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PG&E Letter DCL-00-068

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
10 CFR 50.59 Report of Facility Changes, Tests, and Experiments for the Report Period March 29, 1998 through December 31, 1999

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.59, "Changes, Tests, and Experiments," enclosed is the safety evaluation report for Diablo Canyon Power Plant (DCPP) Units 1 and 2. The report includes safety evaluations, which are part of PG&E licensing basis impact evaluations (LBIEs), prepared during the interval March 28, 1998, through December 31, 1999, unless previously reported in PG&E Letter, DCL-98-136, dated September 28, 1998.

The enclosure includes a summary of all 10 CFR 50.59 evaluations performed as part of the LBIE process for DCPP. The summaries are excerpts from the approved evaluations.

Since the LBIE process is used to perform impact reviews for compliance with regulations in addition to 10 CFR 50.59, some LBIEs did not include a 10 CFR 50.59 evaluation and, therefore, are not included in this report. Additionally, some LBIE numbers were canceled because additional reviews concluded that a LBIE was not required for the associated change.

The Plant Staff Review Committee has reviewed the referenced LBIEs and determined that the changes do not involve unreviewed safety questions or require changes to the DCPP Technical Specifications.

Sincerely,

Gregory M. Rueger

JE47

cc: Steven D. Bloom
Ellis W. Merschoff
David L. Proulx
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Enclosure

GRC/1392

Enclosure
PG&E Letter DCL-00-068

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

MARCH 29, 1998 through December 31, 1999

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

Acronyms

1C9	Unit 1 Cycle 9
1R8	Unit 1 Refueling, Cycle 8
1R9	Unit 1 Refueling, Cycle 9
2R8	Unit 2 Refueling, Cycle 8
2R9	Unit 2 Refueling, Cycle 9
ABS	Auxiliary Building Sump
AFW	Auxiliary Feedwater
AFWP	Auxiliary Feedwater Pump
AFWS	Auxiliary Feedwater Sump
AMSAC	ATWS Mitigation System Actuation Circuitry
AONDB	Axial ODSCC Not Detected by Bobbin
AOT	Allowed Outage Time
AR	Action Request
ASW	Auxiliary Saltwater
ATWS	Anticipated Transient Without Scram
BAST	Boric Acid Storage Tank
BCMS	Boron Concentration Measurement System
BOL	Beginning of Life
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CAP	Chemical Analysis Procedure
CASP	Containment Air Sample Panel
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CCWS	Component Cooling Water System
CDF	Core Damage Frequency
CET	Core Exit Thermocouples
CFW	Condensate and Main Feedwater
CIA	Containment Integrity Analysis
COT	Channel Operational Test
CP	Cathodic Protection
CRDM	Control Rod Drive Mechanism
CS	Containment Spray
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
CWP	Circulating Water Pump
DP	Differential Pressure
DBA	Design Basis Accident
DCM	Design Criteria Memorandum
DCP	Design Change Package
DCPP	Diablo Canyon Power Plant
DDE	Double Design Earthquake
DE	Design Earthquake?
DES	Draft Environmental Statement
DEG	Diesel Engine Generator
DEH	Digital Electrohydraulic
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
ECG	Equipment Control Guidelines
EDG	Emergency Diesel Generator
EDMS	Electronic Document Management System

Acronyms (continued)

EFPY	Equivalent Full Power Years
EH	Electrohydraulic
EOL	End of Life
EQ	Environmental Qualification
ERDS	Emergency Response Data System
ERFDS	Emergency Response Facility Data System
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
ETPO	Electric Transmission Planning and Operation
FHARE	Fire Hazards Appendix R Evaluations
FIN	Fix-It-Now
FMEA	Federal Emergency Management Administration
FSAR	Final Safety Analysis Report
FSARU	Final Safety Analysis Report Update
FWI	Feedwater Isolation
GDC	General Design Criterion
GL	Generic Letter
GM&C	Grid Maintenance and Construction
GTS	Generation Transmission System
HE	Hosgri Earthquake
HEPA	High-Efficiency Particulate Air (filter)
HSP	Hot Shutdown Panel
HVAC	Heating, Ventilating, and Air Conditioning
IDAP	Inter-Departmental Administrative Procedure
IFO	Industrial Fire Officers
ISO	Independent System Operator
IST	Inservice Testing
ITDP	Improved Thermal Design Procedure
ITS	Improved Technical Specifications
LA	License Amendment
LAR	License Amendment Request
LBIE	Licensing Basis Impact Evaluation (50.59)
LDBAP	Licensing and Design Bases Alignment Program
LHUT	Liquid Hold-Up Tank
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LRS	Liquid Radwaste System
LTC	Load Tap Changer
LTOP	Low Temperature Overpressure Protection
MEDT	Miscellaneous Equipment Drain Tank
MELB	Moderate Energy Line Break
MFPCS	Main Feedwater Pump Control System
MFW	Main Feedwater
MFWS	Main Feedwater Pump
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
NCR	Nonconformance Report
NDE	Nondestructive Examination
NFPA	National Fire Protection Association

Acronyms (continued)

NML	Nuclear Mutual Limited
NPDES	National Pollutant Discharge Elimination System
NQS	Nuclear Quality Services
NSAL	Nuclear Safety and Licensing
OD	Outside Diameter
OE	Operability Evaluation
PAM	Post-Accident Monitoring
PASS	Post-Accident Sampling System
PCD	Population Center Distance
PDP	Positive Displacement Pump
POD	Probability of Detection
PORV	Power (Pressure)-Operated Relief Valve
ppb	parts per billion
PPC	Plant Process Computer
ppm	parts per million
PRA	Probabilistic Risk Assessment
PSRC	Plant Staff Review Committee
PSV	Pressurizer Safety Valve
PTLR	Pressure and Temperature Limits Report
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
QE	Quality Evaluation
RCCA	Rod Cluster Control Assembly
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR	Residual Heat Removal
RI	Refueling Interval
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSE	Reload Safety Evaluation
RTD	Resistance Temperature Detector
RVLIS	Reactor Vessel Level Instrumentation System
RWMS	Reactor Water Makeup System
RWST	Refueling Water Storage Tank
SAR	Safety Analysis Report
SBO	Station Blackout
SCMM	Subcooling Margin Monitor
SECL	Safety Evaluation Check List
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SG	Steam Generator
SGBTVC	Steam Generator Blow Down Tank Vent Condenser
SGTR	Steam Generator Tube Rupture
SIP	Safety Injection Pump
SISI	Seismically Induced Systems Interaction
SJAE	Steam Jet Air Ejectors
SLUR	Second Level Undervoltage Relay
SOV	Solenoid Operated Valve
SPDS	Safety Parameter Display System
SR	Special Report
SBLOCA	Small Break Loss-of-Coolant Accident

Acronyms (continued)

SSC	Structures, Systems, or Components
SSERs	Supplement to SER (Safety Evaluation Report)
SSPS	Solid State Protection System
STP	Standard Temperature and Pressure
SUT	Startup Transformers
TADOT	Trip Actuating Device Operational Tests
TMS	Thermocouple Monitoring System
TOC	Transmission Operations Center
TOL	Thermal Overload
TRG	Technical Review Group
TS	Technical Specification
TSC	Technical Support Center
TSIP	Technical Specifications Improvement Program
TSP	Tube Support Plate
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
USQ	Unreviewed Safety Question
VCT	Volume Control Tank
VLPMS	Vibration and Loose Parts Monitoring System

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98-046 ECG 9.1 Accumulator Pressure and Water Level Instrumentation – Surveillance Requirement 9.1.2

Reference Document No.: ECG 9.1
Rev. No: 2
Reference Document Title: ECG 9.1 Accumulator Pressure and Water Level Instrumentation – Surveillance Requirement 9.1.2

Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 Evaluation is required for all ECG changes. In addition, the proposed change to the ECG surveillance requirements for accumulator pressure and water level instrumentation alters the frequency at which the surveillance is conducted, and thus involves a change to the administrative controls over plant activities as described in the FSAR. This change also alters the frequency at which the surveillance test procedures for accumulator instrument calibration are performed. The frequency of these test activities is proposed to be changed from once every 18 months to once per refueling interval (24 months) in accordance with the methodology included in LAR 97-07 and approved by LAs 123/121.

Safety Evaluation Summary:
The extension of the surveillance frequency for the accumulator pressure and water level indication and alarm devices, covered by this ECG, does not alter the way any important to safety structure, system, or component functions, does not change the manner in which the plant is operated, and does not contribute to the probability of occurrence of any accident. The accumulator pressure and level instrumentation only provides indication and alarm functions which cannot initiate an accident already evaluated in the FSAR. Therefore, the proposed change does not increase the probability of occurrence of an accident previously evaluated in the SAR.

The analysis to determine the effect of the longer surveillance frequency focuses on instrument calibration data and addresses the issues discussed in Enclosure 2 of Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." The surveillance test results are the instrument calibrations. The calibrations reflect instrument maintenance and operating experience, in that the instruments are continuously operated, and maintenance is typically performed only as required by the equipment qualification program or in response to an as-found or as-left out-of-tolerance condition which is captured in the calibration data.

Conclusion:
The proposed 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 will not alter the operation of these instrument channels, the intent or method by which the associated surveillances are presently conducted, or the scope or intent of the associated surveillance test procedures. This activity does not result in an unreviewed safety question. This activity does not result in an unreviewed safety question.

98-047 ECG 37.1 Hot Shutdown Panel (HSP) Neutron Flux Indicators (Appendix R) – Surveillance Requirement 37.1.2

Reference Document No.: ECG 37.1
Rev. No: 1
Reference Document Title: ECG 37.1 Hot Shutdown Panel (HSP) Neutron Flux Indicators (Appendix R) – Surveillance Requirement 37.1.2

Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 Evaluation is required for all ECG changes. In addition, the proposed change to the ECG surveillance requirements for the Hot Shutdown Panel (HSP) Neutron Flux Indicators (Appendix R) alters the frequency at which the surveillance is conducted, and thus involves a change to the administrative controls over plant activities as described in the SAR. This change also alters the frequency at which the calibration of these indicators is performed. Calibration of these indicators is performed as part of I&C Maintenance Test Procedure # MP I-1.2-1, "Calibration of the Wide Range Nuclear Instrumentation Monitors NE-51 and NE-52." The frequency of these test activities is proposed to be changed from once every 18 months to once per refueling interval (24 months).

Safety Evaluation Summary:
The Neutron Flux Indicators which are covered by this ECG are located on the Hot Shutdown Panel (HSP). These indicators provide a diverse method of verifying that the reactor is shutdown in the event of a fire in the Control Room and/or the Cable Spreading Room. Among the other means available for verifying that the reactor is shutdown from outside of the Control Room are the Reactor Trip Breaker Position Indication. There are no design basis accident scenarios or events identified in the FSAR Update which consider or take credit for these indicators, nor do these indicators perform any accident mitigation functions.

The HSP Neutron Flux Indicators provide an alternate method of verifying that the reactor is shutdown in the event of a fire in the Control Room and/or the Cable Spreading Room. As such they are used to satisfy the requirements of

10 CFR 50 Appendix R which requires that the ability to bring the plant to a safe shutdown condition after a fire. However, Appendix R also states that "Shutdown systems installed to ensure post-fire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where required for other reasons" (Ref. Appendix R, Section III.L.6). As a consequence, Appendix R rules do not require assumption of coincident fire and design basis accidents.

Conclusion:
The proposed 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 does not result in any physical modifications to these indicators, the components in the associated instrument string, or to any other important to safety structure, system or component. In addition, increasing the surveillance interval will not alter the operation of these indicators or the associated instrument string, the intent or method by which the associated surveillances are presently conducted, or the scope or intent of the associated test procedures. Finally, increasing the surveillance interval does not affect safety function performance, or alter the intent or method by which surveillance tests are performed. This activity does not result in an unreviewed safety question.

98-050 FHARE 37, Rating of Barriers Between CCW Pump Rooms

Reference Document No.: FHARE 37
Rev. No: 3
Reference Document Title: FHARE 37, Rating of Barriers Between CCW Pump Rooms

Safety Evaluation Description:
FHARE 37 revision 3 evaluates the acceptability of non-rated penetrant seal designs and voids in the CCW pump cubicle walls which were not evaluated in previous revisions of the FHARE. FHARE 37 was also revised to clarify the safe shutdown compliance in the CCW pump room cubicles.

Safety Evaluation Summary:
FHARE 37, Revision 3 evaluates the acceptability of non-rated penetration seal designs and voids which were identified on the CCW pump cubicle barriers. This FHARE does not introduce new hazards in the CCW pump room, so the probability of occurrence of a fire is not increased. This FHARE also does not affect the probability of occurrence of other accidents previously evaluated. The results of FHARE 37 have determined that the consequences of a fire will not affect the ability to achieve and maintain safe shutdown conditions. The non-rated features and voids in the CCW pump room cubicle walls will not adversely affect the ability of the wall to provide passive fire protection. The FHARE does not impact other accidents, therefore, the

consequences of other accidents previously evaluated in the SAR will not be increased. This revision to FHARE 37 will not create a new type of accident. The non-rated features will not affect the ability of the CCW pump room cubicle walls to provide adequate passive protection against the hazards of the cubicle rooms. These fire area boundaries are not credited in other design basis accidents.

Conclusion:
FHARE 37 evaluates the consequences of a fire which is postulated to originate in the CCW pump. FHARE 37 has concludes that the consequences of fire damage to circuits in the CCW pump rooms will not affect the ability to achieve and maintain safe shutdown conditions. Therefore, the consequences of fire propagation will not increase the malfunction of equipment.

98-052 FSAR Update, Section 3.2

Reference Document No.: FSAR Update, Section 3.2
Rev. No: Living
Reference Document Title: FSAR Update, Section 3.2
Safety Evaluation Description:
ARs A0451386 and A0451389 identified a number of items in Section 3.2 of the FSAR Update which are of the nature of errors, discrepancies, inconsistencies, etc. This FSAR Update change request disposes of these items, each of which is discussed fully in the change request itself. Since these constitute changes to a licensing basis document (the FSAR Update), an LBIE Screen is documented here.

One change represents only a change in the description of a piping subsystem on the suction side of the AFW pumps; the classification of the piping system is not changed, only a more correct description of the piping system is given. This is not discussed further in this LBIE.

The other represents what may be taken to be a downgrading or reclassification of safety-related HVAC ductwork. However, this is not the case, as is reviewed in the following LBIE.

Safety Evaluation Summary:
Many of the Design Class I HVAC systems are associated with the mitigation of the various accidents/events. The changes to the Piping Schematic Correlation table in the FSAR Update text does not represent any reduction in the capabilities of these HVAC systems to perform their accident mitigation functions since: the ductwork is still classified as Design Class I, it is still Seismic Category I, and it is still covered by the PG&E 10 CFR 50 Appendix B QA Program. A quality code classification for this ductwork was inappropriate

for the vintage of DCPP and the type of commodities (ductwork vs. Piping). Further, a quality code classification would not invoke any additional assurance of quality beyond that which is already provided. As a point of clarification, the containment penetrations which are used for various HVAC functions consist of piping, not ductwork, and are correctly designated as Quality Code Class II. This assures that the required safety function (i.e., containment isolation) is provided by properly classified components.

This ductwork is important-to-safety equipment in that it provides certain accident mitigating functions for which credit has been taken. The ductwork consists of passive components which are not subject to active malfunction. The structural and pressure boundary integrity of the ducting has been assured by the various seismic and other analyses, by the use of quality-assured materials and material properties, and by QA oversight during their original construction and any subsequent modifications and maintenance. The lack of an (unnecessary) quality code class for these commodities does not increase the likelihood of their failure. The ductwork also provides cooling of important, accident mitigating equipment such as pumps, valves, and instrumentation. Since the integrity of this ductwork is not compromised by the described FSAR Update text change, the cooling system is no more likely to cause a malfunction of this equipment.

Conclusion:
There are no physical changes to the plant resulting from this text change. The ductwork itself is not associated with the creation of any accident-type events. Dropping of an unnecessary quality code classification for the ductwork cannot make any previously incredible event more likely. This evaluation does not result in an unreviewed safety question.

98-053 Routine Surveillance Test Of PDP 2-3

Reference Document No.: STP P-PDP-23
Rev. No: 7
Reference Document Title: Routine Surveillance Test Of PDP 2-3
Safety Evaluation Description:
This LBIE evaluates the addition to STP P-PDP-23 of an attachment to provide a dedicated licensed operator to turn to the PDP 9ff upon actuation of a Safety Injection. This step will limit the mass injection to the RCS due to a spurious SI. Based upon the analyses prepared by Westinghouse, when the PDP is assumed to be secured in a timely manner, and with the other margins available, DCPP still meets the accident analysis in the FSAR, that the overfill will be terminated by the operators within the analyzed capability of the pressurized safety valves (PSVs) to relieve water. The current analysis for a spurious safety injection was incorporated into the FSAR in September 1996 and reported to the NRC in January 1997 (DCL-97-013). The analysis for the

event includes modeling water releases through the PSVs.

Errors in the analysis have been identified that make the FSAR analysis non-conservative. These errors are in the empirical modeling of the behavior of PSVs during water relief, based upon testing performed by EPRI. Westinghouse has performed a number or sensitivity analyses since discover of the errors in December. The Westinghouse analyses have identified that the ECCS pump performances used in the analysis of record are conservative with respect to current plant pump performance. Additionally, the analyses show substantial benefit when the PDP is not operating.

Safety Evaluation Summary:
There is no SAR accident that could be initiated as a result of the Operator actions specified by this change. Operator action to secure the PDP upon actuation of a SI will be performed by a dedicated licensed operator who has no concurrent duties or responsibilities to respond to Emergency Operating Procedures. An additional dedicated, qualified operator will be stationed near the PSP breaker cubical, with a speed wrench staged to open the cubical door if needed. The operator by the breaker will be in communication with the dedicated operator in the control room. If the control switch fails to open the breaker form the control room, the operator by the breaker will locally trip the breaker. The operator actions to secure the PDP upon SI actuation is a task which is within the training of a licensed operator. The responsibilities of the operators will be emphasized with a tailboard prior to operation of the pump. These are simple actions, the controls are readily accessible, and are completely in the control room or switchgear room. Misoperation of other controls will not affect the plant response. Once the actions to secure the PDP are completed, these individuals may be assigned other actions. The Operations staff implementation of the EOPs will not be negatively affected. There are no changes to the EOPs and the performance of time sensitive actions will not be changed from that previously evaluated in the FSAR.

Conclusion:
The NRC has not relied upon the specific analysis of the spurious SI in the SER or SSERs. Based on the conservatisms in the assumption analysis, there is no change in the expected operation of the PSVs in a spurious SI event. This activity does not result in an unreviewed safety question.

98-054 Routine Surveillance Test Of PDP 1-3

Reference Document No.: STP P-PDP-13
Rev. No: 5
Reference Document Title: Routine Surveillance Test Of PDP 1-3
Safety Evaluation Description:
This LBIE evaluates the addition to STP P-PDP-13 of an attachment to

provide a dedicated licensed operator to turn to the PDP upon actuation of a Safety Injection. This step will limit the mass injection to the RCS due to a spurious SI. Based upon the analyses prepared by Westinghouse, when the PDP is assumed to be secured in a timely manner, and with the other margins available, DCPP still meets the accident analysis in the FSAR, that the overfill will be terminated by the operators within the analyzed capability of the pressurized safety valves (PSVs) to relieve water.

The current analysis for a spurious safety injection was incorporated into the FSAR in September 1996 and reported to the NRC in January 1997 (DCL-97-013). The analysis for the event includes modeling water releases through the PSVs.

Errors in the analysis have been identified that make the FSAR analysis non-conservative. These errors are in the empirical modeling of the behavior of PSVs during water relief, based upon testing performed by EPRI. Westinghouse has performed a number or sensitivity analyses since discover of the errors in December. The Westinghouse analyses have identified that the ECCS pump performances used in the analysis of record are conservative with respect to current plant pump performance. Additionally, the analyses show substantial benefit when the PDP is not operating.

Safety Evaluation Summary:
There is no SAR accident that could be initiated as a result of the Operator actions specified by this change. Operator action to secure the PDP upon actuation of a SI will be performed by a dedicated licensed operator who has no concurrent duties or responsibilities to respond to Emergency Operating Procedures. An additional dedicated, qualified operator will be stationed near the PSP breaker cubical, with a speed wrench staged to open the cubical door if needed. The operator by the breaker will be in communication with the dedicated operator in the control room. If the control switch fails to open the breaker form the control room, the operator by the breaker will locally trip the breaker. The operator actions to secure the PDP upon SI actuation is a task which is within the training of a licensed operator. The responsibilities of the operators will be emphasized with a tailboard prior to operation of the pump. These are simple actions, the controls are readily accessible, and are completely in the control room or switchgear room. Misoperation of other controls will not affect the plant response. Once the actions to secure the PDP are completed, these individuals may be assigned other actions. The Operations staff implementation of the EOPs will not be negatively affected. There are no changes to the EOPs and the performance of time sensitive actions will not be changed from that previously evaluated in the FSAR.

Conclusion:
The NRC has not relied upon the specific analysis of the spurious SI in the SER or SSERs. Based on the conservatisms in the assumption analysis, there is no change in the expected operation of the PSVs in a spurious SI event. This activity does not result in an unreviewed safety question.

98-056 Issues Related to Audit of the ASW System

Reference Document No.: NCR N0001784
Rev. No: 1
Reference Document Title: Issues Related to Audit of the ASW System
Safety Evaluation Description:
This LBIE is required because we are changing the way we are meeting a commitment to the NRC to trend CCW heat exchanger D/P and calcium carbonate deposition in the CCW heat exchangers that was made in letter DCL-94-037

As part of our response to NRC Inspection Reports 50-275/93-36 and 50-323/93-36 in letter DCL-94-037, PG&E committed to trending CCW heat exchanger D/P. This was to address an NRC concern about calcium carbonate deposition in the outlet waterbox of the unit 2 CCW heat exchangers. This deposition was due to cathodic protection currents going to maximum when the heat exchangers were out of service. In addition AR A0324658 AE 33 documented a verbal commitment to the NRC to measure the inside diameters of a sample of tubes in the outlet waterbox every outage as a method trending calcium carbonate deposition.

