each Unit's Cable Spreading Room (CSR) contains an equipment hatchway which is closed by the use of 1-in. steel hatch covers. This is an unrated configuration that was approved as a deviation in Supplemental Safety Evaluation Report (SSER) 23 (pages 9-18, -19, -31 and -32). A portion of the hatchway is now occupied by three heating, ventilating, and air conditioning ducts which penetrate vertically through this area. The ducts contain rated fire dampers but the exteriors of the ducts, though enclosed with heavy gauge angle steel at the penetration, are not sealed using a rated configuration.

<u>Safety Evaluation Summary</u>

The replacement of one of the hatch cover sections with the fire-dampered ducting does not change the basic fire barrier configuration. The discussions, evaluations, and conclusions in the original SER are still valid for this revised

condition: the lower area is of low fire severity and contains smoke detection leading to an early manual response for suppression; though the upper area has a higher equivalent fire duration, it contains heat and smoke detection and an automatic CO_2 suppression system; smoke and hot gases from a fire in the upper area would rise away from the floor penetration preventing combustion products from moving into the area below. Though both fire areas contain safe shutdown circuits, shutdown capability remains intact should a fire in either area occur, due to redundant circuits and/or manual actions for mitigation.

13. Non-Rated Pipe Penetrations in Ceiling of Unit 1 Turbine-Driven Auxiliary Feedwater Pump Room
FHARE 128, Rev. 0 (Unit 1) (LBIE Log No. 98-049)

A 3-hour rated ceiling exists for the turbine-driven auxiliary feedwater pump room. The firewater pump room is located above this area. Firewater piping (8-in.) passes through 12-in. diameter sleeved penetrations in this ceiling at two locations. A pipe anchor constructed with 3/8-in. steel plate is built directly above each sleeved opening, completely sealing the top of each penetration.

Safety Evaluation Summary

Since the pipe anchor completely seals the penetration, smoke, hot gases, combustible liquids, and fire suppression water is prevented from passing through the barrier. The maximum equivalent fire duration for the two, adjacent areas is 19 minutes; the combustibles are not near to the penetrations. Each area has fire detection and suppression equipment. No safe shutdown redundancy exists between the two areas.

14. Duct Penetrations Through Common Walls Associated With Fire Zones 8-A, 8-D, 8-E, 8-F, 8-G, and 8-H
FHARE 129, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-035)

A return air duct within each unit's half of the control room passes through two Appendix R fire barriers without fire dampers installed within the plane of the barriers and without the required fire resistance of the ducting.

Safety Evaluation Summary

The subject, 16-gauge ducts are well-fitted to the wall penetrations with fire stops and are seismically supported. The fire hazards in the adjacent areas are limited (i.e., maximum equivalent fire duration of 50 minutes) such that the 3-hour rating is regarded as conservative for the existing hazards. Per the NFPA Fire Protection Handbook, ducting of this construction can be credited for fires of up to 1 hour equivalent duration. Detection and suppression equipment is located in or immediately adjacent to these control room fire areas. Since the

control room is continuously occupied, any fire originating in these areas will be quickly detected and suppressed by plant personnel.

15. Unique Blockout Penetration Seal Through Barrier Between the Unit 2 Turbine Building and Containment Penetration Area
FHARE 13, Rev. 3 (Unit 2) (LBIE Log No. 98-051)

Two, 28-in. diameter main steam lines pass through a 3-hour rated wall between the containment penetration area and the turbine building. Due to seismic supports on the piping and thermal movement of the piping, an alternate configuration for the penetration seal is provided. A combination of 34-in. diameter Pyrocreted steel sleeves with double flexible boot seals is provided around each line on the turbine building side of the barrier.

Safety Evaluation Summary

The combustibles in each area translate into an equivalent fire duration of much less than 45 minutes, giving a low fire severity. These combustibles are at least 35 feet away from the penetration on the turbine building side and at least 20 feet away from the penetration on the containment penetration room side. Both areas contain automatic, water suppression systems; a partial area smoke detection system above cable trays is provided in the containment penetration area. Systems required for safe shutdown either have adequate redundancy available or credit is being taken for manual operator actions. As a fire would be confined to one fire area, redundant safe shutdown equipment would remain available.