A reconvene of the TRG for NCR N0001784 has determined that a change to the unit 2 operating procedure OP E-5:IV addresses the root cause of the calcium carbonate deposition and the trending of D/P or carbonate buildup is no longer needed. The change added steps to vent the ASW side of the CCW heat exchangers when the heat exchangers are taken out of service. This venting allows the water to drain from the out of service heat exchanger, eliminating the current path and the deposition. The unit 1 CCW heat exchangers drain when they are taken out of service, so deposition has never been a problem on unit 1.

The TRG has determined that the change to OP E-5:IV addresses the NRC's concern that we were not trending the carbonate buildup. In addition, STP I-1A will still record the CCW heat exchanger D/P and initiate cleaning at 130 inches, so the operation and maintenance of the CCW heat exchangers will not change. We only eliminate a trending program that is not providing value and is no longer required.

Safety Evaluation Summary:
On March 27, 1998, the TRG for NCR N0001784 determined that a change to the Unit 2 operating procedure OP E-5:IV now substantially reduces the build up of calcium carbonate. As a result, the D/P trend program and carbonate buildup measurements are no longer needed. The procedure change added steps to vent the ASW side of the CCW heat exchangers after the heat exchangers are taken out of service. Venting allows the water to drain from the out-of-service heat exchanger, therefore eliminating the cathodic protection current path and the substantially reducing the deposits. Because of a different piping configuration, the Unit 1 CCW heat exchangers drain when taken out of service, so deposits have not been a problem .

Conclusion:
By requiring the CCW heat exchangers on unit 2 to be vented, the calcium carbonate deposition has been eliminated. The venting of the unit 2 CCW heat exchangers should be included in the commitment database to replace the commitments to monitor and trend heat exchanger D/P.

Deletion of the commitments to trend D/P and calcium carbonate buildup dose not result in an unreviewed safety question. The current practice of venting the unit 2 CCW heat exchanger when it is out of service, addresses the calcium carbonate buildup.

98-057 F-0, "Critical Safety Function Status Trees", - Unit 1 Rev. 11, Unit 2
FR-C.1, "Response to Inadequate Core Cooling" - Unit 1 Rev. 11, Unit 2
FR-C.2, "Response to Degraded Core Cooling" - Unit 1 Rev. 10, Unit 2

Reference Document No.:
Rev. No: Rev. 6, Rev. 6, Rev. 5
Reference Document Title:
Safety Evaluation Description:
PG&E calculation PAM-0-07-1310 was revised to incorporate the Westinghouse 24 month fuel cycle RVLIS Uncertainty Report results. As a result of the report and review of the RVLIS functional requirements new values are needed for the RVLIS Full Range setpoints currently in EOPs F-0, FR-C.1, FR-C.2. The following will be updated with revised values:

Recognition of Inadequate Core Cooling: The previous functional requirement was that full range RVLIS indicate 3.5 feet from the bottom of the core with a +/- 2.5 foot accuracy (EOP setpoint 39%). Westinghouse has recommended that a revised functional requirement be implemented to require an EOP setpoint of 7.0 feet from the bottom of the core minus channel uncertainties. This will result in approximately 32% RVLIS full range setpoint to be used in determining an inadequate core cooling situation. The revised setpoint takes into account the larger channel uncertainties and precludes an inadvertent

transition to EOP FR-C.1, "Response to Inadequate Core Cooling" for a design basis small break LOCA. This setpoint revision affects EOPs F-0, FR-C.1, and FR-C.2.

The full range RVLIS operator action setpoint in the vicinity of the core mid-plane will be revised from 39% to approximately 32%. The purpose of this safety evaluation is to evaluate the change in basis of the setpoint and not the exact operator decision point.

Safety Evaluation Summary:
The revision of the operator action setpoint for full range RVLIS in the vicinity of the core mid-plane does not affect the probability of occurrence of an accident. This setpoint may alter the time in the accident scenario when the operator will takes the prescribed action. However, the probability of the accident occurring is independent of the RVLIS operator action values.

Revising the decision point for inadequate core cooling to approximately 32% does not affect the consequences of a design basis small break LOCA (SBLOCA) because this value ensures that the operator will not diagnose a design basis SBLOCA as an inadequate core cooling condition and take inappropriately severe mitigating actions. The vessel water level is not postulated to fall to approximately 32% during a design basis SBLOCA. The EOP network will still provide instructions for an outside design basis condition that leads to implementing procedures that mitigate degraded or inadequate core cooling.

Conclusion:
The revision of the full range RVLIS setpoint does not require that safety related equipment be operated in a different manner than what is described in the SAR. The revision of the decision point to go to the inadequate or degraded core cooling procedures (FR-C.1 and FR-C.2) ensures that the equipment is operated exactly as described for design basis SBLOCA.

98-059 Unit 2 Turbine Building 4.16 kV Switchgear Room Area Tornado Bracing Base Plate Modifications

Reference Document No.: C-050420
Rev. No: 0
Reference Document Title: Unit 2 Turbine Building 4.16 kV Switchgear Room Area Tornado Bracing Base Plate Modifications

Safety Evaluation Description:
This evaluates a change to FSAR Figure 3.1-2 to reflect the revised configuration of bracing connection anchorage to underside of concrete slab at Elevation 140 ft of the Turbine Building.

Safety Evaluation Summary:
The bracing base plate modifications associated with the structural support system for the exterior metal siding inside/outside the 4.16 W Switchgear Rooms do not increase the probability of an accident previously evaluated in the SAR.

The consequences of malfunction of equipment important to safety have not increased since the associated stored equipment and components inside the 4.16 W Switchgear and Cable Spreading Rooms of the Turbine Building have not changed and remain in the same locations, regardless of these DCP base plate strengthening modifications.

There is no possibility that an accident of a different type, than previously evaluated in the SAR, can be created by these DCP base plate strengthening modifications. The Turbine Building exterior metal siding and associated structural support systems have been evaluated for all design basis tornado loads as required by DCM T-9 and for all non-tornado design basis load conditions as required by DCM T-4; and it has been determined that the building superstructure can withstand the various design basis demands.

Conclusion:
The low probability of occurrence for a 200 mph tornado at DCPP site remains unchanged (i.e., approximately 10^{-7}) and the probability of tornado-induced missiles striking DCPP's small vulnerable areas, such as the 4.16 kV Switchgear and Cable Spreading Rooms inside the Turbine Building, remains negligible (Refs: SER 1974 and SSER 7). This activity does not result in an unreviewed safety question.

98-060 Unit 1 Turbine Building 4.16 kV Switchgear Room Area Tornado Bracing Base Plate Modifications

Reference Document No.: C-049420
Rev. No: 0
Reference Document Title: Unit 1 Turbine Building 4.16 kV Switchgear Room Area Tornado Bracing Base Plate Modifications

Safety Evaluation Description:
This evaluates a change to FSAR Figure 3.1-2 to reflect the revised configuration of bracing connection anchorage to underside of concrete slab at Elevation 140 ft of the Turbine Building.

Safety Evaluation Summary:
The bracing base plate modifications associated with the structural support system for the exterior metal siding inside/outside the 4.16 W Switchgear Rooms do not increase the probability of an accident previously evaluated in the SAR.

The consequences of malfunction of equipment important to safety have not increased since the associated stored equipment and components inside the 4.16 W Switchgear and Cable Spreading Rooms of the Turbine Building have not changed and remain in the same locations, regardless of these DCP base plate strengthening modifications.

There is no possibility that an accident of a different type, than previously evaluated in the SAR, can be created by these DCP base plate strengthening modifications. The Turbine Building exterior metal siding and associated structural support systems have been evaluated for all design basis tornado loads as required by DCM T-9 and for all non-tornado design basis load conditions as required by DCM T-4; and it has been determined that the building superstructure can withstand the various design basis demands.

Conclusion:
The low probability of occurrence for a 200 mph tornado at DCPP site remains unchanged (i.e., approximately 10^{-7}) and the probability of tornado-induced missiles striking DCPP's small vulnerable areas, such as the 4.16 kV Switchgear and Cable Spreading Rooms inside the Turbine Building, remains negligible (Refs: SER 1974 and SSER 7). This activity does not result in an unreviewed safety question.

98-062 Control Of Doors Important To Safety

Reference Document No.: DCP C-049423
Rev. No: 0
Reference Document Title: Control Of Doors Important To Safety
Safety Evaluation Description:
The proposed activity is a document change only DCP. It involves revising the door schedule drawings to update information on Maintenance Rule (MR) applicability, safety barrier functions and the applicable Tech Spec's and ECGs. The change will also reformat the drawings to organize the data more effectively.

The activity is being implemented as a result of the NRC Enforcement Conference on ASW Drain Check Valve issue in September 1997. Concern was raised about other non-Class I components that protect and support safety-related functions of Class I systems. A0443772 identified doors as Class 11 components that perform safety barrier functions due to fire, HELB,

Flood and HVAC. In addition, doors that perform safety functions are scoped within the Maintenance Rule program and thus need to be identified in the schedule.

Safety Evaluation Summary:
Plant doors are not the cause of any accident evaluated in the SAR. The procedures and STP are revised and a new ECG is created to enhance the method doors important to safety are controlled, maintained and tested. Design drawings, calculations and DCM's are revised to support the changes to the plant procedures.

As this is a document change only design change and there are no physical changes, the consequences of an accident previously evaluated in the SAR will not be increased. The changes to the procedures and the creation of a new ECG are considered an enhancement to the method doors important to safety are controlled, maintained and tested and therefore increasing their reliability.

The enhancements to documents created by this DCP are being made to further assure that doors will not malfunction and will properly perform their function to mitigate the effects of a fire or an accident. By establishing a program of assuring that doors meet written acceptance criteria and are inspected at specified intervals, the probability of occurrence of a malfunction of a door will actually be decreased.

Conclusion:
The revised procedures and new ECG enhance the methods whereby plant doors will be controlled, maintained and tested. The enhanced method to control doors will provide additional assurance the doors important to safety will perform their function. This activity does not result in an unreviewed safety question.

98-063 Upgrade Intake Debris Screens to Design Class I

Reference Document No.: DCP H-049401
Rev. No: 0
Reference Document Title: Upgrade Intake Debris Screens to Design Class I

Safety Evaluation Description:
The DCP upgrades the debris screens for the 48" containment purge valves from Design Class II to Design Class I. This upgrade involves revision to the Q-List, DCM T-16 and DCM-23A. Updating these documents to change the design classification of components in the plant constitutes a change to the facility design.

Safety Evaluation Summary:
The debris screens, the flexible connection and the duct branch are part of the containment purge system. The vacuum relief dampers are in the containment fan cooler system. These two systems are not involved in nor the cause of any accidents previously evaluated in the SAR.

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 SAR is based on full closure of the isolation valves within the predetermined stroke time. This ensures that the off-site dose limit per 10 CFR 100 is not exceeded. The 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 the structural adequacy of the debris screen and the flexible connection is not affected. This will assure the containment isolation function of the valves. With the containment purge valves fully closed within the prescribed time, the containment integrity is maintained and the off-site dose will remain within the limits stated in the SAR.

Debris lodged in the seat of the containment purge valves may prevent full closure of the valves and cause malfunction of this equipment important to safety. The debris screens are designed to prevent the introduction of debris into the valves. Upgrading the debris screens and associated components to Design Class I will ensure that the debris screens will maintain its structural integrity and not result in an interaction with the containment isolation valves.

Conclusion:
The debris screens, the flexible connection and the duct branch are part of the containment purge system. The vacuum relief dampers are in the containment fan cooler system. One of the functions of these two systems is for mitigation of the consequences of an accident. Upgrading these components to Design Class I will enhance the accident mitigation functions of these two system. This activity does not result in an unreviewed safety question.

98-065 ECG 39.2 Air Ejector Gaseous Effluent Monitors - Surveillance Requirements 39.2.1 and 39.2.3

Reference Document No.: ECG 39.2
Rev. No: 3
Reference Document Title: ECG 39.2 Air Ejector Gaseous Effluent Monitors - Surveillance Requirements 39.2.1 and 39.2.3

Safety Evaluation Description:
Per Procedure OPDC16, a 10 CFR 50.59 Evaluation is required for all ECG changes. In addition, the proposed changes to the ECG surveillance requirements for the Air Ejector Gaseous Effluent Monitors change the frequency at which the channel calibration surveillance is conducted and add a new channel check surveillance requirement. As a result, the proposed surveillance requirement changes involve a change to the administrative controls over plant activities as described in the SAR. These changes also:

1. Alter the frequency at which the surveillance test procedure STP 1-39-RI5.B, 'Condenser Air Ejector Discharge RM-15/15R Radiation Monitor Calibration' is performed. The frequency of these test activities is proposed to be changed from once every 18 months to once per refueling interval (24 months).
2. Add the additional requirement for a channel check to be performed on these instrument channels on a monthly basis. The addition of these monthly channel check requirements will result in a revision of surveillance test procedure STP I-ID, "Routine Monthly Checks Required By Licenses."

Safety Evaluation Summary:
A review of the accident scenarios evaluated in the FSAR Update identifies that the relevant accident analysis is that of a Steam Generator Tube Rupture (SGTR). As indicated in Section 15.4.3.1 of the FSAR Update, the Air Ejector Radiation Monitors will alarm, indicating a sharp increase in the radioactivity concentration in the secondary system. The proposed change to the surveillance interval of the ECG surveillance requirement from a nominal 18 months to a nominal 24 months has no impact on the ability of these radiation monitoring channels to perform their intended detection and indication functions. Likewise, the addition of the channel check surveillance requirement will not have an adverse impact on the ability of these radiation monitoring channels to perform their intended detection and indication functions. In fact the addition of this monthly surveillance requirement will provide added assurance that the instrument channels will be available when required.

The Air Ejector Gaseous Effluent Radiation Monitors (RE-15/15R) will detect and provide indication of a SGTR by monitoring the radioactivity concentration contained in the discharge of the Steam Jet Air Ejectors (SJAE) prior to its release through the plant vent to the environment. These radiation monitoring channels provide a monitoring, indication and alarm annunciation function. They do not contribute to the occurrence of an accident, and do not provide an automatic control function. These radiation monitoring channels are identified in the SAR as one of a diverse set of indicators for a SGTR type accident. These radiation monitoring channels are used by plant operating

personnel for the early detection and assessment of Steam Generator tube leakage. Early detection of Steam Generator tube leakage allows operating personnel to take preemptive actions, as described in OP 0-4, to mitigate the impact and consequences of the detected Steam Generator tube leakage or a SGTR.

Conclusion:
The surveillance, maintenance and operating histories indicate that the Air Ejector Gaseous Effluent Monitors will continue to effectively perform their design function for the longer operating cycles. This activity does not result in an unreviewed safety question.

98-078 Provide Alternate Steam Generator 2-1 Sample Path

Reference Document No.: DCP M-50440
Rev. No: 0
Reference Document Title: Provide Alternate Steam Generator 2-1 Sample Path

Safety Evaluation Description:
This DCP provides an alternate steam generator blowdown sample path for Steam Generator (SG) 2-1 while MS-2-1048 is closed to isolate a body to bonnet leak. The normal blowdown sample path is isolated by closing MS-2-1048 (to stop the body to bonnet leak). The alternate sample path is provided by connecting sample tubing to a drain valve on the normal steam generator blowdown line to the flash tank and tying into the normal blowdown sample line at a drain line upstream of SG 2-1 blowdown sample cooler. The applicable design drawing is DC 108004 Sht. 7. The plant OVID drawing showing the applicable valve numbers is 107704 Sht. 14. This activity interfaces with licensing basis documents by affecting FSAR Update Figure 3.2-4 Sht. 10 of 14. In addition, this change affects the method of providing a blowdown sample for identifying the affected SG in the case of high activity detected in the SG blowdown sample system as described in the SAR (Section 11.4.2.2.1).

Safety Evaluation Summary:
The relevant accidents associated with this change are a Steam Generator Tube Rupture (SGTR), (FSAR Update Section 15.4.3) and Minor Secondary System Pipe Break (FSAR Update Section 15.3.2).

The proposed change affects the method, as described in the SAR of obtaining a blowdown sample from SG 2-1 following a high activity alarm and subsequent blowdown sample isolation. Providing an alternate blowdown sample path cannot initiate a SGTR, or cause a minor secondary pipe break.

FSAR Update Section 11.4.2.2.1 specifically states that subsequent identification of the leaking steam generator (following high radiation isolation) would then be made by manual override of the sample line isolation and drawing separate samples from each steam generator. This modification will require override of the blowdown system isolation valves to provide a sample from SG 2-1. This change in method of obtaining a steam generator blowdown sample cannot initiate an accident.

The small bore tubing is less than 6" and would fall within the category of minor secondary pipe breaks. The piping and tubing components for this change meet or exceed the requirements of the original design specifications. The consequences of Minor Secondary System Pipe Breaks are detailed in Sections 15.2.13 and 15.3.2 of the FSAR Update. Failure of the alternate blowdown sample line is enveloped by this evaluation.

Steam generator blowdown activity is not credited as the primary source for SG identification for the design basis SGTR analysis. High secondary activity, as indicated by the main steamline radiation monitors, and the air ejector radiation monitor, will typically provide the first indication of a design basis SGTR event. Small steam generator tube leaks where the potential for steam generator overfill is not significant are identified by chemistry samples using the blowdown sample system. The alternate blowdown sample pathway provided by this change for SG 2-1 allows samples to be taken and analyzed consistent with the intent of CAP AP-1, Prompt Steam Generator Leak Identification Procedure.

Conclusion:
This modification takes place in the Design Class II portion of the SG blowdown sample system downstream of the blowdown system containment isolation valves which is the equipment important to safety for this modification. Other than closing MS-2-1048, no equipment important to safety is affected by the proposed change of providing an alternate blowdown sample path for SG 2-1. The alternate blowdown sample line will meet or exceed the original design specifications for material and construction practices including the use of proper supports, routing, and environmental qualification criteria for the material. This activity does not result in an unreviewed safety question.

98-079 Ventilation Opening Above Unit 1 & 2 AFW Pumps

Reference Document No.: FHARE 9
Rev. No: 3
Reference Document Title: Ventilation Opening Above Unit 1 & 2 AFW Pumps

Safety Evaluation Description:
FHARE 9 is part of the Fire Protection Program, and FHARE 9 evaluates the adequacy of having a building ventilation opening in the rated barrier above the AFW Pump. This FHARE is performed in accordance with the guidance contained in Generic Letter 86-10.

Safety Evaluation Summary:
FHARE 9 evaluates the as-built condition against the effects of a postulated fire. No new fire hazards are being introduced, and the fire protection features are not being modified or changed. This revision concludes this change will not adversely impact the DCPP fire protection program, and therefore, the probability of occurrence of an accident previously evaluated in the SAR will not be increased.

This revision to FHARE 9 evaluates an existing ventilation opening in a rated barrier for impact on the ability to achieve and maintain safe shutdown. The combustible loading is low in the affected fire areas, and the existing fire protection features are adequate for the low hazards. This FHARE concludes the effects of a postulated fire, the consequences of a design basis accident will not be increased. This FHARE is not involved with any other types of design basis accident described in the FSAR, and therefore, the consequences of an accident previously evaluated in the SAR will not be increased.

The FHARE does not affect the normal function of equipment important to safety, and there is no impact on the ability to achieve and maintain safe shutdown.

Conclusion:
Although the subject ventilation opening s are note dampered, and contain numerous small penetrants, the capability to achieve safe shutdown will not be adversely affected due to the availability of redundant safe shutdown functions which are outside the fire area, the low combustible loading, smoke detection, and the automatic sprinkler system. The effects of the design basis fire do not adversely affect the capability to achieve safe shutdown, and a fire in fire zone 3-R (3-W), is not expected to affect safe shutdown. This evaluation does not result in an unreviewed safety question.

98-081 Install and Remove Isolation Valve in EH Line to Trip Pilot Valve of Main Steam Stop Valves

Reference Document No.: MMD M000071
Rev. No: 1
Reference Document Title: Install and Remove Isolation Valve in EH Line to Trip Pilot Valve of Main Steam Stop Valves

Safety Evaluation Description:
Installation of an isolation valve in the DEH control line to the trip pilot valve for the main steam stop valve involves revision to the Piping Schematic Drawings 102020, Sheet 3 and 108020, Sheet 3. The Piping Schematics are figures in the FSAR. Since installation of this isolation valve may also affect the operation of the main steam stop valve system, this design change constitutes a change to the facility design as described in the SAR. An LBIE is performed to evaluate the impact of this design change on the licensing bases of the plant.

Safety Evaluation Summary:
The main steam stop valve and its associated equipment are non-safety related components. The stop valves are required to close on a turbine trip to prevent overspeed of the turbine. FSAR Section 15.2.7 discusses the loss of external electrical load and/or turbine trip accident which would result in a reactor trip. Addition of an isolation valve in the DEH control line to the main steam stop valve trip pilot valve will not increase the probability of occurrence of a turbine trip since the new valve will not cause the closure of the stop valve.

The main steam stop valve system is provided as an additional safety device to limit overspeeding the turbine, should the governor valve fail to close when the overspeed trip mechanism operates. The turbine overspeed trip will also cause the reactor trip signal if the P-9 limit is exceeded. Tripping of the reactor will ensure that the off-site dose limit per 10 CFR 100 is not exceeded. Addition of the new isolation valve will not affect the reactor tripping signal nor preventing the stop valve from closure, as discussed in FSAR Section 15.2.7 and thus will not increase the consequences of an accident previously evaluated in the SAR.

The main steam stop valve is provided as an additional safety device to limit overspeeding the turbine, should the governor valve fail to close when the overspeed trip mechanism operates. The trip pilot valve is provided to relieve the unbalanced steam pressure acting on the inner end of the stop valve shaft, when the stop valve closes. The trip pilot valve is actuated by hydraulic fluid from the DEH system. The isolation valve added by this design change in the DEH control line to the trip pilot valve of a main steam stop valve could cause a delay in closure of the stop valve if the new isolation valve is inadvertently left in the closed position during normal plant operation. Inadvertent closure of the new isolation valve could trap DEH fluid in the trip pilot valve actuator, causing it to remain closed. Closure of the trip pilot valve could prevent bleeding of the stop valve shaft balancing steam and prevent free movement of the stop valve. This could delay closure of the stop valve when a trip signal comes in. Therefore addition of the isolation valve could result in a condition that might affect the stop valve operation as described in the SAR. However, as part of

the design change, this new valve will be administratively controlled to ensure that the valve will always be open during normal turbine operation, except during performance of STP M-21C. (The administrative control consists of locking the valve in the open position by the use of a wire rope and lock assembly. A lamicoid sign will also be installed at the valve to identify that this valve shall be locked open during normal operation except during performance of STP M-21C.) With the valve locked open and the installation of the lamicoid sign, there is a high degree of assurance that the valve will not be tampered with during normal plant operation. The flow path for the EH fluid will be assured to remain the same as prior to the modification during normal plant operation. The new valve is sized such that it will not have additional restriction on the EH fluid flow that could slow down the stop valve closure. Positioning of the valve will be controlled by STP M-21C and will only be closed during performance of STP M-21C. This will ensure that there will always be EH fluid to open the trip pilot valve to relieve the unbalanced steam pressure to allow closure of the stop valve when the turbine mechanical overspeed trip is actuated.

Conclusion:
With the new isolation valve administratively controlled open, operation of the stop valve is assured to act as a redundant backup to the governor valve. Even if the governor valve malfunctions and does not close, the stop valve will close to prevent overspeeding the turbine. This activity does not result in an unreviewed safety question.

98-082 DCM T-9, Wind, Tornado, and Tsunami

Reference Document No.: DCP N-049395
Rev. No: 0
Reference Document Title: DCM T-9, Wind, Tornado, and Tsunami
Safety Evaluation Description:
DCM T-9 "Wind, Tornado, and Tsunami" provides design basis information for the wind, tornado, and tsunami effects on structures, systems and components (SSCs). Evaluations have been performed to analyze the effects of tornadoes on plant SSCs and these evaluations are documented in Sections 4.3.5 and 4.3.6 of DCM T-9. These sections of DCM T-9 are being revised to update information concerning safety-related equipment potentially exposed to the effects of a tornado and to provide consistency with the FSAR Update. The changes are based on the results of a tornado evaluation which was done in response to AR A0358893/QEQ0011564. This AR noted that the FSAR Update needed revising. The FSAR texts and tables associated with the DCPP tornado analysis (FSAR Section 3.3.2.3) have been revised and the FSAR Update change request and LBIE were approved by the PSRC on 5/23/97.

Safety Evaluation Summary:
There are several conditions (accidents) addressed in DCM T-9 and this DCM change which could be caused by a tornado and which are already analyzed in the DCPP accident analysis described in FSAR Update Chapter 15:

- Spurious Safety Injection Signal caused by damage to main steam line pressure transmitters (DCM T-9 paragraph 4.3.5.6.5)
- Accidental Depressurization of the Main Steam System caused by damage to the main steam line safety valve spindle (DCM T-9 paragraph 4.3.5.6.7).
- Inadvertent closing of a main steam isolation valve caused by damage to the valve's pneumatic supply (DCM T-9 paragraph 4.3.5.6.7).
- Break in the main steam line bypass line caused by a tornado missile (DCM T-9 paragraph 4.3.5.6.7).

The probability of these accidents is dependent upon the likelihood of a postulated tornado of sufficient magnitude to cause the identified impairments. Since this DCM change does not change the frequency or probability of the occurrence of a tornado, the probability of occurrence of these accidents previously evaluated in the SAR is not changed.

This change is only to update DCM T-9 to current conditions and to consolidate tornado design basis information in one document. The change does not cause any change in the plant configuration, and has no effect on the initiation of any plant accidents. The frequency and probability of a tornado has not been changed. Therefore, the probability of occurrence of an accident previously evaluated in the SAR has not been increased.

The radiological consequences from the potential effects of a tornado have been evaluated and as stated in FSAR Update Section 3.3.2.3.1, the loss of Design Class I equipment as the result of a tornado does not produce radioactive releases greater than those resulting from the loss of offsite power (LOOP) event, which is a Condition II fault described in FSAR Update Section 15.2.9. The newly evaluated conditions or the revised existing descriptions in this DCM change do not create any situations which cause the radiological consequences to exceed those of a LOOP condition.

Conclusion:
Although additional equipment was added to the DCM tornado analysis to better reflect current plant conditions, the same criteria and assumptions used for the previous tornado analysis were used for this update. No plant modifications have been made and there has been no increase in the frequency or severity of the potential tornado. As stated previously, damage to this equipment was evaluated by showing that the plant could be placed in

a safe shutdown condition after the damage was sustained. This activity does not result in an unreviewed safety question.

98-083 Exercising of Containment Atmosphere Sample (Post-LOCA) Valves

Reference Document No.: STP V-3T4
Rev. No: 10
Reference Document Title: Exercising of Containment Atmosphere Sample (Post-LOCA) Valves

Safety Evaluation Description:
Reference AR A0460656. A question was raised regarding our IST Program Plan Cold Shutdown (CSD) Justification V-C525. The justification lists several valves, mostly sample valves for Post-LOCA use, which are required by TS.4.6.1.1.a to remain closed when containment integrity is required, except under strict administrative control. See also the Bases for TS.3/4.6.3 The question was asked by we don't perform quarterly testing by implementing the strict administrative controls? This LBIE is being done because that might be a change to a method of performing a function or a change in administrative control as described in an SER

Safety Evaluation Summary:
No accident previously evaluated I the SAR will have its probability of occurrence increased by operating these valves in Mode 1-4.

No Safety Related (or important to safety) equipment is directly impacted by this sample system. The lines are designed for Post-LOCA operation and one valve is operated at a time to maintain containment integrity. Operation of these valves (having them open) for a short period of time for testing while utilizing strict administrative controls (as defined by T.S. Bases 3/4.6.3) is already allowed by our operating license.

Conclusion:
Even for the most limiting LOCA evaluated in the SAR, it is expected that these sample valves will be used Post-LOCA to assess plant conditions. Since the Tech Spec strict administrative controls will be used to ensure rapid closure of any open valve in an accident, the short time a valve is open is bounded by the assumed sample system usage Post-LOCA. Further, the test operates only one valve at a time, so containment integrity is maintained by the second valve. This activity does not result in an unreviewed safety question.

98-085 ECCS Pressure Reducing Orifices

Reference Document No.: N-049286
Rev. No: 0
Reference Document Title: ECCS Pressure Reducing Orifices
Safety Evaluation Description:
The addition of pressure reducing orifice assemblies and trimming orifices on ECCS cold leg injection lines is a change to the facility design which may affect the ECCS performance as required by Technical Specification and described in the UFSAR. Therefore, a 50.59 Safety Evaluation must be performed.

The proposed sizes of the pressure reducing orifice assembly, throttle valves and trimming orifice has been tested at Balch hydro powerhouse. All the design parameters such as flow and differential pressure across the components have been tested at the Balch Facility. It should be noted that large margin is reserved with the throttle valves so that the adjustment can be made during the STP V-15 ECCS Flow Balance Test to maintain the flow rate within the acceptable criteria as defined by T.S. 4.5.2.h. The orifice sizing calculation confirmed that the results for this design change modification is acceptable. Additionally, the same modification had been made to Unit 2 with excellent results.

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 CCPs and SIPs as well as to avoid potential ECCS flow blockage during the sump recirculation phases. Therefore, this ECCS injection line modification provides greater assurance that the ECCS will perform as required for accident mitigation. Since ECCS is not considered an accident initiator, this modification does not affect the probability of occurrence of any accident previously evaluated in the SAR.

The safety function of the ECCS is to deliver flow to reactor vessel for core cooling and provide additional shutdown capability following an accident. In DCCP's accident analysis, the ECCS performance is evaluated by using the minimum and maximum pump curves coupled with the maximum and minimum system resistance which results in the minimum and maximum ECCS injection profiles. The addition of the various components in the ECCS will continue to meet the SAR analyzed ECCS flow requirement with assurance in preventing pump runout of CCPs and SIPs.

The potential flow blockage during the sump recirculation phases will be avoided since the throttling valve settings will allow the same or greater opening than the containment recirculation sump screen opening (defined as

the nominal x-y dimension of 0.125" which is based on the ECCS maximum pump curve as stated in the calculation STA-046). Therefore, the consequences of any accidents previously evaluated in the SAR are not increased.

Conclusion:
The performance of this modification has been demonstrated by actual now tests, including a long term flow test, which verified that a consistent system resistance in the injection line could be maintained. This activity does not result in an unreviewed safety question.

98-086 Change of Test Frequency for Main Steam Safety Valves

Reference Document No.: PG&E Letter DCL-98-086
Rev. No: NA
Reference Document Title: Change of Test Frequency for Main Steam Safety Valves

Safety Evaluation Description:
PG&E is modifying the Main Steam Safety Valve (MSSV) augmented text plan previously committed to by PG&E letter DCL-97093. This schedule change moved the June 1998 testing to July 1998 in order to obtain data at 120 days following return to Mode 1. The need to revise this commitment results form the later than expected finish of 2R8 Refueling Outage. STP M-77B includes validation testing to assure MSSV performance is acceptable without Mode 1 testing after shutdown. Test plan includes Mode 1 testing following 2R8 on 2 of 4 leads (10 Valves) 120 days after reaching Mode 1. The remaining 10 valves in Unit 2 will be tested at 300 days after reaching Mode 1. This will provide comparative data to evaluate the expected improved MSSV performance of the Inconel X-750 discs. Previous successful test data was obtained on Unit 1 following 1R8 at the 90 day mark after reaching Mode 1. The validation plan also provides incrementally extended duriations between tests to assure long-term disc performance. Durations will be modified as data indicates appropriate changes. Minimizing testing is desirable to reduce seat leakage.

Safety Evaluation Summary:
The change in the MSSV testing schedule potentially affects the performance of the safety valves in response to an accident, but does not increase the probability of the accident occurring.

The test schedule change is necessary for the next test plan iteration. The X-750 disc performance validation program is aimed at improving the reliability of the MSSVs and therefore reducing the probability of occurrence of equipment malfunction.

Conclusion:
The purpose of the new schedule is to obtain valuable test data to support the ultimate elimination of augmented testing. Eliminating augmented testing enhances valve seat leakage performance. This activity does not result in an unreviewed safety question.

98-090 Glass Pipe Penetrants

Reference Document No.: FHARE_111
Rev. No: 0
Reference Document Title: Glass Pipe Penetrants
Safety Evaluation Description:
This LBIE is being generated to document the acceptability of having unrated penetration seals around glass pipe penetrants in rated fire barriers located in fire areas/zones 4-A, 3-C, and 3-J-3. FHARE 111 is attached to this LBIE and provides the technical basis of the acceptability of the unrated penetration seals. FHARE 111 is part of the Fire Protection Program, and evaluates the adequacy of having unrated seals around glass pipe penetrants in rated fire barriers. The glass pipe penetrants have not been used in any penetration seal fire tests, and therefore the seals can not be rated. In addition, this LBIE is being performed to document the acceptability of changing the text in FSAR Appendix 9.5A which describes the 3-J-3 and 4-A fire areas. The FSAR revision will indicate the presence of unrated seals in barrier separating fire zones 3-J-3 and 4-A.

Safety Evaluation Summary:
The glass pipe penetrations seal assemblies were installed during the construction pasha at DCPP., The installed foam seal design satisfies the requirement for a three hour rated seal. The glass pipe penetrants have high melting point and a much lower thermal conductivity than the tested configuration which utilized steel penetrants. A fire originating in fire zone 3-J-3 or 3-C would not be a severe fire based on the existing combustible loading, and would present a significant challenge to the penetration seals. The vaporizing of any hazardous liquids contained in the glass pipes would pose a problem because it would be prevented from passing into fire area 4-A by the trap in the line. A fire originating in fire area 4-A would also not challenge the seals containing the glass pipes because the penetrations are located in the floor slab and inside closed metal cabinets. The heat and flames from a fire in fire area 4-A would tend to travel up towards the ceiling and not down through the metal cabinet to the penetration seal.
Therefore, even thought the fire rating of these penetration seals can not be established, it can be assumed with a high degree of confidence that the fire rating of the penetration seals discussed in this evaluation will be able to perform its intended function during a 3 hour fire.

Based on the above evaluation, the two penetration seals containing glass pipes are sufficient to protect safe shutdown circuitry form a fire in the areas discussed in this evaluation.

Conclusion:
In addition, the other fire protection features discussed above, i.e., automatic detection and suppression are considered adequate to ensure that ability to detect and control a fire, thereby limiting the fire damage to one fore area. This, safe shutdown is not adversely affected, and safety related equipment are isolated from unacceptable fire hazards. This evaluation does not result in an unreviewed safety question.

98-093 Control of Combustible Materials

Reference Document No.: OM8.ID4
Rev. No: 6
Reference Document Title: Control of Combustible Materials
Safety Evaluation Description:
This procedure is being revised to allow the storage of flammable liquids in a designated laydown area when unattended. While this is allowed by the NFPA it constitutes a change in the methodology presently being implemented at DCPP. Presently all flammable liquids are required to be stored in flammable liquid storage cabinets when not in use.

Safety Evaluation Summary:
Procedure OM8.ID4, "Control of Flammable and Combustible Materials" is required to comply with commitments as documented in Section 9.5 of the FSAR which states "Storage of flammable liquids meets the intent of NFPA 30, Flammable and Combustible Liquids Code." The change to OM8.ID4 will allow the storage of flammable liquids in a designated laydown area in lieu of a flammable storage cabinet when not in use, i.e., during breaks. This methodology is in keeping with the requirements of NFPA 30. As noted in Section 4-5.1.4, of NFPA 30, "Liquids used for building maintenance, painting, or other similar infrequent maintenance purposes shall be permitted to be stored temporarily in closed containers outside of storage cabinets or inside liquid storage areas, if limited to an amount that does not exceed a l 0-day supply at anticipated rates of use." The quantity of flammable liquids used in an area will continue to be evaluated on transient combustible permits, in accordance with the requirements outlined in OM8.ID4, and the modifications being proposed in this revision will neither increase nor decrease the quantity of transient combustibles being evaluated. As outlined in the procedure flammable liquids will continue to be stored in flammable storage cabinets at the end of every shift. As such, the flexibility to leave flammable liquids in a designated laydown in lieu of placing them in flammable storage cabinets when not in use will not impact DCPP's ability to

achieve and maintain safe shutdown as described in the FSAR.

The Appendix R Safe Shutdown analysis as described in the FSAR and further described in calculation M-928, credits spatial separation, fire rated barriers and fire protection features to ensure system redundancy is maintained.

Conclusion:
The quantity of flammable liquids will continue to be evaluated on transient combustible permits, in accordance with the requirements outlined in OM8.ID4, and the modifications being proposed in this revision will neither increase nor decrease the quantity of transient combustibles being evaluated. As outlined in OM8.ID4 flammable liquids will continue to be stored in flammable storage cabinets at the end of every shift. This activity does not result in an unreviewed safety question.

98-095 ASW Phase II Cathodic Protection

Reference Document No.: C-049185
Rev. No: 0
Reference Document Title: ASW Phase II Cathodic Protection
Safety Evaluation Description:
This design installs Phase 11 of the cathodic protection (CP) system for the existing Class I ASW supply pipelines between the vacuum breaker vaults and the Turbine Building. The work is not Safety-related. Phase I of the CP system for the ASW supply pipes located between the Intake Structure and the vacuum breaker vaults was installed by DCP C49169. This DCP will complete the CP system for the ASW supply pipes and will test the entire system installed under both Phases I and II.

FSAR Section 9.2.7-2.4 and DCMs S- I 7B (Ref. AT DCNIC AR A0403565 (Rev. 4), and S-68 (Ref. AT DCNIC AR A0463086) will be revised to reflect installation of the Phase II CP system: changes to these documents which related to Phase I CP installation were previously incorporated by DCPs C-49169 and C-49207.

Safety Evaluation Summary:
The work involves installation of a buried CP system. Most of the components of this system are not near the ASW pipes.

Installation of the CP system and connection of electrical leads, for future condition monitoring or CP protection of the pipes to existing leads do not affect the operation or pressure boundaries of the ASW system. The ASW pipes will remain buried, structurally adequate and seismically restrained during the work. The modification does not affect any system interfaces and

has no impact on ASW system capability to perform its function or change its operation.

The ASW system, in general, is not identified as the cause of any SAR analyzed accidents.

The ASW pipes are required to mitigate consequences of SAR analyzed accidents. This modification does not change any system interfaces, has no impact on the ASW system to perform its function and does not change its operation. The work activities will be performed per established criteria and standards, plant procedures, and detailed instructions provided to preclude adverse impact on the ASW pipelines and their pressure boundaries,

Conclusion:
The work does not modify the ASW pipes nor affect the operation of any equipment. The structural integrity and seismic support of the pipes will not be affected nor degraded. Therefore, the work done by this DCP does not create a possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. This activity does not result in an unreviewed safety question.

98-096 Undampered Duct Penetrations 4-B, TB-7/19-E, TB-7/19A

Reference Document No.: FHARE 58
Rev. No: 3
Reference Document Title: Undampered Duct Penetrations 4-B, TB-7/19-E, TB-7/19A

Safety Evaluation Description:
This LBIE is being generated to document the acceptability of having undampered duct penetrations and a unsealed duct penetration in rated fire barriers located in fire areas/zones 4-B, TB-7/19-E, and TB-7/19-A. FHARE 58 provides the technical basis of the acceptability of the unsealed penetration and undampered duct penetrations. FHARE 58 is part of the Fire Protection Program, and, this LBIE also is being performed to document the acceptability of changing the text in FSAR Appendix 9.5A. The FSAR revision will indicate the presence of the unsealed penetration in the barrier separating fire areas/zones TB-7/19-E and 4-B.

Safety Evaluation Summary:
FHARE 58 evaluates the as-built condition against the effects of a postulated fire. No new fire hazards are being introduced, and the existing fire protection features are not being modified or changed. This FHARE concludes this condition will not adversely impact the DCPP fire protection program, and therefore, the probability of occurrence of an accident

previously evaluated in the SAR will not be increased.

FHARE 58 involves the evaluation of an as-built configuration involving undampered duct penetrations and a unsealed duct penetration in 3-hr rated barriers, and updating the FHARE combustible loading discussion to reflect current combustible loading in the affected fire areas. As discussed in the Fire Protection Section of this LBIE, these changes will not affect the ability to achieve and maintain safe shutdown. The combustible loading is low in the affected areas, and the existing fire protection features are adequate for the low hazards. Because the FHARE evaluates the effects of a postulated fire, the consequences of a design basis accident will not be increased. FHARE 58 does not impact other types of design basis accident described in the FSAR, and therefore the consequences will not be affected.

Conclusion:
FHARE 58 evaluates an as-built configuration against the effects of a fire. The FHARE does not affect the normal function of equipment important to safety, and there is no impact on the ability to achieve and maintain safe shutdown. This evaluation does not result in an unreviewed safety question.

98-097 Abandon Incore Instrument Room Chiller System in Place

Reference Document No.: DCP M-049416
Rev. No: 0
Reference Document Title: Abandon Incore Instrument Room Chiller System in Place

Safety Evaluation Description:
The incore instrument room chiller system is shown in a number of licensing and design documents. Abandoning the incore instrument room chiller system involves revision to these documents, including but not limited to the FSAR Update, the Q-List, DCM's S-16 and S-23A and a number of design drawings. Since the system will not be used, system operations as described in the SAR will also need to be updated. For these reasons, an LBIE needs to be performed to evaluate the impact of this design change on the licensing bases of the plant.

Safety Evaluation Summary:
The incore instrument room chiller system is a nonsafety-related ventilation system which will not cause the occurrence of an accident. This system has not been in use since plant was started up. Abandoning this system will not increase the probability of an accident previously evaluated in the SAR.

The incore instrument room chiller system was originally designed to provide cooling to the containment incore instrument room to allow access for maintenance on the incore instruments. However, due to high radiation dose

rates in the room when the plant is operating, this room is not accessible. When the plant is shutdown, the temperature in the room is relatively low such that operation of the system is not necessary. Therefore, the system has not been used ever since the plant has started. Other than to maintain containment integrity, this system is not required to support plant operation and mitigation of any accident previously evaluated in the SAR. The existing containment isolation valves will remain as active valves and will be closed during plant operation to preserve the containment isolation function. Therefore, abandoning the system will not increase the consequences of an accident previously evaluated in the SAR.

Conclusion:
The incore instrument room chiller system is not designed nor required to support the operation of any plant equipment important to safety. This activity does not result in an unreviewed safety question.

98-102 Circulating Water Pump (CWP) Trip on High Condenser Pit Level

Reference Document No.: ECG 17.5
Rev. No: 0
Reference Document Title: Circulating Water Pump (CWP) Trip on High Condenser Pit Level

Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 evaluation is required for all ECG changes. This ECG creates administrative controls for the CWP trip on high condenser pit level.

Safety Evaluation Summary:
Proposed ECG 17.5 creates administrative controls for the CWP trip on high condenser pit water level. The FSAR credits the CWP trip for eliminating the need for operator action to protect the safety-related EDGs from circulating water system leakage.

The CWP trip does not initiate a flooding event but mitigates such an event by tripping the CWP on high condenser pit water level to preclude flooding the EDGs. The controls established by this ECG are consistent with the current design and licensing bases. They provide additional assurance that the CWP trip will remain operable and able to perform its required function when needed to mitigate a flooding event.

Conclusion:
Placing the CWP trip on high condenser pit level under the control of an ECG will provide additional assurance that the CWP trip will remain operable and able to perform its required function when needed to mitigate a flooding event. There is no change to the design, function, or operation of the CWP trip. This activity does not result in an unreviewed safety question.

98-103 TS 1-7 Maintenance TO 9805

Reference Document No.: TS 1-7 Maintenance TO 9805
Rev. No: 0
Reference Document Title: TS 1-7 Replacement
Safety Evaluation Description:

This LBIE is being written to support the TS 1-7 replacement work. The scope of this LBIE is the operating procedure that governs the work, TO-9805, the installation of the temporary screen and the temporary relocation of the screen delta-level sensor.

This LBIE evaluates a procedure that installs a temporary screen to be used in place of the existing auxiliary saltwater system traveling water screen. It also re routes the high side sensing point of LT-240C, ASW screen differential pressure (level) sensor to a point in front of the temporary screen. The temporary filtering screen differs from the existing screen in that it is stationary and does not have a cleaning system, thus manual cleaning on the screen will be required. Also, the level transmitter function is described in FSAR section 9.2.7.2.3.