16. Inaccessible Jumbo Duct Penetrants
FHARE 130, Rev. 0 (Unit 1) (LBIE Log No. 98-084)

Three 4-in. by 4-in., steel tubes ("jumbo ducts"), welded side-by-side, penetrate the 36-in. thick concrete, 3-hour rated fire barrier between the cable spreading room (CSR) and the containment penetration room. Due to their partial physical inaccessibility on one side of the barrier, a penetration seal configuration cannot be assigned. Glass-like epoxy resin is used to seal the cables inside the ducts.

Safety Evaluation Summary

These seal assemblies are similar to adjacent assemblies that have been tested to verify their ability to withstand a 3-hour fire. In fact, in some respects, the subject assemblies are more conservatively configured than the tested assemblies. The maximum fire severity is on the CSR side of the barrier and has an equivalent fire duration of 44 minutes. Most of the combustible material in this area is electrical wiring insulation in cable trays; however, fire stops along the tray runs would be expected to limit the fire to a localized area. Fire

detection and suppression equipment exists in both fire areas. Redundant safe shutdown equipment will remain adequately protected as previously credited in the licensing basis.

17. Unrated HVAC Duct Penetrations
FHARE 136, Rev. 0 (Units 1 & 2) (LBIE Log No. 98-109)

Fire hazards Appendix R evaluation (FHARE) 136 evaluates the acceptability of having unsealed duct penetrations in rated fire barriers located in Fire Areas/Zones 13D, 12A, 13E, 12B, 24D, and 23A. FSAR Update Appendix 9.5A has been revised accordingly to address the unsealed penetrations.

Safety Evaluation Summary

The unsealed duct penetrations in the specific fire areas/zones have been evaluated and determined to not adversely impact the DCPP fire protection program. The combustible loading is low in the affected areas and the existing fire protection features are adequate. Therefore, it is concluded that no unreviewed safety question is involved.

18. Concrete Equipment Hatches
FHARE 14, Rev. 3 (Units 1 & 2) (LBIE Log No. 98-072)

Concrete hatches are installed in the plant to aid in equipment access. Some of these are located in 3-hour rated Appendix R fire barriers. The evaluation was originally written to evaluate the existence of up to 2-3/8-in. wide, unsealed gaps around the hatch perimeters. This revision added 4 hatches to those evaluated under this fire hazards Appendix R evaluation (FHARE). These hatches are at the top of each of the residual heat removal (RHR) pump/heat exchanger vaults. Additionally, for the hatches previously covered by the FHARE, revised equivalent fire severities and safe shutdown equipment discussions are provided.

Safety Evaluation Summary

The revisions to discussions related to the original hatches and those being added to this FHARE's scope do not change the reasoning or conclusions of the previous 50.59 evaluation: (1) the revised combustibles loading still results in low equivalent fire severities; and (2) the RHR pump and heat exchanger rooms have partial smoke detection and water spray suppression. Therefore, due to the automatic and manual fire protection features, lack of continuity of combustibles, tortuous path of travel for a fire on an upper level to propagate downward to affect redundant safe shutdown. The only redundancy is the H Bus circuits located on the 76 foot elevation in fire zone 10 (20) and the G Bus circuits on the 107 foot elevation in fire zone 12-B (23-B). The spatial

separation between these two elevations represents a tortuous path for products of combustion or fire propagation. Therefore, the capability to achieve safe shutdown is not affected.

19. <u>CCW-1-TCV-130 Replacement (HOT TAP)--DCP M-049319</u>
TES 2-001-N (Units 1 & 2) (LBIE Log No. 97-087)

This design changed replaced the existing TCV-130 with one having better control characteristic for the given system parameters. Changes in the cooling water flow to the let down heat exchanger caused letdown temperature fluctuations resulting in boron concentration/reactivity. The existing TCV-130 was too large to maintain fine control, operating in the lower 2 percent of its control range. To allow the replacement of the valve without draining header "C" it was necessary to use a HOT TAP to install a by-pass line and smaller TCV-130. A HOT TAP allows the installation of branch connections to existing pipe while the system is "LIVE" which was required because component cooling water (CCW) for the spent fuel pool can not be isolated, even during outages.