Safety Evaluation Summary:

The existing TWS 1-7 is Design Class II. The temporary screen will be designed with similar criteria as TWS 1-7 for filtering capability (mesh size and flow area). The structural integrity and loads due to flow and debris collection have been evaluated and are reflected in operating limits in this procedure. The temporary screen will be constructed of wire mesh fastened to a steel frame. The mesh openings are approximately 3/8". The screening area of the temporary screen (6'x23.75'=142.5 ft^2) is similar to that of TWS 1-7 (5'x30'=150 ft^2). The materials of construction are adequate for temporary service in saltwater; No significant corrosion damage is expected to the steel structure during its limited installation. The temporary screen has been designed with SISIP consideration.

Existing and temporary instrumentation and procedures for monitoring ocean conditions, debris loading and ASW suction bay level will be utilized to ensure safe-operation of the ASW system. The screen will be maintained by manual cleaning to eliminate impact on the flow available to the ASW and screen wash pumps. In the event that the screen gets loaded with debris, indication will be provided to the control room operators will be provided through the Plan Process Computer by the modified LT-240C. In the event of a catastrophic failure of the screen, the unit crosstie may be used to provide cooling flow to the standby CCW heat exchanger. Crosstie capability between the circulating water and ASW forebays can also be used as a supplementary source of supply water to the Unit 1 ASW forebays per existing procedures and instructions in this procedure. These design features,

maintenance activities and enhanced operator instruction provide adequate means to identify a degraded condition and take appropriate action.

Conclusion:
Existing design features and this procedure address handling the situation of a loss of ASW, with special emphasis on operator response to indications of a failure of the temporary stationary screen. The ability to provide an alternate flow path through the demusseling line as allowed per FSAR section 9.2.7.2.3 or to establish CCW heat exchanger flow from the other unit through FCV-601 will not be impacted by this procedure. This activity does not result in an unreviewed safety question.

98-104 Fuel Oil Transfer Pump Vault Floor Drain Check Valves

Reference Document No.: ECG 21.1
Rev. No: 0
Reference Document Title: Fuel Oil Transfer Pump Vault Floor Drain Check Valves

Safety Evaluation Description:
This LBIE evaluates the acceptability of placing administrative controls for the fuel oil transfer pump vault floor drain check valves under the controls of the ECG program specified by Administrative Procedure OP1.DC16, "Control of Plant Equipment Not Required by the Technical Specifications."

OP1.DC16 requires a 10 CFR 50.59 Safety Evaluation to be performed for all ECGs.

Safety Evaluation Summary:
The diesel fuel oil storage and transfer system is provided to supply fuel oil to the emergency diesel generators (EDGs) for Units 1 and 2. The design incorporates sufficient redundancy so that a malfunction or failure of either an active or a passive component will not impair the ability of the system to supply fuel oil. In order to support the design, fuel oil transfer pump vault floor drain check valves were installed to provide back flood protection for the fuel oil transfer pump vaults in the event of a blockage in the drain line concurrent with flooding in another vault sharing the same drain header.

The proposed ECG uses an existing surveillance procedure to annually verify the operability of the fuel oil transfer pump vault floor drain check valves. The ECG is an administrative control. The action to verify the operability of the fuel oil transfer pump vault floor drain check valves is a valve inspection in accordance with an approved plant procedure. The ECG and valve inspection are not event initiators because they do not introduce a flooding source to the vaults. They have no impact on the probability of a flooding event occurring.

Verifying the operability of the fuel oil transfer pump vault floor drain check valves ensures they are capable of fulfilling their specified function as described in the SAR. The controls established by this ECG are consistent with the current design and licensing bases. Also, there is no change to the plant configuration or procedures.

The ECG and inspection of the fuel oil transfer pump vault floor drain check valves do not adversely impact on the ability of plant systems or radiological barriers to mitigate the radiological consequences of design basis events evaluated in the SAR.

Conclusion:
The action to verify the operability of the fuel oil transfer pump vault floor drain check valves is a valve inspection in accordance with an approved plant procedure. It does not create a failure mode not previously considered in the SAR. This activity does not result in an unreviewed safety question.

98-105 Portable Fuel Oil Transfer Pump

Reference Document No.: ECG 21.2
Rev. No: 0
Reference Document Title: Portable Fuel Oil Transfer Pump
Safety Evaluation Description:
This LBIE evaluates the acceptability of placing administrative controls for the Portable Fuel Oil Transfer Pump 0-1 under the controls of the ECG program specified by Administrative Procedure OP1.DC16, "Control of Plant Equipment Not Required by the Technical Specifications."
OP1.DC16 requires a 10 CFR 50.59 Safety Evaluation to be performed for all ECGs.

Safety Evaluation Summary:
The portable fuel oil transfer pump is a backup system to the diesel fuel oil storage and transfer system. The operation of the portable fuel oil transfer pump is a recovery action to restore/maintain the EDG function under postulated beyond design basis events. Its availability enhances the overall reliability of the diesel fuel oil storage and transfer system and the EDGs.

The proposed administrative control does not impact the probability of an accident since the portable fuel oil transfer pump is used for mitigating a beyond design basis event.

The ECG uses a surveillance procedure to verify the availability and operability of the portable fuel oil transfer pump. The ECG is an administrative control. The ECG and operation of the portable fuel oil transfer pump in accordance with the surveillance procedure are not event initiators because they do not

degrade or alter the function of the diesel fuel oil storage and transfer system.

The portable fuel oil transfer pump is part of the Diablo Canyon licensing basis because SSER 34 evaluated the final PRA-1988 model submitted as part of the Long Term Seismic Program. The PRA-1988 model and the current Diablo Canyon PRA model have recovery actions for the fuel oil transfer system, which include manually aligning a portable fuel oil transfer pump and manipulating the valves in the EDG rooms when both normal fuel oil transfer supply trains are unavailable. This condition would be a beyond design basis event because the diesel fuel oil storage and transfer system design incorporates sufficient redundancy so that a malfunction or failure of either an active or a passive component will not impair the ability of the system to supply fuel oil.

Conclusion:
The availability and operation of the portable fuel oil transfer pump enhances the overall reliability of the diesel fuel oil storage and transfer system and the EDGs. This activity does not result in an unreviewed safety question.

98-106 CO_2 System

Reference Document No.: ECG 18.5
Rev. No: 4
Reference Document Title: CO_2 System
Safety Evaluation Description:
This LBIE evaluates the acceptability of revising ECG 18.5, "CO_2 System," to clarify the intent of ECG 18.5.1 Action B. Action B provides an acceptable method of removing the cable spreading room CO_2 system from service when plant technicians were actively working in the area. Action B is intended to apply only when MP I-2.35-1, "Cable Spreading Room Cardox System Removal/Return to Service," is in effect. This is an administrative change to clarify when it is acceptable to enter ECG 18.5.1 Action B. This clarification is consistent with the 10 CFR 50.59 evaluation that was done for revision 1 to ECG 18.5, dated February 14, 1994, which added Action B to ECG 18.5.1. Per Procedure OP1.DC16, a 10 CFR 50.59 evaluation is required for all ECG changes.

Safety Evaluation Summary:
Proposed revision 4 to ECG 18.5, "CO_2 System," clarifies the intent of ECG 18.5.1 Action B by having it refer to MP I-2.35-1, "Cable Spreading Room Cardox System Removal/Return to Service." ECG 18.5.1 Action B was added to ECG 18.5 in revision 1, dated February 14, 1994, to provide an acceptable method of removing the cable spreading room CO_2 system from service when plant technicians were working in the area. Action B only applies when MP I-2.35-1 is in effect. However Action B does not reference

MP I-2.35-1, and there has been confusion as to when Action B applies.

Neither proposed revision 4, nor the cable spreading room (CSR) CO_2 system itself, is an accident initiator. The CO_2 system is designed to mitigate the consequences of a fire. Disabling the CO_2 system does not provide an additional ignition source, nor preclude a fire from starting.

FSAR, Section 9.5A, "Fire Hazards Analysis," assumes loss of all equipment in the CSR and states that safe shutdown capabilities located outside the CSR remain available to achieve and maintain a safe shutdown. Therefore automatic actuation of the CO_2 system for the CSR is not required for safe shutdown, but is provided to limit the damage which could be caused by a fire.

Conclusion:
The proposed revision results in no change to the design, function, or operation of the CO_2 system, beyond the change approved in revision 1 (the addition of Action B to ECG 18.5.1). The ECG 18.5 controls remain in compliance with the current design and licensing bases. This activity does not result in an unreviewed safety question.

98-107 FSAR Update, Section 3.8.1.5.3 Factors of Safety; Table 3.8-5B Containment and Auxiliary Buildings Comparison of Displacements and Separations

Reference Document No.: FSAR Update Section 3.8.1.5.3/Table 3.8-5B
Rev. No: 11A
Reference Document Title: 3.8.1.5.3 Factors of Safety; Table 3.8-5B Containment and Auxiliary Buildings Comparison of Displacements and Separations

Safety Evaluation Description:
Changes are made to FSAR Update Section 3.8.1.5.3 and Table 3.8-5B to clarify the method of calculating the factor of safety and in some cases revise the factors of safety values for gaps that exists between structures. The changes thus involve the facility design as described in the FSAR but was not described in the FSAR Update. Therefore Question 1.a of the screen is answered YES.

Safety Evaluation Summary:
The proposed changes are to FSAR Update statements and entries regarding existence and values of gaps, and the method for calculation the factors of safety. The probabilities of seismic events or plant accidents are not dependent on either parameter. The probability of an accident previously evaluated is thus not impacted by these changes.

Since the proposed changes are being made only to the existing gaps and to calculations for factors of safety, they do not alter the seismic qualification of structures, systems, or components (SSCs) in the event of a DDE or HE. In a seismic event, the SSCs will continue to perform their safety functions regardless of the gaps and the calculation method for safety factors because there is no physical contact between the structures. The revised values maintain gaps with sufficient factors of safety against actual contact in a seismic event, so that there still is no contact. Therefore, there is no increase in the consequences.

Conclusion:
Changing the width of gaps or methods for calculating factor of safety does not alter plant operation. No safety systems e affected and the design basis is not being revised. Accidents that had been previously analyzed, including the occurrence of seismic events, do not vary with the width of gaps or the calculation method for factors of safety, as long as sufficient gaps and factors of safety remain to preclude contact. Therefore, should an analyzed accident occur, the nature of the accident occur, the nature of the accident would be the same regardless of the width of gaps or the calculation method. This activity does not result in an unreviewed safety question.

98-110 FSAR Revision: Thermal And Hydroelectric Plants Underfrequency Setpoint Changes

Reference Document No.: FSAR Section 8.2.2.2
Rev. No: 11
Reference Document Title: FSAR Revision: Thermal And Hydroelectric Plants Underfrequency Setpoint Changes

Safety Evaluation Description:
PG&E Electric Transmission Planning and Operation (ETPO) has revised the Hydro generating underfrequency setpoints based on WSCC guidelines. The Hydro generation underfrequency trip setpoint description is in FSAR section 8.2.2.2 "Operation during severe system disturbances". The revised setpoints are contained in PG&E Generation Transmission System (GTS) Standard ET-SO-S0426.

Safety Evaluation Summary:
PG&E's hydroelectric underfrequency setpoint is coordinated through PG&E's ETPO, the WSCC, and NAERC to increase transmission system reliability and minimize equipment damage. The previous hydroelectric generation underfrequency setpoint was designed so that hydroelectric generation would be the last units on the grid to supply power. This premise is still valid with the new setpoint. The thermal generation units (DCPP included) would

continue be tripped before the hydroelectric generation. Hydroelectric units connected to the transmission system have broad capability to operate during underfrequency conditions and most hydroelectric units do not have underfrequency control. The DCPP underfrequency setpoints have not changed. Faulted or degraded conditions on the transmission system (e.g. loss of external electrical load and loss of off-site power) would result in onsite power transferring to the diesel generators. There will not be increased challenges to the electrical systems important to safety, nor will there be additional reliance on the emergency diesel generators or increased diesel starts. Therefore, changing the Hydroelectric generation underfrequency setpoint does not increase the probability of occurrence of any accident previously evaluated in the SAR.

Should there be a severe degraded grid condition or loss of offsite power, the onsite power source (diesel generators) will provide power to the safety related loads. The FSAR already includes an evaluation for Station Blackout should there be a total loss of onsite and offsite power.

Conclusion:
This change to the FSAR does not result in a change in operation, maintenance, physical change or procedural change that could create the possibility of an accident that is of a different type than previously evaluated in the SAR. This activity does not result in an unreviewed safety question.

98-111 Alternate Power Source To SFP Pumps

Reference Document No.: DCP E-049381
Rev. No: 0
Reference Document Title: Alternate Power Source To SFP Pumps
Safety Evaluation Description:
This design change adds an alternate power source to each of the Spent Fuel Pool Pumps from Vital 480V Bus F via manual transfer switches. FSAR Update section 9.1.3.2.1.1 describes the power to the Spent Fuel Pool Pumps. It indicates that, even though there are no Class I E electrical loads in the spent fuel pool system, the SFP pumps are powered from a Class I E source and that during electrical bus outages, the standby/redundant pump may be temporarily aligned to an alternate Class I E source or a non-vital power source until its primary Class I E power supply is returned to service. Living FSAR Update Rev. 12 added the alternate Class IE source per DCP E-050381. The alternate power source that is being installed by this design change is from a Class I E power source and the transfer switches are being purchased and installed to meet Class I E requirements. FSAR Update Figures 08.03-06, 08.03-07 and 08.03-08 (single line diagrams 437916, 437542 and 437543) are being revised to show the alternate feeds.

Safety Evaluation Summary:
The Spent Fuel Pool Pumps do not cause or contribute to the initiation of accidents described in FSAR Chapter 15. Therefore, the probability of accident occurrence is not being increased by the addition of an alternate power source to the SFP pumps.

Providing an alternate power source to the SFP pumps from 480V Bus F will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR. Upon loss of offsite power, the SFP pumps are shed from the bus to limit loading on the emergency diesel generator. SFP pump 1-2 is shed from Bus H by a "T" signal (Phase A Containment Isolation) which is initiated by either a Safety Injection (SI) "S" signal or a manually actuated Phase A Containment Isolation Signal. This load shedding is not being changed by this design change. The 480V Bus F will also be isolated from a fault on the alternate feed by new Class 1 E breakers.

Conclusion:
Adding an alternate source to the SFP pumps from 480V Bus 1F will not create the possibility of overloading the bus. Per DCM No. S-64, the 480V system is designed to supply the maximum running load demand without exceeding the continuous ratings of the electrical equipment as shown in Calc 357A-DC. This activity does not result in an unreviewed safety question.

98-112 Abandon Incore Instrument Room Chiller System in Place

Reference Document No.: DCP M-050416
Rev. No: 0
Reference Document Title: Abandon Incore Instrument Room Chiller System in Place

Safety Evaluation Description:
The incore instrument room chiller system is shown in a number of licensing and design documents. Abandoning the incore instrument room chiller system involves revision to these documents, including but not limited to the FSAR Update, the Q-List, DCM's S-16 and S-23A and a number of design drawings. Since the system will not be used, system operations as described in the SAR will also need to be updated. For these reasons, an LBIE needs to be performed to evaluate the impact of this design change on the licensing bases of the plant.

Safety Evaluation Summary:
The incore instrument room chiller system is a nonsafety-related ventilation system which will not cause the occurrence of an accident. This system has not been in use since plant was started up. Abandoning this system will not increase the probability of an accident previously evaluated in the SAR.

The incore instrument room chiller system was originally designed to provide cooling to the containment incore instrument room to allow access for maintenance on the incore instruments. However, due to high radiation dose rates in the room when the plant is operating, this room is not accessible. When the plant is shutdown, the temperature in the room is relatively low such that operation of the system is not necessary. Therefore, the system has not been used ever since the plant has started. Other than to maintain containment integrity, this system is not required to support plant operation and mitigation of any accident previously evaluated in the SAR. The existing containment isolation valves will remain as active valves and will be closed during plant operation to preserve the containment isolation function. Therefore, abandoning the system will not increase the consequences of an accident previously evaluated in the SAR.

Conclusion:
The incore instrument room chiller system is not designed nor required to support the operation of any plant equipment important to safety. This activity does not result in an unreviewed safety question.

98-113 Containment Electrical Penetration Heat Loading

Reference Document No.: NCR N002061
Rev. No:
Reference Document Title: Containment Electrical Penetration Heat Loading

Safety Evaluation Description:
This safety evaluation addresses changes to the methodology used for determining electrical penetration heat load (watts/foot), and for developing penetration thermal damage curves. The original methodology for developing the heat load and the thermal damage curves was provided to the NRC in letters dated February 26, 1979 and September 5, 1979. These letters are referenced in Supplemental Safety Evaluations 8 & 9 (SSERs 8&9).

The old methodology calculates the penetration heat load (watts/foot) based on conductor resistance at 90 degrees C, maximum continuous loading, and a diversity factor. The calculations 363-DC and 364-DC show that the pressurizer heaters penetration heat load will exceed the penetration vendor's allowable heat load (25 watts/foot), if the old methodology is used. The new methodology, which is utilized only for the pressurizer heaters calculates heat load using rated current of the connected load. The new methodology assures that the total steady state heat load of the penetrations for the pressurizer heater circuits remains within the limits specified by the penetration supplier. The approach used is consistent with the IEEE 317-1976, section 4.2.2. The old methodology will continue to be used for all other loads because the heat load is not dependent on penetration

conductor load current and it is easier to apply.

The old methodology develops penetration conductor time-current (thermal damage) curves using an analytical approach. The curves developed were dependent on the load current and the worst case temperature (equal to LOCA peak temperature and the penetration qualification temperature). The new methodology develops the thermal damage curve for each penetration conductor by using test data at the peak LOCA temperature. The new thermal damage curves have the advantage of not being dependent on load current passing through the penetration conductors. The new thermal damage curves are more conservative than the existing curves that were based on the methodology provided in letters to NRC.

Safety Evaluation Summary:
The proposed changes described in this LBIE are associated with a change in methodology for Steady State Heat Load (watts/foot) calculation and thermal damage curve development. The methodology changes cannot affect system performance in a manner that could initiate an accident. Containment integrity is ensured because the selected protective devices will trip prior to the penetration conductors reaching their thermal capability, and the penetration canister heat dissipation is less than 25 watts/foot with an assumed ambient of 50 °C. The containment integrity is not affected. The probability of a breach of containment integrity (through the electrical penetrations) previously evaluated in the SAR will not be increased. Calculations 363-DC & 364-DC show that the Steady State Heat Load of all penetrations except the penetrations for the pressurizer heaters is less than 25watts/foot with the old methodology. The old methodology for calculating the Steady State Heat Load for all penetrations was very conservative. The new methodology uses the actual load for each connected pressurizer circuit and adds a margin of 5% to account for voltage fluctuations. The new methodology results in a maximum Steady State Heat Load of maximum 17.3 watts/foot. This value is less than the penetration vendor's acceptance criteria of 25 watts/foot. The methodology is consistent with IEEE 317-1976 and industry practice. Because the pressurizer heaters are a purely resistive load, there is no possibility of load fluctuations (except due to voltage variations that are already accounted for) or additional heat dissipation. Therefore, the containment integrity will not be challenged. The primary and backup penetration overcurrent protection for the pressurizer heater circuit is redundant and will protect the containment integrity in the event of an electrical fault. There is no basis for a mechanistic failure of these protective devices concurrent with a LOCA. Therefore the radiological consequences of a LOCA will not be increased.

Conclusion:
The calculations show that with the methodology change documented in this LBIE, the penetration will not be damaged by short term overloads and fault

currents and the heat load for pressurizer heater penetrations will be within the penetration nozzle-concrete interface design limits. This activity does not result in an unreviewed safety question.

98-114 FSAR Update - Electrical Power

Reference Document No.: FSAR Update Chapters 3 and 8
Rev. No: 11A
Reference Document Title: FSAR Update - Electrical Power
Safety Evaluation Description:
FSAR Update Section 3.1.1.1 "Definitions" is being revised to clarify the source of normal onsite power. During normal operation, station auxiliaries are provided power from the main generator. A turbine trip will initiate the tripping of the 500kV generator breakers and the main generator field.

FSAR Update Section 3.1.8.3 "Criterion 39 - Emergency Power for Engineered Safety Features (Category A)" is being revised to state that the requirements of Criterion 39 have been superseded by 1971 Criterion 17.

FSAR Update Section 8.1.4.1 "Design Basis" is being revised to properly characterize the availability of the electrical system. During auto transfer of the 4kV buses to the 230kV system, bus voltage is allowed to decay prior to transferring. Therefore, the electrical system is not designed to provide a "continuous" supply of electrical power. This is consistent with the requirements of 1971 GDC 17 and Safety Guide 32.

FSAR Update Sections 8.2.1 & 8.2.2 for the offsite power system are being revised to reflect the design requirements and the emergency operating procedure utilized for backfeeding the plant from the 500kV system.

FSAR Update Section 8.2.2 is being revised to reflect Operating Instruction O-23 transmission lines requiring compensatory measures when not operable.

Safety Evaluation Summary:

The changes to the UFSAR will not increase the probability of an accident previously evaluated in the FSAR Update. Should there be a LOCA and a loss of the 230kV system, the Class 1E electrical buses will be powered from at least two diesel generators. Station blackout analysis demonstrated that the diesel generators are sufficiently reliable that at least one diesel generator would be available for a plant shutdown. The 500kV system would still be available as a delayed source of power to the Class 1E electrical buses.

Conclusion:
The 500kV System meets the intent of 1967 GDC 39, as reworded by Westinghouse. The offsite power system does not meet the single failure

criterion, but the combinations of the offsite plus the onsite power systems meet the single failure criterion. However, as noted in SER 0, DCPP was evaluated against 1971 GDC 17 and 18, instead of 1967 GDC 39. 1971 GDCs 17 and 18 form part of the DCPP licensing basis.

DCPP meets the requirements of 1971 GDCs 17 and 18, IEEE 308-1971, and SG 32 -8/11/72. The 500kV system meets the requirements for a delayed offsite source to shut down the plant. Fuel design limits and reactor pressure boundary requirements will not be exceeded during the approximately 30 minutes it takes to backfeed from the 500kV system.

There are no physical changes as a result of this revision. This activity does not result in an unreviewed safety question.