<u>Safety Evaluation Summary</u>

The CCW is a safety related system which during refueling outages is relied upon to cool the spent fuel pool. Per FSAR UpdateTable 9.2-7 there are 5 CCW system malfunctions and consequences. Of these, two where potentially effected by the use of a HOT TAP to install the by-pass line. CCW system leakage was eliminated because the HOT TAP machine and associated fittings, flanges and valves where rated for the design pressure and temperature of the system. As a precaution the HOT TAP machine was hydrostaticlly tested prior to breaching the CCW system. The second possible effect was CCW heat exchanger tube rupture which could allow RCS inleakage into the CCW. Due to the relatively small size of debris anticipated in the HOT TAPPING process and the configuration of the system it was determined to be highly unlikely that a tube rupture could occur. Even if a tube where to rupture this would not effect the primary safety function of the CCW to cool safety related loads during Modes 1-4. As a precaution the work was to be completed during Mode 0 (Reactor Defuelled). Completing the work during Mode 0, should a tube rupture, there would not have been inleakage of RCS into the CCW because RCS is depressurized during refueling. Based on the above conclusions the possibility of an accident of a different type than any previously evaluated in the FSAR Update was not created.

20. <u>Non-Rated Features in the Units 1 and 2 Centrifugal Charging Pump Rooms</u>
FHARE 25, Rev. 3 (Units 1 & 2) (LBIE Log No. 98-080)

An additional, triangular-shaped, nonrated penetration of approximately 6-in. by 12-in. is included in this Fire Hazards Appendix R Evaluation (FHARE). It is

immediately adjacent to one of the undampered duct penetrations evaluated in an earlier revision of this same FHARE. One side of the penetration is completely covered by a section of structural steel angle. In addition, the combustible loading description for all the related fire areas was updated and a safe shutdown capability analysis was provided.

Safety Evaluation Summary

Using the same reasoning as that provided for the originally-evaluated adjacent duct penetrations, the additional penetration does not create an unreviewed safety question: there is low equivalent fire duration in the related fire zones, automatic fire detection and wet pipe sprinklers are present, a minimum spatial separation between redundant safe shutdown components of 30 feet exists, and there is an absence of significant quantities of combustibles near the penetrations in Fire Zone 3-C.

21. Undampered Duct Penetrations in Fire Areas/Zones 4-B, 19-E, and TB-7/19-A
FHARE 58, Rev. 3 (Unit 2) (LBIE Log No. 98-117)

Fire hazards Appendix R evaluation (FHARE) 58 was revised to acknowledge the absence of a penetration seal around the ductwork that was previously evaluated for not having fire dampers at the Appendix R fire barriers. Combustible loading and the resulting equivalent fire durations were also revised for the related areas, all of them still remaining in the low fire severity category. A more detailed description of fire protection features in these areas was also provided as well as a safe shutdown capability analysis.

Safety Evaluation Summary

The same fundamental arguments for the original conclusion of no unreviewed safety question still apply: combustibles loadings having low fire severities, automatic smoke detection and sprinkler systems, manual fire fighting capabilities, and the ability to achieve safe shutdown even in the unlikely event a fire was to breach the subject boundary.

22. Lead Shielding Request Per Procedure RP1.ID2
TSR 97-011, Rev. 0 (Unit 1) (LBIE Log No. 97-059)

To allow the plant to install and remove temporary lead shielding in Unit 1 containment for Lines 508, 509, 927, and temporary steel attached to containment annulus structure, located above the residual heat removal (RHR) sump in containment. Shielding will be installed in Modes 5 and 6 only. Shielding will be removed prior to entering Mode 4. Shielding will be installed on operable piping which creates a condition that might affect safe operation of the plant not evaluated in the FSAR Update.

Safety Evaluation Summary

Seismic calculations demonstrate that the components are not adversely affected. The tie-down arrangement of the lead blankets will be such that it will not create a new source, per seismically induced systems interaction, or affect any other safety-related systems, structures or components. The only potential concern would be for the subject residual heat removal lines to suffer a medium energy line break, near the shielding location. In this case during Mode 5 operation, the residual heat removal sump is not required to mitigate the line rupture. The impact of the shielding on the sump could render the sump inoperable, but it would not affect the safe shutdown of the plant. The DCPP accident analysis does not postulate a Mode 5 loss-of-coolant accident or line break. Based on the above criteria and justification, an unreviewed safety question is not involved.