98-115 Containment Systems - Containment Leakage

Reference Document No.: ECG 45.1
Rev. No: Recision
Reference Document Title: Containment Systems - Containment Leakage
Safety Evaluation Description:
This LBIE evaluates the acceptability of rescinding ECG 45.1, "Containment Systems - Containment Leakage." ECG 45.1 is a relocated Technical Specification (TS) approved for relocation by License Amendments (LAs) 120/118 dated February 3, 1998. LAs 120/118 also approved adding a new Surveillance Requirement (SR) 4.6.1.1.c to TS 3/4 .6.1, "Containment - Containment Integrity." SR 4.6.1.1.c requires performing containment leakage rate testing in accordance with the Containment Leakage Rate Testing Program. As a result of the addition of SR 4.6.1.1.c to TS 3/4.6.1, the requirements of ECG 45.1 are duplicated by the requirements of TS 3/4.6.1 and TS 6.8.4.j, "Containment Leakage Rate Testing Program."

Per Procedure OP1.DC16, a 10 CFR 50.59 evaluation is required for all ECG changes.

Safety Evaluation Summary:
ECG 45.1 requires containment leakage rates to be limited in accordance with the Containment Leakage Rate Testing Program. This requirement is duplicated by TS 3/4.6.1 and TS 6.8.4.j. The applicable modes and action statements for ECG 45.1 and TS 3/4.6.1 are the same. Recision of ECG 45.1 will leave in place identical requirements in TS 3/4.6.1.1 and TS 6.8.4.j. Therefore ECG 45.1 may be rescinded.

The Containment Leakage Rate Testing Program was established to ensure that the containment will perform its design function when required during an accident. The testing program, and its controls,

whether covered by TS or ECG, are not accident initiators. Leaving the test program controls in the TSs and removing them from ECGs will not result in the initiation of any accident.

Conclusion:
Recision of ECG 45.1 will eliminate duplicate requirements and will result in no change in: (1) the way the Containment Leakage Rate Testing Program is scheduled, or conducted, or (2) in any limiting conditions for operation, mode applicability, or required actions. This activity does not result in an unreviewed safety question.

98-116 Chemical and Volume Control System Principal Component Data Summary

Reference Document No.: FSAR Table 9.3-6
Rev. No: 12
Reference Document Title: Chemical and Volume Control System Principal Component Data Summary

Safety Evaluation Description:
As part of the Licensing and Design Basis alignment Effort (LDBAP), the contents of FSAR Table 9.3-6 was reviewed against other controlled documentation. This FSAR Update Change request is being submitted to rectify discrepancies that were identified in AR A0452803.

Changes are being made to FSAR Table 9.3-6 to rectify discrepancies between FSAR and other controlled documents as part of the Licensing and Design Basis Alignment Effort. Hence, this does involve a change to the facility design, function or method as described in the SAR.

Safety Evaluation Summary:
The changes are editorial clarifications which reflect actual design capabilities and do not change the manner in which the plant is operated. The changes are not relevant to equipment performance which are credited in the accident analyses or equipment safety functions. The changes made do not increase the probability of occurrence of an accident previously evaluated in the SAR.

The changes do not alter the method by which the performance of any structure, system or component could result in an increase in radiological dose to the public. The changes are not relevant to equipment performance which are credited in the accident analyses, equipment safety functions, or Tech. Spec. bases. The change would not prevent the ability to determine a failure mechanism not already being monitored under existing programs. The change does not change the manner in which the plant is operated, i.e. the dose from any accident or equipment malfunction associated with these changes is no different than previously evaluated.

Conclusion:
The changes do not result in any physical modifications or change the manner in which the plant is operated as defined in the basis of the Tech. Spec. There are no known mechanisms (instrument drift, system response time, thermal hydraulic conditions, etc.) that would degrade the performance of the equipment. This activity does not result in an unreviewed safety question.

98-119 Doors Required for HELB, HVAC, or Flood Protection

Reference Document No.: ECG 80.1
Rev. No: 0
Reference Document Title: Doors Required for HELB, HVAC, or Flood Protection

Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 evaluation is required for all ECG changes. This revision to ECG 80.1 provides additional administrative controls for control room doors. Specifically it provides an allowed outage time (AOT) of 15 minutes to return a non-functional control room door to functional status. The 15 minute AOT is based on not exceeding post-LOCA dose rates in the control room specified by General Design Criterion 19.

This revision also provides administrative controls for rendering one or more control door non-functional for more than 15 minutes. Compensatory measures are provided to assure the door(s) are monitored (security), a tailboard is held with control room personnel, and that personnel are assigned to shut the door(s) in an emergency.

Safety Evaluation Summary:
This revision to ECG 80.1 creates additional administrative controls for control room doors. The doors are not accident initiators, but are designed as security, fire, HELB and HVAC barriers to protect the control room and its occupants. An accident or event must first occur before the control room doors are required. If a control room door is not functional, it may not immediately perform its design function, but this situation would not increase the probability of accident initiation.
The proposed ECG revision establishes a 15 minute allowed outage time (AOT) for non-functional control room doors. It also provides compensatory measures to be taken if a control room door will be rendered non-functional for greater than 15 minutes. The ECG revision will not change the design, materials, nor the required function of the doors. The ECG revision should increase the probability that a given door is functional when needed during an accident.

Conclusion:
Control room doors are designed to mitigate the consequences of malfunctions, and are not malfunction initiators. An accident or event must first occur before the control room doors are required. If a control room door is not functional, it may not immediately provide required protection, but this situation would not increase the probability of a malfunction of equipment important to safety previously evaluated in the SAR. This activity does not result in an unreviewed safety question.

98-119A Fire Rated Assemblies

Reference Document No.: ECG 18.7
Rev. No: 3XPR
Reference Document Title: Fire Rated Assemblies
Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 evaluation is required for all DCG revisions.

The proposed revision to ECG 18.7 modifies and restricts use of a provision that allowed a fire watch to be considered continuous if the fire watch patrol could monitor the area every 15 minutes. The proposed revision will allow use of a single fire watch person to monitor multiple rooms or fire areas within 11 specific fire detector zones. These 11 fire detector zones were selected because the rooms/areas they encompass:

1. Are in the same detector zone,
2. are on the same elevation, and
3. do not require traversing through security doors or radiological barriers.

In addition, the proposed revision requires fire watch personnel to stay within the assigned fire detector zone and continuously transition between each room or area (i.e., the fire watch shall not remain in a room or fire area for a prolonged period of time). Each room or fire area within the assigned detector zone shall be surveyed for fire every 15 minutes or less.

All other areas of the plant covered by ECG 18.7 shall continue to have continuous fire watch dispatched to each fire area within the affected fire detector zone. The fire watch shall be continuously dedicated to the fire area.

Safety Evaluation Summary:
Fire watches are intended to discover fires in their incipient stages, and to mitigate the consequences of fires. Fire watches do not cause fires to start nor do they increase the probability of a fire occurring.

The proposed revision to ECG 18.7 modifies a provision that allowed a fire watch to be considered continuous if the patrol could monitor the area every

15 minutes. The proposed revision will restrict use of the 15 minute provision to fire detector zones listed in ECG 18.7, Table 18.7-1. The rooms and fire areas they encompass (1) are in the same fire detector zone, (2) are on the same level, and (3) do not require transitioning through security doors or radiological barriers. Also the proposed revision will require personnel assigned to these fire watches to spend approximately the same amount of time in each area and not spend prolonged periods in just one area. Modifying and restricting use of the 15 minute provision by this revision will result in increased monitoring in the affected fire detector zones. This provides conservatism in detecting a fire in its incipient stages, resulting in early action to protect safety-related equipment, mitigation of the consequences of the fire, and additional assurance that a fire will not escape the fire area.

Conclusion:
The fire rated assemblies are not addressed in any Technical Specification or in the basis for any Technical Specification. The proposed revision to ECG 18.7 will modify a provision that allowed a fire watch to be considered continuous if the patrol could monitor the area every 15 minutes. Modifying this provision, and restricting its use to selected fire detector zones, will result in increased monitoring of the affected zones and is more conservative. This activity does not result in an unreviewed safety question.

98-120 500kV Differential Line Protection

Reference Document No.: DCP E-49374 & Associated GM&C Design Change Package (8/25/98)

Rev. No: 0

Reference Document Title: 500kV Differential Line Protection

Safety Evaluation Description:
The proposed modifications necessitate a revision to the working in FSAR Update Section 8.2.1 and related Chapter 7 figures in order to reflect the new 500kV current differential protection scheme.

Safety Evaluation Summary:
FSAR Update Section 15.2.7 discusses the "Loss of External Electrical Load and/or Turbine Trip" Accident Analysis:

The proposed Unit 1 modifications improve the 500 kV "Channel A" fault line protection by providing primary, secondary and backup overcurrent protection schemes in accordance with the recommendations from PG&E's System Protection Engineering Department. The new protection schemes will trip the Unit in order to protect the integrity of the 500 kV lines in the event that an overcurrent condition exists.

The existing primary current differential protection scheme has been disabled as a result of relay malfunctions experienced outside of DCPP which could result in a spurious Unit trip. Therefore, the 500 kV lines are presently only protected via the secondary protection scheme (directional overcurrent relays). The proposed modification uses state-of-the-art, microprocessor-based current differential relays which are not prone to spurious actuation. The proposed modifications will reinstate the number of "Channel A" protection schemes to existing PG&E standard design (primary and secondary operable protection schemes) for 500 kV lines.

FSAR Update Section 8.2.1 (Offsite Power System) describes the existing 500 kV transmission line protection schemes as consisting of "a differential pilot wire relay system with directional and ground overcurrent relays". The total number of discrete protective relays associated with these existing schemes is six (two devices for the current differential scheme and four devices for the directional overcurrent relays). The new proposed protection schemes will use a total of four discrete protective relays (two current differential relays for the primary protection scheme and two current differential relays for the secondary protection scheme). Based on the smaller number of protection relays used (four as compared to six) and the fact that the new proposed relays are of modern design and not prone to spurious actuation, the proposed modifications will decrease the probability of a spurious unit trip. FSAR Update Section 8.2.1 will be revised to reflect the new protection schemes implemented by this package.

Conclusion:
In the unlikely event that the new backup protection scheme (existing directional overcurrent relays modified to include a time delay relay) is ever put in service, the number of discrete protection relays would be six (two modern current differential relays plus four existing directional overcurrent relays with a time delay). This is the same number of protective relays that are presently described in the FSAR Update with the only exception that two of the six relays are new state-of-the-art current differential relays which are not prone to spurious actuation. This configuration (new primary or secondary protection scheme plus the new backup protection scheme in service) would therefore also decrease the probability of spurious Unit trips. This activity does not result in an unreviewed safety question.

98-121 FSAR Update Revision

Reference Document No.: AR A0453386
Rev. No: N/A
Reference Document Title: FSARU Revision
Safety Evaluation Description:
This LBIE was prompted by the proposed changes to the FSARU

documented in AR A0453386. The AR documents discrepancies on the main turbine runback rate and the subsequent automatic plant action between the FSARU and other plant documents.

Subsequent review performed identified that the statement in the FSARU is incorrect. The turbine runback rate due to loss of stator cooling water should be approximately 40% load drop per min instead of 20% as stated in the FSARU and the subsequent auto plant action should be a unit trip instead of the tripping of the 500kV power circuit breakers only. However, because of the fact that the facility design and the function are different from that described in SAR, the answer to section 1a of LBIE screen has to be yes. This triggers a LBIE to document that the proposed FSARU changes will not create any unreview safety question.

The original design bases of the stator cooling water turbine runback is based on Westinghouse drawing 8301D19 (DC 663292-16). The drawing states that the runback will cause a load reduction of 3 1/3 % per each second interval and there is one pulse per 5 sec. Therefore, in one minute, there will be 60/5= 12 pulses or 12 X 3 1/3 % = 40 % load drop. Also the same drawing shows that if the runback does not start in 45 sec, it will trip the generator breaker(s) and the exciter breaker. PG&E drawings 437559 (U1) 441247(U2) clarify that this unit trip feature is actuated via the actuation of lockout relay 86G11(U1)/86G21(U2).

Safety Evaluation Summary:
The changes will correct the statement made in the FSARU concerning turbine runback rate and subsequent automatic plant action to match the original design feature documented in Westinghouse drawing 8301D19. There is no change to the existing plant design. The turbine runback feature is for the protection of the main generator which is not safety related. The correction of the runback rate from 20% to 40% is within the bounds of the Condition II "Loss of External Electrical Load and/or Turbine Trip" accident analyzed in FSAR section 15.2.7 which is more limiting in overpressurizing the primary side.

Conclusion:
The stator cooling water turbine runback feature is for the protection of the main generator and is not safety related. This feature is not described in the basis of the Technical Specification. This activity does not result in an unreviewed safety question.

98-122 Full Flow Test of Accumulator Discharge Check Valves, SI -8956A-D and RCS Cold Leg Injection 1st Off Check Valves, SI-8948A-D

Reference Document No.: STP V-18A
Rev. No: 12

Reference Document Title: Full Flow Test of Accumulator Discharge Check Valves, SI -8956A-D and RCS Cold Leg Injection 1st Off Check Valves, SI-8948A-D

Safety Evaluation Description:
The proposed test is not described in the FSAR, SER, SSERs or other licensing documents. The full flow test of the accumulator discharge check valves by blowing down the accumulators while fuel is in the vessel and the steam generator nozzle dams are installed has never been performed at DCPP. Additionally, this test does not permit RHR flow to the injection flow path affected by the test. Note that an accumulator blowdown test was performed as part of the DCPP pre-operational testing and it is described in the SAR.

Safety Evaluation Summary:
The only accidents relevant to this test are the Fuel Handling and Uncontrolled Boron Dilution Accidents. There is no increase in risk of these events occurring because the test will not be done with fuel movement in progress, existing controls over load movement over the core are not altered, and administrative controls over reactivity insertion will not be altered.
Other factors relevant to fuel damage are:

A) Fuel in the core will not be adversely affected by the maximum flow-rate achieved during this test since the lifting force due to accumulator discharge flow is an order of magnitude less than the weight of a fuel assembly.
B) Upper internals will not be lifted by the accumulator discharge flow because sufficient lifting forces cannot be generated. Additionally, the upper internals have been installed and tested without fuel loaded during the past 4 outages and no lifting has been observed.
C) Irradiation sample plugs were dislodged during 1R6 during the initial test with upper internals suspended above the vessel. Since then, this test has been successfully performed with internals fully installed,
D) During 2R6, there was a debris concern related to Accumulator 2-1 discharge line, which had been known to have a sand-like deposit (AR A0299266). Injecting through and thus flushing this line during the process of this test during 2R6 had no Chemistry or particulate impact on the RCS.

As part of the Fuel Handling Accident, fuel overheating was considered. Fuel overheating requires loss of cooling. As stated in the technical evaluation prior to these questions, the design of the S/G nozzle dams provides ample assurance that they will not fail as a result of this test. Additionally, this test does not provide a mechanism to increase the probability of RHR failure. Thus, this change will not increase the probability of a loss of water from the refueling cavity or increase the

probability of a loss of cooling.

Relevant to the Boron Dilution accident, RCS & cavity boron levels will not be appreciably affected during this test because the SI Accumulators are above 2000 PPM and will thus not result in a boron dilution accident.

Conclusion:
The possibility of a loss of RHR due to accidental nitrogen injection was reviewed and determined to not be probable. Past experience with this test supports the contention that the test can easily be performed without a problem related to reduction of boron concentration in the RCS.

Since the reactor core will have had well in excess of 57 hours cooldown time before performing this test (approx. 150 hours), reduced RHR flow will be in compliance with T.S. basis.

Two loop RHR operation will provide more than adequate cooling during the time the test is being performed. The addition of several hundred cubic feet of cold borated water from each accumulator will assist with core cooling during the period of two loop RHR operation.

Performing this test during the cavity filling phase of the Refueling sequence will actually result in filling the Reactor Cavity faster than the normal method. There is no reason this testing cannot be performed during cavity fill. Each accumulator will be injected when a predetermined cavity level is reached. This activity does not result in an unreviewed safety question.

98-123 FSAR Update Change Request

Reference Document No.: FSAR Update Change Request
Rev. No: 0
Reference Document Title: Update Tables 7.5-1, -2, -3 , & -4
Safety Evaluation Description:
This change makes corrections to FSAR Update Tables 7.5-1 "Main Control Board Indicators and/or Recorders Available to the Operators (Conditions II and III Events)", 7.5-2 "Main Control Board Indicators and/or Recorders Available to the Operators (Conditions IV Events)", Table 7.5-3 "Main Control Board Indicators and/or Recorders Available to the Operator to Monitor Significant Plant Parameters During Normal Operation", and Table 7.5-4 "Post Accident Monitoring Panel Indicators and/or Recorders Available to the Operator". The changes correct typographical errors and refer the Reader to Plant design documents for instrument uncertainties. The values in the tables are given to a level of precision sufficient to indicate their suitability for indicating Plant conditions. Values to greater precision are available in Plant design documents.

Safety Evaluation Summary:
This change is not a physical change to the Plant, but rather is a change to allow variations to instrument accuracies due to calibration or hardware changes without changing the FSAR Update. If the change to the uncertainty is within the accuracy value in the Table or is within the reading accuracy of indicator (typically ½ of a minor division), the FSAR Update does not need to be changed. These changes would not be readable by the Operator. In addition, these types of changes are controlled by existing administrative procedures and the design change process. The Table provides evidence of the suitability for the instrumentation to meet the operational requirements. Further information on precise numbers is available via design documents. This will keep the numbers from being continually revised when a calibration practice or frequency may change the indicator accuracy. There is no change on the reliability of the equipment made by this change.

Conclusion:
This modification has no impact on equipment qualification or the ability of the instrumentation to perform its intended function. There is no increase to any radiological consequences resulting from this change. This activity does not result in an unreviewed safety question.

98-124 FSAR Update, Section 7.2.3.1.4 & 7.3.2.1.5.2

Reference Document No.: A0465952 & A04650927
Rev. No: 0
Reference Document Title: FSAR Update, Section 7.2.3.1.4 & 7.3.2.1.5.2
Safety Evaluation Description:
This LBIE is being performed for a FSAR Update. The method by which a Channel Operation Test is performed per FSAR section 7.2.3.1.4 is being revised to correspond to the definition in T.S.1.3 and the standard industry definition. Currently the COT description implies that the signal input is injected as close to the protection rack as practicable rather than as close to the sensor as practicable.

FSAR section 7.3.2.1.5.2 is being revised to clarify that the basis for acceptability for ESF interlocks is receipt of proper indication upon introducing a trip. The paragraph prior to the change states that acceptability of ESFAS is based on overlapping tests performed on the initiating system and the ESFAS. One of the overlapping tests trips the Eagle 21 output and verifies proper indication. Verification of the setpoint is performed by a different overlapping test. The statement currently implies that the proper indication is verified when the setpoint is verified.

Safety Evaluation Summary:
There are no accidents previously evaluated in the SAR whose analysis is dependent upon the description of a COT or the method by which the ESF interlocks are tested. The description of a COT will correspond T.S. 1.3 definition. No procedure or administrative changes are required.

This FSAR clarifies the method by which a COT is performed and ESF interlocks are tested. T.S. definition 1.3 and plant procedures currently reflect these FSAR changes. Consequences of an accident previously evaluated in the SAR are not affected by clarifying the method of testing.

This FSAR change is required to correspond to the COT definition in T.S. 1.3 and the industry standard definition. Injecting the COT signal closer to the sensor is a more conservative testing method. The basis for acceptability of ESF interlock testing is consistent with the prior FSAR paragraph.

Conclusion:
Changing the description of testing methodology in a conservative manner to reflect the actual method of performance does not create the possibility of a different type of accident. No operation or administrative changes are required which could adversely affect the operation of any system, structure or component. This activity does not result in an unreviewed safety question.

98-125 AT FSAR ARs: Spent Fuel Pool Cooling And Clean-Up System Issues

Reference Document No.: AR NOS. A0450613/A0464088
Rev. No: n/a
Reference Document Title: AT FSAR ARs: Spent Fuel Pool Cooling And Clean-Up System Issues

Safety Evaluation Description:
The FSAR change request involves changes to FSAR Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System", and associated Table 9.1-2 for purposes of clarifying, correcting, and enhancing the existing information based on current design and operating practices.

Safety Evaluation Summary:
The changes proposed in this FSAR change request are not representative of any design feature or operating, practice that could increase the probability o occurrence of any accident previously evaluated in the SAR.

These changes do not reflect any changes in operating practices regarding design imposed pool levels or boron concentration requirements. Therefore, the consequences of any accident previously evaluated in the SAR, including a chapter 15 fuel handling accident, is not increased.

This revision does not reflect any past design changes or current operating practices that could affect the capability to maintain normal SFP water levels, or add additional make-up water to the pool. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

Conclusion:
The FSAR changes related to SFP boron concentrations, and both normal and inadvertent pool water levels are in agreement with existing licensing commitments for the safety-related Portions of the SFP system. The changes reflect current design and operating practices, and are in agreement with existing SFP related Technical Specifications including TS Section 3/4.9.11, "Water Level - Spent Fuel Pool", 3/4.9.14, "Spent Fuel Assembly Storage", and TS Design Feature 5.6.2, "Drainage". This activity does not result in an unreviewed safety question.

98-126 Revisions to DCM S-14, T-9 & FSAR Section 3.3.2.3.2.7

Reference Document No.: DCP M-049447
Rev. No: 0
Reference Document Title: Revisions to DCM S-14, T-9 & FSAR Section 3.3.2.3.2.7

Safety Evaluation Description:
The subject design change revises DCM S-14, T-9 and FSAR Section 3.3.2.3.2.7 to reflect the acceptability of the CCW System (CCWS) to perform its intended safety-related function without crediting excess flow check valve CCW-1/2-FCV-1429 and check valve CCW-1/2- 698 to mitigate the effect of a tornado damage to the Design Class II portion of the system.

The DCMs and FSAR section were revised previously (to be included in Rev. 12 of the FSAR Update) to specifically identify the excess flow check valve and the return line check valve to prevent any CCWS leakage in the event of a tornado damage to the exposed Design Class II portion of the CCWS. However, the design function of these valves were never tested, as required by DCM S-14 (Section 5.2) and an evaluation performed by calculation. M-1030 concludes that failure of these valves to function will cause CCWS leakage to be significantly less than the maximum allowable system leakage of 200 gpm, identified in the FSAR Section 9.2.2.3.3 and SSER-16.