23. Mode 4 to Mode 3, Obtain New RVLIS DP3 coefficients
AR A0425503 (Unit 1) (LBIE Log No. 97-095)

This LBIE was a part of Attachment 9.10 of Operations Procedure OP L-0 that obtained Plant Staff Review Committee approval for Unit 1 to transition from Mode 4 to Mode 3 with Train A of reactor vessel level indication system (RVLIS) out-of service. RVLIS is required per Technical Specification (TS) 3.3.3.6 for Modes 1-3.

Train A of RVLIS was required to be out-of service to perform the data collection required to obtain new DP3 coefficients while the plant heats up from refueling to Mode 3, normal operating pressure/normal operating temperature conditions.

The action to reperform the DP3 curve was corrective action from NCR N0002016.

Safety Evaluation Summary

The LBIE screen for this condition screened "Yes" as a change to the system operation as described in the FSAR Update. The safety evaluation determined that an unreviewed safety question is not involved based on the Technical Specifications allowing the plant to transition modes while under Action (A) for TS 3.3.3.6 for RVLIS (ie TS 3.0.4 exempt). With Train B fully in service during this evolution and the system exempt from the provisions of TS 3.0.4, the facility design or license was not impacted.

24. Steam Generator Tube Support Plate Thin or Missing Ligaments
AR A0432415, Rev. 0 (Units 1 & 2) (LBIE Log No. 97-099)

Thin or missing steam generator (SG) tube suport plate (TSP) ligaments were identified by review of DCPP Units 1 and 2 SG eddy current data and confirmed by visual inspections of DCPP Unit 1 SG TSPs conducted in 1R8.

Safety Evaluation Summary

Operation of the DCPP Units 1 and 2 SGs with thin or missing TSP ligaments will not adversely affect SG tube structural and leakage integrity during normal operation and accident conditions. An active tube wear mechanism is not occurring at locations of thin or missing TSP ligaments, and no additional tubes are expected to experience deformation during a postulated loss-of-coolant accident plus seismic event.

25. Continued Operation With Cable Dampers for Steam Generator U-Bends
FTI Document 51-1264525, Rev. 1 (Units 1 & 2) (LBIE Log No. 98-023)

The subject of this licensing basis impact evaluation (LBIE) was the qualification of steam generator (SG) U-bend dampers designed by Westinghouse and installed in eight potentially susceptible tubes (November 1988 in the Unit 2 second refueling outage and October 1989 in the Unit 1 third refueling outage) to increase margins against flow-induced vibration in response to NRC Bulletin 88-02. This LBIE extended the qualification of the damper to full lifetime without the need for inspection based on further damper testing and the supporting 50.59 evaluation performed by Framatome Technologies.

Safety Evaluation Summary

Tubes that were susceptible to flow-induced fatigue cracking (WCAP 12064) have been plugged and dampened by installation of Westinghouse cable dampers to meet the requirements of NRC Bulletin 88-02. Therefore, in the dampened tubes, the tube plugs act as the reactor coolant system (RCS) pressure boundary, and the damper will not affect the function of the plugs. The increased wear in the damper/tube system has been evaluated through testing and analysis and has been determined to not affect the dynamic characteristics of the system or result in failure of the system. The relevant accident that has been previously evaluated in the FSAR Update is a SG tube rupture (SGTR). The probability of occurrence of an SGTR event is not increased because the dampers are qualified for continued performance of their safety function. Based upon the above criteria and justification, an unreviewed safety question is not involved. Also, a change to the Technical Specifications is not involved.

26. Determination of the Uncovered Screen Area and Pressure Drop Across the Screens in the Recirculation Sump After Paint and Insulation Severs From Objects Inside Containment During Post-LOCA Environment
Calculation M-591, Rev. 11 (Units 1 & 2) (LBIE Log No. 97-100)

Calculation M-591 was inconsistent with insulation assumptions described in the FSAR Update. It was shown that the sump analysis did not conflict or invalidate FSAR Update statements regarding the acceptability of paint chips plus transport of 100 percent of damaged insulation during a large break loss-of-coolant accident (LOCA). Instead, this calculation revision identified that a more limiting scenario exists, shredded insulation during a small break LOCA, and evaluated that scenario with new assumptions about insulation transport. Two key changes to the evaluation methodology were made: containment flood levels were determined for small break LOCAs, and less than 100 percent of fiberglass insulation in the form of shredded debris was assumed to reach the sump and deposit evenly over the screen. The impact of paint chip and fiberglass insulation debris were thus evaluated together for the first time.

Safety Evaluation Summary

LOCAs are the only accidents evaluated in the FSAR Update that rely upon sump operability. The function of the sump is to screen out debris while providing sufficiently low flow resistance such that the residual heat removal (RHR) pumps will not draw down the sump level and cavitate. The FSAR Update statement that transport of 100 percent of loosened fiberglass insulation concurrent with transport of degraded paint particles would not prevent the sump from being operable only applied to large break LOCAs with the assumption that the insulation remained intact. Calculation M-591 now credits less than 100 percent of insulation transport due to obstacles, the high specific gravity of fiberglass, and the low flow velocities through the containment during recirculation. A more limiting scenario was identified for screen operability, a small break LOCA with shredded fiberglass insulation spread out over the entire screen, and M-591 Revision 11 demonstrated that the sump remains operable under this new scenario.

27. Fiberglass Insulation Debris From HELB Inside Containment
Calculation N-042, Rev. 1 and N-051, Rev. 1 (Units 1 & 2) (LBIE Log No. 97-093)

Calculations N-042 and N-051 were revised to incorporate the latest design basis methodology for determining the quantity of fiberglass insulation debris generated due to various high energy line breaks (HELBs) inside containment. The leak-before-break (LBB) methodology was incorporated into the calculations to eliminate the dynamic effects of reactor coolant loop HELBs on fiberglass insulation, and certain non-terminal end breaks were eliminated from

consideration based on the latest jet impingement study and pipe whip study. Although use of the LBB methodology was approved by the NRC for DCPP in March 1993, Revision 11A of the FSAR Update did not reflect this revised licensing basis, so this safety evaluation was required to take credit for LBB in the insulation calculations.

Safety Evaluation Summary

General Design Criterion 4 states that structures, systems and components shall be appropriately protected against the dynamic effects, including missiles, pipe whip, and discharging fluids that may result from equipment failures outside the nuclear power unit. However, the dynamic effects associated with postulated pipe ruptures in the nuclear power unit may be excluded from design basis when the analysis reviewed and approved by the NRC demonstrates that the probability of a fluid system rupture is extremely low under conditions consistent with the design basis for the piping. The NRC determined that class 1 piping breaks at DCPP are sufficiently low that the dynamic effects associated with postulated primary pipe breaks need not be a design basis. The NRC approved DCPP's LBB evaluation in March 1993. Thus, elimination of the dynamic effects of postulated RCS loop piping ruptures from fiberglass insulation debris evaluations is within the licensing basis of the plant.

28. LHUT Dose Reanalysis/Calculation N-160
Calculation N-160, (Units 1 & 2) (LBIE Log No. 97-092)

FSAR Update Section 15.5, Liquid Holdup Tank (LHUT) Rupture, was reanalyzed to conform with Regulatory Guide (RG) 1.29, "Seismic Design Class." RG 1.29 specifies that some systems must meet seismic qualification criteria or the design basis accident offsite dose consequences must be less than 0.5 rem whole body. The LHUTs and associated piping were purchased seismic qualified but not maintained seismic qualified. The dose consequences from the original LHUT rupture was 1.44 rem. The reanalysis results are 0.152 rem. Therefore RG 1.29 is met.

Safety Evaluation Summary

The reanalysis of the postulated offsite dose from a LHUT rupture does not involve any changes to plant systems, structures or components. The reanalysis is based on conservative assumptions with respect to the original analysis contained in the FSAR Update. The reanalysis results show a reduction in offsite dose rates from the postulated LHUT rupture. Thus, the consequences of the LHUT rupture previously evaluated in the FSAR Update are reduced, not increased.