Safety Evaluation Summary:
The CCWS is not an accident initiator and is not associated with any FSAR Chapter 6 or 15 occurrence of an accident previously evaluated. This proposed FSAR change establishes that tornado damage to the steam generator blow down tank vent condenser (SGBTVC) or associated supply

and return lines will not adversely affect the CCWS safety-related functions to mitigate the consequence of an accident. DCPP's licensing basis does not require postulation of a concurrent tornado with a design basis accident. Any impacts to Operator response requirements or system performance as a result of tornado damage does not impact previously evaluated in the SAR..

The CCWS provides the safety-related function to cool both ESF components and the containment atmosphere during design basis accidents. The system also provides cooling to support a plant (normal and abnormal) safe shutdown. This proposed FSAR change establishes that tornado damage to SGBTVC or its supply and return lines does not adversely impact the operation of the CCWS to mitigate the consequence of an accident. To establish this, a calculation (M-1030) was completed to show that any credible tornado damage to SGBTVC or the supply and return lines will result in a maximum of 90 gpm leakage from the CCWS. This is substantially less than the CCWS leakage limit of 200 gpm which is documented in FSAR Section 9.9.2.2.3.3 and SSER 16. As stated in FSAR & SSER, the CCWS surge tank volume provides operators with 20 minutes to locate and stop the leak (with leakage rate of 200 gpm), or realign the CCWS into two separate trains before the system operation becomes impaired due to water loss. This time could be extended if the normally aligned, automatic make up system (portions of which are Design Class II) functions as designed.

Conclusion:
As evaluated in FSAR Section 9.2.2.3.3 and SSER 16, the CCWS is designed to accommodate a non-mechanistic leak of 200 gpm. The CCWS surge tank volume provides operators with 20 minutes to locate and stop the leak, or realign the CCWS into two separate trains before the system operation becomes impaired due to water loss. The maximum CCWS leakage due to tornado effects is 90 gpm (Calculation M-1030) which is significantly below the maximum acceptable leakage rate. Therefore, this design change does not reduce the margin of safety, implied or defined, in the basis for any Technical Specification. This activity does not result in an unreviewed safety question.

98-127 Pyrrolidine as The pH Control Agent For Chemical Control Injection

Reference Document No.: OP F-5:II
Rev. No: 20
Reference Document Title:
Safety Evaluation Description:
This LBIE is being performed because the use of pyrrolidine as the pH control agent for chemical control injection into the secondary system is different from that described in the reference document above, and the FSAR, and applicable DCMs.

Safety Evaluation Summary:
The use of pyrrolidine should increase the life and efficiency of the steam generators by reducing condensate/feedwater system iron transport to the steam generators. It will not decrease the life or efficiency of the steam generators, based on pyrrolidine's similar chemical properties to the current pH control agent, ethanolamine (ETA). The probability of occurrence of an accident previously evaluated in the FSAR will not be increased.

The FSAR discusses a steam generator tube rupture with subsequent contamination of the secondary system with iodine. 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 evaluation more conservative.

Conclusion:
The effect of the use of pyrrolidine as the pH control agent in the secondary system has been evaluated. The potential for excessive corrosion or other adverse effects as the result of the proposed use of pyrrolidine has not been identified nor would these effects be expected to occur. It is concluded that the use of pyrrolidine will not increase corrosion rates of secondary system materials, affect the integrity of any secondary system components, or result in any degradation of primary system boundaries. This activity does not result in an unreviewed safety question.

98-128 DCPP Units 1 & 2 FSAR Update, Rev. 11

Reference Document No.: FSAR
Rev. No: 11
Reference Document Title: DCPP Units 1 & 2 FSAR Update
Safety Evaluation Description:
This FSAR change involves a change to the facility design or function as described in the FSAR. In response to open item 275/87-27-02, DCPP committed to install a computer based fire alarm system capable of providing the features specified in NFPA 72D. This new fire alarm system is backed up powered by an uninterruptable power supply or by batteries which meets the requirements of NFPA 72D. The FSAR states that fire detection and actuation systems are connected to the plant emergency power supply.

Safety Evaluation Summary:
The fire detection system is not an accident initiator or involved with any accidents described in the SAR other than a fire. The requirement for a back up power supply exists for the current design in accordance with NFPA 72D.

The plant fire alarm and detection systems are credited in mitigating the effects of a fire. These systems rely on a source of back up power to continue

operating in the event the main power source is lost. The existing Pyrotronics panels (PFoA,B,D) and the Honeywell computer are powered from a vital bus. The upgraded computer based system panels (PFACs 1-5) contain local batteries which satisfy the requirements of NFPA 72D.

This change to the FSAR is to clarify the back up power source to the upgraded portion of the alarm system.

Conclusion:
The plant fire detection and alarm system was designed and installed in accordance with the requirements of NFPA 72. NFPA 72 requires the fire detection and alarm system to be provided with a main and standby power source. The standby power source consists of dc batteries installed locally inside the panels with an automatic switching system in the event the main power source is lost.

The plant fire detection and alarm system has no safety related function, but is credited as a key element of the defense in depth fire protection program at DCPP. These panels do not initiate any fire protection systems, but do provide an early warning system that allows for immediate fire brigade response to any fire that is detected by the system. The detection and alarm system also provides a monitoring function for important system parameters like pump status, valve position, and flow switch activation. This change to the fire detection system as described in the FSAR will not adversely affect the ability to achieve and maintain safe shutdown. This activity does not result in an unreviewed safety question.

98-129 EQ Bases for LCV-113 & LCV-115

Reference Document No.: OE-98-11
Rev. No: 0
Reference Document Title: Incomplete Environmental Qualification Bases for LCV- 113 & LCV- 115

Safety Evaluation Description:
OE 98-11, R0. Describes actions taken to isolate auxiliary feedwater to the faulted steam generator during a MSLB in the GE/GW area. Shutting down AFW Pump No. 3 requires taking the control switch on Vertical Board No. 3 to the stop position. This stops AFW flow to the faulted SG which allows the SG to boil dry and terminate the MSLB event. To assure that AFW Pump No.3 will not automatically restart when the control switch is released requires holding the control switch for the AFW pump on the Vertical Board No. 3 in the "STOP/RESET" position. The shift foreman may authorize use of a temporary manual interlock to lock the AFW Pump No. 3 control switch on VB-3 in the "STOP/RESET" position until plant conditions allow the control switch to be spring returned to the normal "AUTO START" position.

Safety Evaluation Summary:
The AFWS is designed to perform its safety function under the following design bases conditions: Loss of normal feedwater, with or without offsite power. Secondary system pipe rupture, feed line or steamline. Loss of all AC power. Loss-of-coolant accident (LOCA), small and large break. LOCA concurrent with a leaking SG tube [Ref. WCAP-7451, Rev. 2, & DCM S-3B].

The overall system performance is unchanged by the temporary use of the control switch interlock. Holding the control switch in the "STOP/RESET" position for a longer duration has no effect on the switch contacts or the control circuit since the closed switch contacts are connected in series with other contacts which interrupt the pump motor trip signal, and remain open after the motor circuit breaker opens. The open contacts also electrically block the AFW Pump No. 3 auto start signals. The AFWS is designed to automatically start the Motor-driven Pumps for 1. Two-out-of-three low-low level signals in any one SG, 2. Trip of both main feedwater pumps, 3. Safety injection signal, 4. Transfer to diesel without safety injection signal, or 5. An AMSAC signal. Ref. U1 schematic 437583 rev.23 & U2 schematic 441302 rev.20. This does not cause the switch or the circuit to operate outside of previously established parameters.

The AFW Pump No. 3 control switch interlock has no effect on any radiation barrier or off site dose, for this application the temporary manual interlock is installed on the AFW Pump No.3 control switch during a break of a Main Steam Line in the GE/GW area. The purpose of installing the temporary mechanical interlock is to prevent additional injection of water into the faulted steam generator to preclude exceeding the EQ profile for equipment in the GE/GW area. Use of the temporary mechanical interlock allows a control room operator to be freed up to perform other accident mitigation functions.

Conclusion:
A review of Technical Specification paragraphs 3/4.3.2, 3/4.3.3.5, 3/4.7.1.2, 3/4.7.1.3 & Bases 3/4.7.1.2., do not identify any Technical Specification margins associated with the main control board or the AFW Pump No. 3 control switch.

Paragraph 3/4.7.1.2 Bases discusses the AFW System Safety Analysis, the motor driven pump capacity and how one motor driven AFW pump would deliver flow to a broken MFW header at the pump runout flow until the problem was detected, and flow terminated by the operator. The temporary manual interlock is used by the operator as an aid in stopping the No.3 motor driven pump when required. This activity does not result in an unreviewed safety question.

98-130 DEG 1-1: Replace Cylinder Heads

Reference Document No.: A0441510
Rev. No: 0
Reference Document Title: DEG 1-1: Replace Cylinder Heads
Safety Evaluation Description:
It is desired to replace the cylinder heads on EDG 1-1 while the plant is in Mode 1. The proposed head replacement on line is required EDG maintenance to correct problems seen on other EDGs. Although the problems with the heads have not yet resulted in the EDG being inoperable, the heads must be replaced now or in the near future. The diesel will be inoperable during the replacement of the heads. The duration of the head replacement is scheduled for approximately 4 days.

License Amendments 44 and 43 for Units 1 and 2 respectively extended the AOT for the EDGs from 72 hours to 7 days. The safety evaluation for the LAs contains a discussion stating that pre-planned preventative maintenance which "incapacitated" the engine would no longer be required. The replacement of the heads on DEG 1-1 will require the engine to be made inoperable. This LBIE will evaluate the diesel generator being inoperable for up to 168 hours (7 days) for the EDG 1-1 head replacement project.

The proposed activity would allow one EDG to be inoperable for up to 168 hours for head replacement. The EDGs are designed to mitigate an accident by supplying power to the vital 4kV busses following an event that results in a loss of offsite power. Per TS 3.8.1.1, an EDG may be inoperable for up to 7 days (168 hrs) before a shut down of the plant is required. Since EDG 1-1 is inoperable, a single failure of the other 2 EDGs is not postulated and the remaining two EDGs are capable of supplying power to at least one complete train of ECCS, assuring that the assumptions associated with all accident analyses are satisfied.

Safety Evaluation Summary:
The inoperability of one EDG for up to 168 hours will not change the fact that the EDGs are plant equipment used to mitigate an accident and cannot cause an accident.

EDG 1-1 is the emergency power source for bus H. CFCU 1-4 receives its power from bus H. CFCU 1-2, which is powered by bus F, is currently out of service. Consequently, the minimum requirement of three operable CFCU's will be met.
Therefore, the minimum number of EDGs required to assure that the minimum ESF equipment is available. If the minimum required equipment is available, no increase in consequences will occur.

A risk assessment was performed regarding performance of this work at

power consistent with Maintenance Rule requirements. The assessment indicates that the performance of the maintenance at power is not risk significant.

The station blackout (SBO) analysis for DCPP assumes the loss of all offsite power concurrent with the failure of two emergency generators on one unit. The DCPP SBO analysis considered the reliability of the EDGs to be greater or equal to 0.95. Removing one EDG for its TS AOT does not have an impact on the reliability assumed in the SBO analysis. Performing the cylinder head replacement is expected to improve EDG reliability.

Conclusion:
The mean outage time for diesel generator maintenance has been relatively constant since the DG-AOT study. As of the last PRA update, the mean outage time was 16.5 hours. The frequency has also remained relatively constant. DCPP plant data in most areas have shown a decrease in overall CDF when actual plant data has been factored into the Diablo Canyon living PRA. When the approximate 4 days for the head replacement is factored in, the overall effect is small and does not invalidate the assumptions and conclusions of the DG-AOT study. This activity does not result in an unreviewed safety question.

98-131 Generic 21-Month Cycle DCP

Reference Document No.: N-049415
Rev. No: 0
Reference Document Title: Generic 21-Month Cycle DCP
Safety Evaluation Description:

This LBIE reviews the following areas to ensure there are no unreviewed safety questions in operating Diablo Canyon Fuel Cycles for a nominal length of up to 21 months.

1. Use of 1.25 grace for any 1R9 STPs not yet extended to 24 months
2. Changes to instrument setpoint and PAM calculations which support refueling intervals up to 24 months
3. Evaluation of SG Tube integrity

Safety Evaluation Summary:
Therefore, as discussed above, sufficient technical bases exist for using the 1.25 grace per TS 4.0.2 for 1R9 tests affected by the unapproved LAR/ECG. The grace period may support a 1R9 start prior to March 6, 1999. This deferment of STPs continues to maintain the design and licensing bases for STPs, and thus does not introduce an accident initiating mechanism affecting the previously analyzed accidents in the FSAR Update. It does not alter the intent or method by which the inspections, tests, or verifications are conducted; does not alter the way any structure, system, or component

functions and does not change the manner in which the plant is operated. There are no changes to the plant hardware, safety systems or plant operating procedures except for the longer operating cycle. In summary, there are no changes which affect the probability of occurrence of an accident previously evaluated in the DCPP FSAR Update.

Steam Generator Tube Integrity: An operation assessment has been performed for 1C9 steam generator integrity (Attachment 6-1). The assessment concluded that the tubes' structural and leakage integrity requirements will be satisfied at the end of 1C9 and has no effect on the probability of occurrence of an accident analyzed in the FSAR Update.

PAM and Setpoint Calculation Changes: These calculations have been identified and revised per ICE 10 and DCM 24 as discussed in Attachment 3-1. All of the revised calculations were found to support 24-month cycles. No changes to any engineering controlled setpoints were found to be needed. Changes to acceptable as-found criteria in surveillance tests, as a result of the revised calculations, are not accident initiators but are appropriate to ensure equipment operability. The uncertainty in post-accident indications and instrument setpoints have no effect on the accident initiators analyzed in the FSAR Update. The resultant changes to EOPs are accomplished separately. Therefore, there is no effect on the probability of an accident analyzed in the FSAR Update as a result of the calculation changes.

Conclusion:
In summary, the 1R9 STP deferment associated with the TS/ECG requirements does not affect the consequences of any accident previously evaluated in the DCPP FSAR Update.

PAM and Setpoint Calculation Changes: The uncertainty in post-accident indications has been included in Emergency Operating Procedures (EOPs) in order to ensure that "true decision points", or the actual process values at which action should be taken, are met. By adjusting the EOP setpoint values the consequences of accidents are assured to be within the FSAR Update analyzed results. See the evaluations below for PAM indications which had calculation results outside of the FSAR Update Tables 7.5 values. There have been no instrument setpoint changes other than that contained in LAR 3 for the pressurizer level reactor trip from 92% to 90%. Changes to acceptable as-found criteria in surveillance tests, as a result of the revised calculations, establish appropriate criteria for equipment operability and thereby also assure that consequences are not affected by the setpoint calculation changes. This activity does not result in an unreviewed safety question.

98-136 Removal of SSPS Halon System

Reference Document No.: M-049295
Rev. No: 0
Reference Document Title: Removal of SSPS Halon System
Safety Evaluation Description:
The Halon suppression system was installed in fire area 8-G (Unit I SSPS Room) to meet commitments to Appendix A of BTP (APCSB) 9.5-1 (SSER 8). Operability of the Halon system has been questioned in Quality Evaluations (QE) Q001 1205, "SSPS Room Ceiling Tile" and QOOI 1849, "U-2 SSPS Room Halon System Test Results Outside Licensing Bases." Due to safety and environmental concerns with maintaining the Halon system, a corrective action for the QE is to remove the Halon suppression system in the SSPS rooms. Removal of this system will involve:

- Issuing an FSAR change (Section 9.5.1 and Appendices 9.5A, 9.5B and 9.5H) to delete reference to the halon system,
- Issuing new FARE 112, "Removal of the Halon System from the SSPS Rooms" to justify acceptability of changing NRC-approved commitment
- Revising FHARE 75, "1-hour Fire Barrier" to remove reference to the Halon system
- Revising FHARE 80, "Fire Dampers Installed at Variance with Manufacturer's Instructions" to remove reference to the Halon suppression system
- Deleting ECG 18.6, "Halon System," and associated surveillance procedures (STP M-19A and M-19B)
- Revising DCM S-18, "Fire Protection Systems" to delete reference to the Halon system

These documents are all part of the approved Fire Protection Program. As required by the Unit 1 Operating License Condition 2.C.(5)b, any changes to the approved program will need to be evaluated to determine if the changes will adversely affect the ability to achieve and maintain safe shutdown in the event of fire.

Safety Evaluation Summary:
Removal of the Halon suppression system does not affect the current combustible loading or ignition controls in the SSPS room. The design basis accident is a fire, and this design change will not increase the probability of occurrence of a fire in the SSPS room or in any other fire area. Removal of the Halon system will not affect operation of the SSPS computers or other safety-related equipment in the room. Therefore, the probability of occurrence of a previously analyzed accident is not increased. In addition, the revision of documents affected by this design change will not increase the probability of occurrence of previously analyzed accidents.

Removal of the Halon suppression system eliminates the automatic fire

suppression system in the SSPS room. This design change will not increase the consequences of occurrence of a fire in the SSPS room, because current analysis assumes the loss of a SSPS train as a result of a fire. Safeguards signals (e.g. automatic reactor trip, safety injection signal, containment isolation, etc.) are generated from these cabinets, but once the reactor is tripped, the equipment in the SSPS room is not required. Manual actions and redundant safe shutdown components are currently credited for mitigating the effects of a fire in this area (see FHARE 112), and provide the capability to achieve and maintain safe shutdown. The operation of the SSPS computers will not be affected by this change, and the consequences of occurrence of a previously analyzed accident is not increased.

Conclusion:
This change does not affect the performance or operation of any safety related equipment or systems. The Halon system in the SSPS room is not credited in any accident analyses, and is not used in the bases of any technical specification. This activity does not result in an unreviewed safety question.

98-138 ECG 4.1 ATWS Mitigation System Actuation Circuitry – Surveillance Requirement 4.1.2

Reference Document No.: ECG 4.1
Rev. No: 3
Reference Document Title: ECG 4.1 ATWS Mitigation System Actuation Circuitry – Surveillance Requirement 4.1.2

Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 Evaluation is required for all ECG changes. In addition, the proposed change to the ECG surveillance requirement for AMSAC alters the frequency at which the surveillance is conducted, and thus involves a change to the administrative controls over plant activities as described in the SAR. This change also alters the frequency at which the following surveillance test procedures are performed. STP I-92B, 'AMSAC External Input Calibration Check'; STP M-22A, 'Functional Testing of Turbine Trip From AMSAC Output Relay Actuation'; STP M-22B, 'Functional Testing of Motor-Driven Auxiliary Feedwater Pumps Start and Blowdown Isolation From AMSAC Output Relay Actuation'; and STP M-22C, 'Functional Testing of Turbine Driven Aux. Feedwater Pump Starts From AMSAC Output Relay Actuation'. The frequency of these test activities is proposed to be changed from once every 18 months to once per refueling interval (24 months).

Safety Evaluation Summary:
AMSAC provides a method, that is diverse and independent of the Reactor Protection System (RPS), to trip the turbine and start the auxiliary feedwater pumps in the event an ATWS results in the loss of the secondary heat sink. AMSAC functions as a back-up to the RPS. In the event of a failure of

AMSAC, the RPS would trip the reactor and start the AFW pumps if the secondary heat sink were lost. AMSAC does not cause an accident nor does it contribute to the occurrence of an accident previously evaluated in the SAR. AMSAC is designed to mitigate the consequences of an event.

The proposed 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 does not result in any physical modifications to either AMSAC or any important to safety structure, system or component. In addition, increasing the surveillance interval will not alter the operation of AMSAC, the intent or method by which the surveillance is presently conducted, or the scope or intent of the associated surveillance test procedures. Finally, increasing the surveillance interval does not adversely affect safety function performance, or alter the intent or method by which surveillance tests are performed.

Conclusion:
The proposed change does not alter the way any structure, system, or component functions, and does not change the manner in which the plant and the system is operated. The increased surveillance interval does not alter the intent or method by which the surveillance is presently conducted, nor does it change the scope or intent of the associated surveillance test procedures. This activity does not result in an unreviewed safety question.

98-139 ECG 4.4, Revision 1, Turbine Overspeed Protection System

Reference Document No.: ECG 4.4
Rev. No: 1
Reference Document Title: Turbine Overspeed Protection System
Safety Evaluation Description:
Technical Specification ¾.3.4.1, Instrumentation - Turbine Overspeed Protection," was relocated to ECG 4.4 as approved by License Amendment 120 (Unit 1) and 118 (Unit 2) dated February 3, 1998. FSAR Section 16.1, "Technical Specifications and Equipment Control Guidelines," and Administrative Procedure OP1.DC16, "Control of Plant Equipment not Required by the Technical Specifications," require a 10 CFR 50.59 safety evaluation for all ECG changes.

This 10 CFR 50.59 safety evaluation evaluates the acceptability of extending the frequency of surveillance requirement SR4.4.2, "Perform a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems," from 18 months to 24 months to coincide with extended fuel cycles.

Safety Evaluation Summary:
A review of the accident scenarios evaluated in the FSAR Update identifies that the relevant accident analysis is that of the depressurization of the main steam system. The depressurization of the main steam system may be caused by several events including the collision of a turbine generated missile with a steam lead. Turbine generated missiles are the result of a number of factors including the failure of the overspeed protection system. The proposed change to the surveillance frequency for the turbine overspeed protection system channel calibration, from a nominal 18 months to a nominal 24 months will not change the consequences of this previously analyzed accident.

Furthermore, the proposed change does not alter the way any structure, system, or component functions, and does not change the manner in which the plant and the system is operated. The increased surveillance interval for the turbine overspeed protection system channel calibration does not alter the intent or method by which the surveillances are presently conducted, nor does it change the scope or intent of the associated surveillance test procedures.

The surveillance, maintenance and operating histories of the turbine overspeed protection system do not show any significant problems, and tend to indicate that the system will continue to effectively perform its design function for the longer operating cycles. In addition, there have been no time or cycle length dependent problems associated with this system.

Conclusion:
This change will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria. This activity does not result in an unreviewed safety question.