29. Control Room Vent
AR PK15-06, Rev. 8 (Unit 2) (LBIE Log No. 97-144)

This procedure revision revised the response to losing subtrains of control room ventilation system (CRVS) cooling, as described in the FSAR Update. Previously, there was no guidance provided for this event, since there are four equally redundant subtrains available to perform CRVS functions. However, to prevent the possibility of overlooking the FSAR Update-described response, these actions were added to the procedure.

Safety Evaluation Summary

Since the proposed procedure revision adds the detail as described in the FSAR Update, there is no change to the facility or operation as described in the FSAR Update. The evaluation was performed since the FSAR Update describes response to loss of all CRVS in moderate detail. An unreviewed safety question is not involved.

30. Revision of the Bases for Technical Specification 3/4.9.2, "Refueling Operations - Instrumentation"
Technical Specification Bases 3/4.9.2 (Units 1 & 2) (LBIE Log No. 97-219)

The revision of the Bases for Technical Specification (TS) 3/4.9.2, "Refueling Operations - Instrumentation" allows use of an alternate source range (SR) monitor during Mode 6 (refueling) in the event one of the two normal SR channels becomes inoperable. (Note: This has already been reported to the NRC in Letter DCL-97-035, dated March 18, 1997)

Safety Evaluation Summary

The use of alternate SR indication provided by post-accident neutron flux monitors is equivalent to use of a portable detector allowed by TS Bases per License Amendments 46 and 45. This condition does not involve an unreviewed safety question. PG&E believes there is reasonable assurance that the health and safety of the public will not be adversely affected by this TS Bases revision.

31. Revision of Technical Specification Bases 3/4.7.3 and 3/4.7.12 - Change Component Cooling Water System Design Basis Temperature
Technical Specification Bases 3/4.7.3 and 3/4.7.12 (Units 1 & 2) (LBIE Log No. 97-221)

The change to Technical Specification (TS) Bases 3/4.7.3, "Vital Component Cooling Water," and 3/4.7.12, "Ultimate Heat Sink," increased the maximum temperature at which the component cooling water (CCW) system may operate after a design basis event from 132°F for 120 minutes to 140°F for six hours

after the event, returning to a maximum of 120°F thereafter. This revision reflects upgraded qualifications of CCW components and equipment cooled by CCW to show that they can function with the higher temperature cooling water. An administrative change to relocate the temperature limit from Bases 3/4.7.12 to Bases 3/4.7.3 was also made to improve consistency.

Safety Evaluation Summary

PG&E has performed a detailed, component level review of the CCW equipment, support system equipment, and the equipment cooled by CCW. Each device was reviewed using vendor information as needed, and found to function properly with the increased cooling water temperature. The change did not require modification to any equipment or system, other than a minor adjustment to CCW heat exchanger auxiliary saltwater (ASW) discharge throttle valves to mitigate possible cavitation and resetting a post-accident sampling system (PASS) temperature switch setpoint to accommodate the new temperature limit. These valves are normally throttled already, and the required minimum valve position imposed by this change is within the range of the current normal operation of these valves. The imposition of these valve position requirements serves to minimize potential cavitation effects and has no detrimental impact on the capability of the ASW to perform its normal and emergency functions. Based on detailed evaluations of all affected systems, components, and structures, it has been demonstrated that they will perform their intended safety functions with the increased CCW water temperature conditions.

32. Evaluation of Zinc Addition in Cycle-9 at Diablo Canyon Unit 1
Westinghouse Letter SECL-97-207 (Unit 1) (LBIE Log No. 98-038)

The addition of zinc to the reactor coolant system (RCS) will be done for the purpose of decreasing the incidence of primary water stress corrosion cracking in the steam generator U-tubes. Zinc acetate will be injected via the chemical and volume control system (CVCS) system to achieve an RCS zinc concentration of 35-40 ppb during a nine month trial period. A secondary benefit will be the reduction.of radiation fields in the RCS. Tests will be performed to determine its effectiveness.

Safety Evaluation Summary

The injection of zinc into the RCS has not previously been done at DCPP nor is it described in the FSAR Update. Analysis of zinc injection performed at Farley 2 demonstrated that zinc did not have a deleterious effect on the function or operation of any RCS components with the potential exception being fuel. A root cause evaluation concluded that zinc may have a small detrimental effect on fuel cladding oxidation. For this reason, a conservative penalty was included in