98-140 ECG 7.7 Reactor Vessel Head Vents – Surveillance Requirements 7.7a, 7.7b, 7.7c

Reference Document No.: ECG 7.7
Rev. No: 2
Reference Document Title: ECG 7.7 Reactor Vessel Head Vents – Surveillance Requirements 7.7a, 7.7b, 7.7c

Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 Evaluation is required for all ECG changes. The proposed change to the ECG surveillance requirement for the reactor vessel head vent alters the frequency at which the surveillance is conducted, and thus involves a change to the administrative controls over plant activities as described in the SAR. This result in a "Yes" to a

10CFR50.59 screen question and this Safety Evaluation is required.

This change alters the frequency at which the following surveillance test procedures are performed. STP V-2T2 " Exercising and Position Indication Verification of Reactor Vessel Head Vent Valves RCS (Both Units) 8078A, 8078B, 8078C and 8078D". This surveillance test procedure satisfies the requirements of ECG's 7.7b and 7.7c. The surveillance which satisfies the requirements for ECG 7.7a is the completion of the Sealed Valve Checklist OP K-10J (both units). This checklist is required to be current and valid in Modes 1, 2, 3 and 4 which is corresponds to the applicability of ECG 7.7a. No frequency or surveillance interval is defined in OP K-10J or the other operations procedures as it remains continuously in effect during the applicable modes and is verified during mode changes per OP A-2:II.

Safety Evaluation Summary:
The reactor vessel head vents allow the removal of noncondensable gasses from the reactor vessel head. This assists in the maintenance of natural circulation in the event that RCS flow is lost and gases build up in the reactor vessel head. These valves are ASME Section III Class I and PG&E Design class I. The failure of the vent system or its valves would result in a small LOCA of not greater than 1 inch diameter which is evaluated in the SAR. This SAR evaluated fault is not made more probable nor does it contribute to the occurrence of an accident previously evaluated in the SAR because the valves would have to fail and the proposed change does not contribute to any pressure boundary failure. No occurrence of seat failure is evident since the proposed change extends the verification of each valve's position and the ability of the solenoid operated valves to actuate. The manual sealed valves are static in all modes applicable to the ECG and therefore no credible event could lead to the disturbance of these valves and increase in the probability of the described LOCA fault or any other of an accident previously evaluated in the SAR.

The surveillance maintenance, and operating histories of reactor head vent valves indicate that the equipment will continue to perform satisfactorily with a longer surveillance interval, and that there is no time dependency associated with the encountered component failures. In addition, the performance of RCS leakage surveillance testing on a daily basis will expose any position change of the SOVs during normal plant operations.

Conclusion:
Evaluation of the historical surveillance, operating, and maintenance data indicates that there have been no significant problems with the reactor vessel head vent system and that its reliability for valve actuation has been high. The characteristic of the vent system related to Technical Specification bases

is the maintenance of RCS pressure boundary required by Tech. Spec.3.4.6.2. This negates any need for the basis of the ECG 7.7 to incorporate any safety or design limits for RCS leakage. There is no other effect on any safety limit, protection system setpoint, or limiting condition for operation. This activity does not result in an unreviewed safety question.

98-141 Boration Injection Flow Paths

Reference Document No.: ECG 8.4
Rev. No: 1
Reference Document Title: Boration Injection Flow Paths
Safety Evaluation Description:
The proposed change would revise the test surveillance frequency requirements from 18 months to 24 months, at least once per REFUELING INTERVAL (nominal 24 months, maximum 30 months):

1. Valve Actuation - requires that each automatic valve in the CVCS boration flow path (either flow path (a) or (b) as defined below) actuates to its correct position on a safety injection test signal.
2. Boration Capability - requires that at least one CVCS boration flow path (either flow path (a) or (b) as defined below) be operable and be capable of pumping at least 30 gpm of 7,000 ppm boron solution or equivalent into the reactor coolant system.

The Two Boron Injection Flow Paths are defined as:

a. From the boric acid tanks via a boric acid transfer pump and a charging pump to the reactor coolant system.
b. From the refueling water storage tank via a charging pump to the reactor coolant system.

Safety Evaluation Summary:
The increased surveillance interval does not alter the intent or method by which the testing is conducted, does not alter the way any structure, system, or component functions, and does not change the manner in which the plant is operated. Slave relays could potentially initiate an accident if they fail and generate an actuation signal in failing. However, as documented in WCAP-14117 and WCAP-13878, approved as part of License Amendment 115/113, there is no known time dependent mechanism that would degrade the performance of the equipment during normal plant operation over the proposed increased surveillance interval.

The operability of two subsystem trains ensures that the CVCS boration capability will be available in the event of an actuation signal, assuming the loss of one subsystem through any single failure consideration. The surveillance, maintenance and operating histories of the components within the CVCS boration flow paths indicate that they will continue to perform satisfactorily with a longer surveillance interval.

Conclusion:
Evaluation of historical surveillance and maintenance data indicates there have been few problems within the CVCS boration injection flow paths. There are no indications that any potential problems would be cycle-length dependent. There is no safety analysis impact since this change will have no effect on any safety limit, protection system setpoint, or limiting condition of operation. There is no hardware change that would impact existing safety analysis acceptance criteria. This activity does not result in an unreviewed safety question.

98-142 Vibration Loose Parts Monitor Equipment Control Guideline

Reference Document No.: ECG 46.1
Rev. No: 1
Reference Document Title: Vibration Loose Parts Monitor Equipment Control Guideline

Safety Evaluation Description:
The proposed change to the Equipment Control Guideline ECG-46.1 would revise the Vibration and Loose Parts Monitoring System (VLPMS) Surveillance Requirements (SR's) to change the surveillance frequency for calibration from at least once every eighteen (18) months to at least once every twenty-four (24) months per refueling interval. The current surveillance requirements of the VLPMS consist of a daily channel check, a monthly qualitative assessment, a quarterly alarm circuit functional test, and an eighteen month channel calibration.

Safety Evaluation Summary:
The FSAR Chapter 15 events of steam generator tube rupture, rod cluster control assembly (RCCA) misoperation, partial loss of forced reactor coolant flow, small break LOCA, complete loss of forced reactor coolant flow, large break LOCA, and single reactor coolant pump locked rotor are the events associated with the reactor coolant system and which may be affected by VLPMS operability.

Steam Generator Tube Rupture: VLPMS is not credited with preventing steam generator tube rupture, therefore, the loss of VLPMS cannot increase the probability of the event. Moreover, it is not credible that loose parts in the RCS which are small enough to enter a steam generator tube (are not filtered out by the tube sheet) would cause a tube rupture. It is likely that a loose part which is able to enter a tube will travel through it, not causing a rupture, to the cold side of the steam generator. The operable VLPMS may increase the possibility of detectionof incipient steam generator tube rupture if the rupture is preceded by vibration.The loss of the VLPMS does not increase the probability of a steam generator tube rupture event.

RCCA misoperation: In particular, the event involving statically misaligned RCCA. This event, as described in the FSAR cannot be caused, prevented, detected or anticipated by the use of the VLPMS.

Partial loss of forced reactor coolant flow: The possible cause related to the VLPM may be a failure of the reactor coolant pump (RCP) due to mechanical pump (impeller/volute) damage. The RCP is downstream from the steam generator which acts as a large filter for loose parts in the RCS. Therefore, it iscredible that only small loose parts can pass through the steam generator. A largeloose part impacting a RCP would have to originate downstream from the steam generator. Such a loose part, large enough to cause a mechanical failure and resulting in a RCP flow loss , would have been left in the RCS after maintenance. The VLPMS would be of no credible value in preventing such damage.

Therefore, the loss of the VLPMS does not increase the probability of this event.

Small break LOCA: Loose parts are not credible initiators for this event. Large loose parts are filtered from the RCS by the steam generator or the vessel internal components. Small loose parts are not credible initiators of pipe breaks. Thus, the loss of VLPMS would have no effect on the probability of this event.

Complete loss of reactor coolant flow: This event can only occur due to electrical system failure and, therefore, the VLPMS operability has no effect on the probability of this event.

Large break LOCA: Large loose parts are filtered from travel in the RCS by the steam generators or the vessel internal components. Small loose parts, which are able to travel the RCS are not credible initiators of this event and, thus, the operability of the VLPMS has no effect on the probability of this event.

VLPMS is a monitoring system and cannot cause any of these accidents (meaning any of the listed Condition II, III, or IV events of Chapter 15 of the FSAR. Nor is the VLPMS relied upon to prevent these Chapter 15 events.

Extension of the channel calibration to 24 months does not violate the design of the VLPM. The removal of a channel input to the VLPM results in one remaining redundant channel in which loose parts monitoring can be compensated for. Due to the software in the Loose Parts Events Analysis Computer (LPEAC), the signal treatment normally requires 2 valid signals out of 2 channels in a zone to trigger a valid loose parts alarm. The software can

and has occasionally been changed to conservatively convert the valid alarm discrimination to 1 out of 1. Thus, the redundant channel will signal the loose part event singularly. The result is that the alerting of the potential existence of a loose part is maintained even with a postulated channel failure.

Conclusion:
The proposed change does not alter the way any important to safety structure, system, component functions, and does not change the manner in which the plant is operated. The increased surveillance interval does not alter the operation of the system, the intent or method by which the surveillance is presently conducted, nor does it change the scope or intent of the associated surveillance test procedures. The VLPMS is a non-safety related system and cannot create a safety related equipment failure. This activity does not result in an unreviewed safety question.

98-144 FSAR Update, Appendix 9.5A, Revision 12, Fire Hazards Analysis

Reference Document No.: FSAR Update Appendix 9.5A
Rev. No: 12
Reference Document Title: FSAR UPdate, Appendix 9.5A, Revision 12, Fire Hazards Analysis

Safety Evaluation Description:
AR A0453070 was generated during a recent revision (Rev. 8) to calculation M-824, "Controlled Combustible Loading Tracking Calculation". During revision 8 to calculation M-824, fire protection engineering identified numerous instances were new combustible material types were added to fire areas in the plant via design changes and the appropriate section of the FSAR was not updated with the newly added material. In addition, it was identified that the floor areas of several fire areas had been modified and the appropriate FSAR section had not been updated. Also, this evaluation will cover the addition of a generic statement to the "Transient Combustible Materials" section of FSAR, Appendix 9.5A, Fire Hazards Analysis. These changes are corrections to the combustible loading description contained in the FSAR and reflect the current plant configuration. There are no design changes , procedure changes, or calculation changes associated with this FSAR change.

Safety Evaluation Summary:
The changes to the FSAR related to this evaluation reflect the current plant configuration. This change does not introduce new combustible loading or fire hazards and no design changes, calculations or procedures are being revised as a result of this change. All combustible materials, both insitu and transient are tracked via PG&E calculation M-824, "Controlled Combustible Loading Tracking Calculation" and Interdepartmental Administrative Procedure (IDAP) OM8.ID4, "Control of Flammable and Combustible Materials". Therefore, the

probability of occurrence of a fire or an accident previously evaluated in the SAR is not affected. Also, these changes do not adversely impact the DCPP fire protection program.

Conclusion:
The changes evaluated in this safety evaluation update and correct the combustible loading descriptions in Appendix 9.5A to the FSAR. The changes reflect the current plant configuration and bring the FSAR combustible loading description into alignment with the current as-built configuration of the plant. The addition of combustible materials described in this FSAR change have been previously evaluated in accordance with the DCPP design change process. This change is a document change only and does not involve physical modifications to the plant fire areas. The addition of insitu combustible material to DCPP fire areas is strictly controlled via the design change process at DCPP. All proposed modification to the plant that add permanent combustible materials are reviewed by a fire protection engineer for potential impacts on the Fire Hazards Analysis. All combustible materials, both insitu and transient are tracked via PG&E calculation M-824, "Controlled Combustible Loading Tracking Calculation" and Interdepartmental Administrative Procedure (IDAP) OM8.ID4, "Control of Flammable and Combustible Materials". These changes to the FSAR do not affect plant equipment or configuration. Therefore, since these changes do not affect equipment required for safe shutdown and do not adversely impact the DCPP fire protection program, this change will not affect the ability to achieve and maintain safe shutdown in the event of a fire. This activity does not result in an unreviewed safety question.

98-145 Electrical Equipment Protective Devices - Motor Operated Valves Thermal Overload Protection and Bypass Devices

Reference Document No.: ECG 64.1
Rev. No: 1
Reference Document Title: Electrical Equipment Protective Devices - Motor Operated Valves Thermal Overload Protection and Bypass Devices

Safety Evaluation Description:
Per Procedure OP1.DC16, a 10 CFR 50.59 evaluation is required for all ECG changes. This LBIE evaluates the proposed change to the ECG surveillance requirements for Motor Operated Valves Thermal Overload (TOL) Protection and Bypass Devices. This change alters the frequency at which the surveillances are conducted, and thus involves a change to the administrative controls over plant activities as described in the SAR.

This change alters the frequency at which the following surveillance test procedures are performed; STP M-84B, "Motor Operated Valve Thermal

Overload Relay Calibration"; STP M-84F, "Bus F Motor Operated Valve Thermal Overload Bypass Relay Trip Actuation Device Operational Test"; STP M-84G, "Bus G Motor Operated Valve Thermal Overload Bypass Relay Trip Actuation Device Operational Test"; and STP M-84H, "Bus H Motor Operated Valve Thermal Overload Bypass Relay Trip Actuation Device Operational Test".

The frequency of these test activities is proposed to be changed from once every 18 months to once every 24 months. These changes will result in the frequency of the TRIP ACTUATING DEVICE OPERATIONAL TESTS (TADOT) for the bypass relays being extended to 24 months. The CHANNEL CALIBRATION of individual MOV TOLs will remain at least once every six years (one bus done each 24 month refueling interval).

Safety Evaluation Summary:
The motor operated valve thermal overload protection and bypass devices are not accident initiators for any DCPP accident analyses. The MOV TOLs do not cause an accident nor do they contribute to the occurrence of an accident previously evaluated in the SAR.

The MOV TOL and bypass devices are associated with valves in systems used for the mitigation of design basis accidents at DCPP. The proposed change does not alter the way the devices or valves function, and does not change the manner in which the plant and the affected systems are operated. The increased surveillance interval for the Bypass Relay Trip Actuating Device Operational Tests (TADOT) does not alter the intent or method by which the surveillance is presently conducted, nor does it change the scope or intent of the associated surveillance test procedures. The surveillance interval for the individual MOV TOL channel calibration remains the same. To test all MOV TOLs within the present six year period, one vital bus will continue to be done each refueling interval (24 months).

The accident analyses in the SAR consider a worst case single active failure. No additional failures are required to be considered unless there is reason to consider a common mode failure. The surveillance, operating and maintenance history of bypass relays indicates that the equipment will continue to perform satisfactorily with a longer surveillance interval for TADOTs and does not present a common mode failure concern. Therefore the proposed change does not affect equipment operability and does not involve an increase in the consequences of an accident previously evaluated in the SAR.

Conclusion:
The proposed 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 TADOT does not result in any physical modifications to either the MOV TOLs, their bypasses, or any important to safety structure, system or component. The overall MOV TOL required channel calibration interval of six years for each relay, remains unchanged. In addition, increasing the TADOT surveillance interval will not alter the operation of the MOV TOLs and their bypasses, the intent or method by which the surveillance is presently conducted, or the scope or intent of the associated surveillance test procedures. Finally, increasing the surveillance interval does not adversely affect safety function performance, or alter the intent or method by which surveillance tests are performed. This activity does not result in an unreviewed safety question.

98-146 FSAR UPdate, Section 9.5H

Reference Document No.: FSAR Update, Section 9.5H
Rev. No: 12
Reference Document Title: DCPP Units 1 & 2 FSAR Update
Safety Evaluation Description:
This change will replace the title of the Fire Marshall with the Supervisor, Safety and Fire Protection, and change the title of the IFO to Fire Fighter. This change is necessary because the duties previously assigned to the Fire Marshal have been delegated to the Supervisor, Safety and Fire Protection, and the IFOs have been replaced with a full time fire department. This change also deletes the name of the insurance carrier and relocates some of the program responsibilities to the correct line organization.

Safety Evaluation Summary:
The personnel associated with the Fire Protection Program are not involved with any accidents described in the SAR other than a fire. This change will replace the title of the Fire Marshall with the Supervisor, Safety and Fire Protection, the title of the IFOs, with the title Fire Fighter. This change also aligns the responsibilities in accordance with the current organization. The functions that the IFOs performed will be increased because the Fire Fighters are now part of a permanent plant fire department staffed with professional fire fighters. The Supervisor, Safety and Fire Protection will assume all of the functions currently described in the FSAR for the Fire Marshal. Moving the responsibilities to the correct line organization will not affect the program because none of the functions have been altered or deleted. Therefore, the probability of occurrence of an accident previously evaluated in the SAR is not increased.

The qualifications or training requirements for the plant Fire Brigade have not been diminished by this change. Due to other organizational changes, there is now a Fire Captain available 24 hours a day to direct the brigade in the event of a fire. Removing the requirement to have an IFO compliment the required five member fire brigade does not decrease the effectiveness of the program, because the fire brigade will now consist primarily of permanent professional fire fighters.

Conclusion:
These changes will replace the title of the Fire Marshall with the Supervisor, Safety and Fire Protection, and change the title of the IFO to Fire Fighter. These changes also align the responsibilities in accordance with the current organization. The functions of the Fire Fighters (IFOs) will be increased by this change because the Fire Fighters are now part of a permanent plant fire department staffed with professional fire fighters charged with responding in the event of a fire. This change represents a program enhancement due to the training and professional skills inherent in the creation of a full time permanent Fire Department at DCPP. The Supervisor, Safety and Fire Protection will assume all of the functions currently described in the FSAR for the Fire Marshal, and moving the responsibilities to the correct line organization will not affect the program because none of the functions have been altered or deleted. This change alters the make up of the fire brigade as described in the Emergency Plan because the requirement to compliment the fire brigade with an IFO has been deleted due to the fact the fire brigade will now consist of primarily professional fire fighters. This activity does not result in an unreviewed safety question.

98-147 Drain Holes In Pyrocrete Panels in Fire Areas 10 And 20

Reference Document No.: FHARE 138
Rev. No: 0
Reference Document Title: Drain Holes In Pyrocrete Panels in Fire Areas 10 And 20

Safety Evaluation Description:
This LBIE is being generated to document the addition of two (2) ¾ in. drain holes in the Pyrocrete panels located at the 76' elevation of Fire Areas 10 and 20. FHARE 138 is part of the Fire Protection Program, and evaluates the acceptability of the drain holes in the Pyrocrete panels. In addition, this LBIE is being performed to document the acceptability of changing the text in FSAR Appendix 9.5A which describes Fire Areas 10 and 20. The FSAR revision will indicate the presence of these drain holes in the Pyrocrete panels.

Safety Evaluation Summary:
FHARE 138 was generated to evaluate the adequacy of the Pyrocrete panels located in the cable spreading rooms, below the 12 kV switchgear rooms, with two (2) ¾ in. drain holes provided at the base of each panels. These holes will allow the drainage of water, which builds up behind these panels during periods of high rainfall. The Pyrocrete panels are located in Fire Areas 10 and 20, for Units 1 and 2, respectively, and provide a 2-hour rated barrier for the protection of redundant trains of safe shutdown circuits located at the 76' elevation of Fire Areas 10 and 20. This FHARE evaluates the ability of the panels to provide a 2-hour rated barrier with the addition of these drain holes.

FHARE 138 evaluates an as-built configuration against the effects of a fire. The FHARE does not affect the normal function of equipment important to safety. FHARE 138 shows that based on the low combustible loading in the affected area and the existing fire protection features (i.e., smoke detection, CO_2 hose-reel stations at the 85' elevation and portable fire extinguishers) in these areas the DCPP fire protection program is not impacted by the addition of these drain holes.

Conclusion:
As noted in FHARE 138, even with the two (2) ¾ in. drain holes in the bottom of the Pyrocrete panels a fire in the cable spreading room will have little affect on the panels. These openings equate to an opening of approximately 1 sq. in. total. Because of the small opening, and the location of them, it is felt that these openings will have little or no impact on the overall fire rating of the Pyrocrete barriers. In addition, for a postulated fire in this area, it is anticipated that due to the open stairwell, which communicates directly with the 85' elevation, the products of combustion will ascend to the upper elevation thereby forming the hot gas layer at the ceiling of the elevation above. Therefore, it is anticipated that the 76' elevation of these fire areas will see little increase in the room temperature.

The introduction of combustible material are controlled administratively. As a result, it is not credible to postulate large adverse quantities of combustible materials being introduced into these areas without the proper compensatory measures.

DCPP maintains a fully staffed fire brigade which can respond promptly to the notification of a fire. As such, given the fire detection system located in each of these areas which will provided early notification of a fire, it is reasoned that any fire occurring in these area will be quickly detected and extinguished in its incipient stages.

The drain holes provided in the Pyrocrete panels will not present a condition which impedes or prevents the ability to achieve and maintain safe shutdown. Therefore, the circuits located within the Pyrocrete/pull box vaults will not be jeopardized. The drain hole provisions will not affect the 2-hour rating of the Pyrocrete panel and will not affect the ability to achieve and maintain safe shutdown.

98-148 Start Up Transformer (SUT 11 or SUT21)

Reference Document No.: OP J-2:II
OP J-2:III
OP J-2:II
OP J-2:III

Rev. No: Rev 6, Rev 7, Rev 8, Rev 6

Reference Document Title: Start Up Bank Return to Service, Unit 1
Start Up Bank - Shutdown and Clearing, Unit 1
Start Up Bank Return to Service, Unit 2
Start Up Bank - Shutdown and Clearing, Unit 2

Safety Evaluation Description:
This 10 CFR 50.59 Safety Evaluation is being issued to demonstrate using one Start Up Transformer (SUT 11 or SUT21) is acceptable for providing adequate capability to both Unit 1 and Unit 2. The purpose for allowing one SUT to provide power to both Unit 1 and Unit 2 is to facilitate maintenance and provide operational flexibility in managing 230kV offsite power. The use of one Startup Transformer is described in the FSAR Update, Revision 12, Section 8.2. The following Operations procedures implement this single Startup Transformer configuration:

OP J-2:II	Start Up Bank Return to Service	Unit 1 rev 6
OP J-2:III	Start Up Bank - Shutdown and Clearing	Unit 1 rev 7
OP J-2:II	Start Up Bank Return to Service	Unit 2 rev 8
OP J-2:III	Start Up Bank - Shutdown and Clearing	Unit 2 rev 6

Safety Evaluation Summary:
Operation with the configuration of one Start Up transformer supplying Unit 1 and Unit 2 will expand the flexibility of the 230/12 kV systems. The Start Up transformers with automatic load tap changing capability provide a better response to offsite power and onsite power requirements. This configuration of a bus tie for one transformer supplying two units is discussed in the FSAR. The cross tie configuration implements a reduced transfer of certain nonvital loads on the 12 kV system and 4 kV system. The loads blocked from transferring are one nonvital 12 kV bus consisting of 2 RCPs (for each unit). The nonvital 4 kV loads consist of blocking the start of the standby condensate booster pump.

The intent of this configuration is to increase the reliability and availability of the 230kV offsite power system. Clearing the 230 kV system previously put both Unit 1 and Unit 2 into a 72 hour technical specification LCO. An accident or trip in the previous condition would rely on natural circulation cooling and diesel generators. The configuration described in this LBIE would increase the 230 kV reliability and availability to each unit. Although this configuration is discussed in the FSAR 8.2.2 as a possible failure scenario, this configuration would be used for purposes of increasing reliability and availability. DCPP normal practice of minimizing the amount of time in this configuration would continue.

In this configuration, the single SUT transformer is capable of providing adequate power to meet the shutdown requirement for the accident unit while providing standby power for the other unit. As the accident unit approaches a stable shutdown condition, the shutdown loads will be transferred from the startup transformer to the 500 kV source through backfeeding. Once the accident unit has been transferred to the 500kV source, the SUT is fully capable of supporting any type of shutdown or trip of the other unit for whatever reason. The operating instructions outlined in "O-23, the Operating Instructions for Reliable Transmission Service for DCPP" will be applicable. The operating instructions outlined in O-23 will still be applicable for other contingency scenarios, i.e., line outages and capacitor banks.

DCPP practice after unit trips and transfers to Start Up power is to restore power alignment back to the 500 kV system (delayed preferred) as soon as practical. This practice would continue for this configuration. Transferring back to the 500 kV system may require the manual operation of the LTC. The transfer would be a short duration. The case for operating the transformer in manual for both units has not been analyzed at this time. Conservatively then both units would enter the Technical Specification action statement declare the 230kV system inoperable. The tripped unit may then be manually transferred. The action statement exited on both units with the LTC restored to automatic. Once the LTC is restored to automatic both unit's configuration would be restored to analyzed scenarios. [Reference Calc 359DC rev 1]

The ISO has been provided with the allowed voltage range and minimum pre-transfer voltage at the DCPP switchyards, as well as the load required to accommodate a design basis accident in one unit and a safe shutdown in the other unit and is contractually required to maintain these voltages. In addition, the ISO is contractually required to follow Operating Instruction O-23 which details minimum voltages required for various line configurations and compensatory action required by DCPP to assure that 230 kV offsite power is operable. These voltage levels assure that the voltages at the vital 4 kV and 480V vital buses will be adequate to support the operability of required safety

related equipment.

The protective relaying for the 230 kV transmission lines has been extremely reliable and the total circuit lengths are relatively short. Should there be a disturbance on the 230 kV system, vital loads are protected and would be transferred to the diesel generators. The protective relaying for the 12 kV system with the cross tie breaker closed has been evaluated also. There is proper coordination between the relay systems to ensure adequate protection of 12 kV loads and sources.

The Station Blackout (SBO) analysis assumes the loss of all power including the loss of offsite power. The SBO analysis is not affected by this configuration.

Conclusion:
The use of one Start Up 230/12kV transformer is within and conforms to the current licensing and design basis. The implementation of reduced transfers of nonvital loads while in this configuration has been reviewed and determined to be within and conform to the current licensing and design basis.

98-149 Non-Class 1E Penetration Overcurrent Protection

Reference Document No.: DCP E-049441
Rev. No: 0
Reference Document Title: Non-Class 1E Penetration Overcurrent Protection

Safety Evaluation Description:
Change Description This safety evaluation addresses changes to the FSAR Update Section 8.3.1.4.8, "Penetrations" and related changes to the System and Topical Design Criteria Memoranda (DCMs), the Q-List and the PIMS Component Data Base. The changes clarify the implementation of PG&E's Letter to the NRC dated December 19, 1978, on the documentation of the seismic capability of the non-Class 1E containment penetration overcurrent protection. The changes:

- Clarify that since the Class 1E protective devices are not tested for operation (interrupt fault current) during the actual ground motion shaking caused by an earthquake, the NRC did not intend to impose documentation or additional testing to demonstrate operability of non-Class 1E penetration overcurrent protection devices during an earthquake.

- Credit the existing procurement documentation and engineering evaluation for overcurrent protection to demonstrate the capability to maintain containment integrity during and after a Design Earthquake (DE).

- Clarify that the primary overcurrent protection was procured prior to the issuance of PG&E's letter of December 19, 1978. Engineering judgment was applied to the basis to determine that the seismic capability met the intent of PG&E's commitments.

- Apply a failure modes and effects analysis (FMEA) to provide additional documentation that the primary and backup overcurrent protection are capable of maintaining containment integrity during and after a Design Earthquake (DE).
- Clarify that the manufacturer and model number are sufficient to document that protective devices are of high industrial quality.

Safety Evaluation Summary:
There are two issues raised here: (1) lack of documentation verifying high industrial quality; (2) protective device actuation during a DE.

The proposed change clarifies the use of procurement documentation to verify that penetration overcurrent protection devices are of high industrial quality. Though it is the policy of PG&E to procure only high industrial quality materials for DCPP, there is no uniform definition of what constitutes high industrial quality. Penetration overcurrent protection is considered to be of high industrial quality if it is: 1) manufactured by a nationally known manufacturer and, 2) a type that would normally and customarily be specified for use in industrial applications. Though not a requirement, most of the backup penetration overcurrent protection devices are of the same type as utilized in Class 1E equipment. Use of high industrial quality breakers, fuses, overload relays, and overcurrent relays is self evident by the manufacturer and model number. Therefore, the requirement for high industrial quality need not be specified as part of the procurement documentation.

The primary and backup penetration overcurrent protection serve to protect containment penetrations by opening the circuit during a short circuit or overload condition. Procurement documents or engineering evaluations of the backup penetration overcurrent protection verified the capability to withstand a DE and operate afterward. However, the procurement documents do not demonstrate the capability to actuate during a DE. In addition to the high seismic capabilities of molded case breakers, similarity to Class 1E breakers, and procurement documentation for most (but not all) of the backup breakers, a failure modes and effects analysis performed by PG&E demonstrates that there are no credible failure modes that would result in a failure to protect the penetrations for a DE.

Conclusion:

- PG&E's commitments for redundant overcurrent protection to protect the penetration during and following an accident are met.

- PG&E's commitments for the non-Class 1E protective devices to be of high industrial quality verifiable by procurement documentation (by the use of manufacturer and model number) are met.

- PG&E's commitments, as clarified by NRR in 1997, for penetration overcurrent protection that has a seismic capability of remaining operable after a DE. This is verified by procurement documentation, evaluation, or comparison with Class 1E protection are met.
- PG&E's commitment for non-Class 1E penetration protective devices to protect the penetration during a DE using engineering evaluations are met.

98-150 Diablo Canyon Power Plant Unit 2 Zinc Addition Program

Reference Document No.: Westinghouse SECL 98-117
Rev. No: 1
Reference Document Title: Diablo Canyon Power Plant Unit 2 Zinc Addition Program

Safety Evaluation Description:
This LBIE is being performed by Turbine/Steam Generator Engineering to present SECL 98-117 to the PSRC to provide for the injection of zinc into the Unit 2 RCS via the CVCS system. Zinc injection is planned for the last eleven months of Unit 2 Cycle-9 at a concentration of approximately 15 to 40 ppb. The injection of zinc into the RCS has previously been done on DCPP Unit 1 under LBIE No. 98-038.

Safety Evaluation Summary:
The planned program to initiate zinc addition at Diablo Canyon Unit 2 includes the following objectives:

- Determine the effectiveness of zinc addition in reducing the radiation dose rates associated with the primary system components,
- Determine the effectiveness of zinc in inhibiting the occurrence of PWSCC in Alloy 600 components, and
- Verify that the presence of zinc in the RCS has no unacceptable effect on fuel cladding corrosion and fuel region crud deposits.

The initial zinc addition date to the Unit 2 reactor coolant system (RCS) is scheduled for mid-November 1998, eleven months from the completion of Cycle 9.

The nominal concentration of zinc in the primary coolant for the majority of the injection period will be 40 ppb. During shutdown or periods of reduced RCS temperature, due to the retrograde solubility of zinc with temperature, the RCS zinc concentration may increase to levels greater than this nominal value. Based on previous experience at Farley Unit 2 and DCPP Unit 1, if zinc injection is stopped with the RCS below the nominal operating temperature, the zinc concentrations are efficiently reduced by the Chemical and Volume Control System (CVCS) demineralizers. Zinc additions will not be resumed until the analyzed zinc concentration falls below approximately 40±5 ppb.

The nominal RCS zinc concentration will be achieved by injecting a solution of zinc acetate in demineralized water from a mixing tank into the charging pump suction via the Volume Control Tank (VCT). The common unit injection skid was placed in service in June of this year and has been used successfully to maintain Unit 1 zinc concentration at the desired levels. The mass injection rate will be limited to a maximum of 4.5 grams/hour. The current practice at DCPP Unit 1, demonstrates that a zinc mass injection rate of 1.6 to 2.2 grams/hour is sufficient to establish and maintain an RCS zinc concentration of 40 ppb.

Routine sampling and off-line analyses of the reactor coolant using DCPP generated procedures will be employed to monitor the zinc concentration.

Conclusion:
Based on the results of the analyses and evaluations, it is concluded that the DCPP Unit 2 Zinc Addition Program may proceed without violating any of the design bases or licensing bases conditions or assumptions as described in the FSAR or Technical Specifications. No failure mechanisms are created which detrimentally affect the fuel or the RCS pressure boundary. The injection of zinc at a nominal concentration of 40 ppb will not adversely affect any component or system in the RCS and, therefore, does not represent an unreviewed safety question in accordance with the requirements and definitions presented in 10 CFR 50.59(a)(2).

98-151 Post Accident Sampling System

Reference Document No.: ECG 11.1
Rev. No: 7
Reference Document Title: Post Accident Sampling System
Safety Evaluation Description:
OP1.DC16 requires that an LBIE be prepared for all ECG changes. Additionally, the proposed change constitutes a change to the operation of the facility as describe din the SAR since the proposed will eliminate the requirement to take particular samples using PASS.

The proposed change would eliminate the requirement for PASS samples for reactor coolant dissolved oxygen and containment hydrogen and oxygen.

The proposed changes were submitted to the NRC in DCL 97-050. Included in this DCL was a 50.59 evaluation, which is included in this LBIE. During discussions with the NRC, they indicated that they wanted to review the proposed PASS changes regardless of whether the 50.59 evaluation indicated that the changes constituted a USQ. The NRC approved the changes in a letter dated February 5, 1998 (CHRON 234942). The NRC did not indicate in their safety evaluation that the proposed changes constituted a USQ.

Safety Evaluation Summary:
The changes proposed in this LBIE have been previously reviewed and approved by the NRC in a letter dated February 5, 1998. The changes below are from the 50.59 evaluation prepared and submitted to the NRC that resulted in the February 5, 1998 letter.

The changes to the equipment control guidelines (ECG) and the Final Safety Analysis Report (FSAR) Update have no adverse impact on the probability of occurrence of an accident previously evaluated in the Diablo Canyon Power Plant (DCPP) FSAR Update. No mechanism has been identified by which the proposed ECG and FSAR Update changes will degrade the performance of any other plant safety system. The changes do not adversely affect the normal plant operating conditions or postulated accident conditions. Further, the proposed ECG and FSAR Update revisions will not increase challenges to the safety systems assumed to function in the accident analysis, such that safety system performance is degraded below the design basis. Therefore, the probability of any accident previously identified will not increase.

There is no change to the effectiveness of the Emergency Plan or the capability for post-accident sampling and analysis as required by NUREG-0737. No increased leakage is expected as a result of these changes. The proposed ECG and FSAR Update changes were not found to have any adverse effect on systems or equipment necessary to mitigate the consequences of previously evaluated accidents. No failure or damage due to the ECG or FSAR Update revisions has been identified which would result in greater post-accident consequences as defined by the acceptance criteria of the guidelines of 10 CFR 100 for those accidents previously analyzed. Therefore, the consequences of any accident previously identified will not increase.

Conclusion:
The proposed ECG changes will not affect the operability of any nuclear steam supply system or component and will not decrease the margin of safety in the basis of any technical specification (TS) within their respective area of responsibility. There is no creation of any condition more limiting than that assumed in the plant safety analyses, and no reduction in margin of safety in any TS. This activity does not result in an unreviewed safety question.

98-152 Steam Generator Tubesheet Stabilizer (FTI)

Reference Document No.: M000048-1
Rev. No: 1
Reference Document Title: Steam Generator Tubesheet Stabilizer (FTI)
Safety Evaluation Description:
The steam generators are described in Section 5.5.2 and 15.4.3 of the FSAR and the stabilization of the steam generator tubes has a potential impact on the performance of the steam generator and a potential impact on the accident analyses. The LBIE is also being performed because a vendor supplied safety evaluation is being used in support of this design change.

Safety Evaluation Summary:
The modifications affect the RCPB. Inside the steam generator (SG), the previously evaluated RCPB accident is the SGTR. Defective tubes and tube remnants are removed from service by plugging and are no longer capable of causing an SGTR. The tubes adjacent to certain defective tubes and tube remnants, however, will remain a possible site for an SGTR. Therefore, the degradation of or any increased threat to their integrity could increase the probability of occurrence of this accident.

Tubes or tube remnants exhibiting axial or circumferential defects at the tubesheet face or TSP1 that are shown by analysis to be subjected to unacceptable fluid induced excitation or vibrations are stabilized by the insertion of a cable stabilizer. The stabilizer consists of a 1/2 inch OD, 6x19 wire rope, attached on one end to a rolled plug. Tube cable stabilizers are used to limit the motion of a cantilevered tube caused by a single tube severance or unstable tube remnants left after tube pulls. The stabilized tubes and unstabilized tubes remaining in the SG have been analyzed to show that no compromising contacts with adjacent tubes will occur during normal operations and accident conditions.

A pressure boundary failure of the installed tube plugs would be similar to an SGTR event. The cable stabilizer is attached to the tube plug and therefore the integrity of the tube plug must be assured in this configuration. The integrity of the plugs is assured by the following:

- Evaluations have shown that loading imparted to the plug by the stabilizer is negligible and has no effect on the pressure boundary integrity of the plug.
    - Mechanical rolled plugs are reliable.
    - Plug material is not susceptible to degradation.

The relevant, evaluated accidents of Chapters 6 and 15 typically assume 15% of the SG tubes are plugged. Since this design change requires that the cable stabilizer be installed with a mechanical plug, the percentage of plugged tubes cannot be increased beyond that permitted by tube plugging criteria. Therefore, the consequences of accidents (e.g., FSAR Update, Section 15.5) cannot be increased since the plant condition remains within the assumptions of the analyses.

Conclusion:
It was concluded that the added mass (stabilizing 10% of the total steam generator tubes that are permitted to be plugged based on the plugging margin) of the stabilizers does not have a significant effect on steam generator seismic loadings. Thus, the number of installed stabilizers is limited to a maximum of 10% of the total S/G tubes that are permitted to be plugged based on the plugging margin.

By maintaining the required integrity of the RCPB, these passive, structural modifications will not change any consequences of previously analyzed events (both LOCA and non-LOCA). By assuring plug integrity, Containment integrity is maintained for an MSLB outside Containment upstream of the MSIVs. The SGTR event scenario (including mitigating, operator responses) is not impacted by the localized, passive modifications inside the SG. This activity does not result in an unreviewed safety question.

98-153 Steam Generator Tube U-Bend Stabilizer (FTI)

Reference Document No.: M000057-1
Rev. No: 2
Reference Document Title: Steam Generator Tube U-Bend Stabilizer (FTI)
Safety Evaluation Description:
The steam generators are described in Section 5.5.2 and 15.4.3 of the FSAR and the stabilization of the steam generator tubes has a potential impact on the performance of the steam generator and a potential impact on the accident analyses. The LBIE is also being performed because a vendor supplied safety evaluation is being used in support of this design change.

Safety Evaluation Summary:
The modifications affect the RCPB. Inside the steam generator (SG), the previously evaluated RCPB accident is the SGTR. Defective tubes and tube remnants are removed from service by plugging and are no longer capable of

causing an SGTR. The tubes adjacent to certain defective tubes and tube remnants, however, will remain a possible site for an SGTR. Therefore, the degradation of or any increased threat to their integrity could increase the probability of occurrence of this accident.

Defective tubes or tube remnants that are shown by analysis to be subjected to unacceptable fluid induced excitation or vibrations are stabilized or dampened (rows with unsupported U-bends) by the insertion of a U-bend cable stabilizer. The stabilizer consists of a 1/2 inch OD, 6x19 wire rope, attached on one end to a rolled plug. Tube cable stabilizers are used to limit the motion of a cantilevered tube caused by a single tube severance or unstable tube remnants left after tube pulls and to dampen tubes with unsupported U-bends. The stabilized tubes and unstabilized tubes remaining in the SG have been analyzed to show that no compromising contacts with adjacent tubes will occur during normal operations and accident conditions.

A pressure boundary failure of the installed tube plugs would be similar to an SGTR event. The cable stabilizer is attached to the tube plug and therefore the integrity of the tube plug must be assured in this configuration. The integrity of the plugs is assured by the following:

- Evaluations have shown that loading imparted to the plug by the stabilizer is negligible and has no effect on the pressure boundary integrity of the plug.
    - Mechanical rolled plugs are reliable.
    - Plug material is not susceptible to degradation.

The relevant, evaluated accidents of Chapters 6 and 15 typically assume 15% of the SG tubes are plugged. The total number of allowed tubes with U-bend stabilizers is only 3%. Therefore, the consequences of accidents (e.g., FSAR Update, Section 15.5) cannot be increased since the plant condition remains within the assumptions of the analyses.

Conclusion:
It was concluded that the added mass (stabilizing 3% of the total steam generator tubes that are permitted to be plugged based on the plugging margin) of the u-bend stabilizers does not have a significant effect on steam generator seismic loadings. Thus, the number of installed u-bend stabilizers is limited to a maximum of 3% of the total S/G tubes that are permitted to be plugged based on the plugging margin.

By maintaining the required integrity of the RCPB, these passive, structural modifications will not change any consequences of previously analyzed events (both LOCA and non-LOCA). By assuring plug integrity, Containment integrity is maintained for an MSLB outside Containment upstream of the

MSIVs. The SGTR event scenario (including mitigating, operator responses) is not impacted by the localized, passive modifications inside the SG. This activity does not result in an unreviewed safety question.

98-154 Ranges for RM-84 and RM-85

Reference Document No.: AT FSAR A0466715
Rev. No:
Reference Document Title: Ranges For RM-84 And RM-85 Shown Incorrectly On FSAR Table 11.4-1

Safety Evaluation Description:
On FSAR Table 11.4-1, the ranges for R-84 and R-85 are shown to be 0. 1 to E4 mR/hr. The correct range for these devices is 0.1 to E3 R/hr as provided in the Radwaste Storage Building Panel for RM-84 and RM-85.

Safety Evaluation Summary:
Solid Radwaste Inspection Station Radiation Monitors RM-84 and RM-85 are provided to permit the assessment of the contact (RM-84) and one-meter (RM-85) radiation dose rates being given off from material containers being prepared for storage and shipment as solid radioactive waste. These monitors do not control radiological releases, and do not act to mitigate an accident or transient. On FSAR Table 11.4-1, the ranges for RM-84 and RM-85 are shown to be 0.1 to E4 mR/hr. This is incorrect. The correct range for these devices is 0.1 to E3 R/hr as provided in Radwaste Storage Building Panel for RM-84 and RM-85. Radiation level measured can go up to 100 R/hr and the range on FSAR Table goes only up to 10 R/hr. As the plant gets older, the radiation level is bound to go up. Hence, 0.1 to E3 R/hr is the appropriate range for RM-84 and RM-85 as provided in the plant.

Conclusion:
The ranges for the Radiation Monitors R-84 and R-85 on FSAR Table 11.4-1 will be changed to show the correct ranges.

98-155 Compensatory Measures for Open ASW Pump Room Doors

Reference Document No.: ECG 80.1
Rev. No: 2
Reference Document Title: Doors Required for HELB, HVAC, or Flood Protection

Safety Evaluation Description:
This revision to ECG 80.1 allows opening an ASW pump room door for more than 15 minutes, if necessary to support operations or maintenance activities, provided the following compensatory measures are established: (1) notify Security and the shift foreman, (2) check that the redundant ASW pump is available and that its watertight door is secured, and (3) establish a "door

watch" with instructions to close the ASW room door if there is a tsunami warning, an earthquake or other flooding event. ECG 18.7, "Fire Rated Assemblies," fire watch requirements must be met as well because the open door creates an impaired fire barrier.

Safety Evaluation Summary:
ECG 80.1 is being modified to allow an ASW pump room door to be open for more than 15 minutes to support operations and maintenance activities, without declaring the associated pump inoperable, by providing appropriate compensatory measures including establishment of a "door watch" to close the door in a flooding event

The basis for allowing the door watch is that there is sufficient warning of a flooding event to allow a cognizant individual to close the pump room door. For internal flooding, the intake structure fills at a low enough rate to allow the door to be closed. For a tsunami generated from a distant event, sufficient warning will be provided and existing procedures require that the intake be evacuated on receipt of a tsunami warning, and that the door be closed. For tsunamis generated by a nearby event, the earthquake activity will provide sufficient warning to allow the watertight door to be closed.

An open ASW pump room door causes the pump room fire barrier to become impaired. Therefore, the fire watch requirements of ECG 18.7 must be followed.

Conclusion:
The use of a door watch provides an adequate compensatory measure for a flooding event to allow an ASW room door to be left open to support operational, maintenance and inspection activities in accordance with ECG 80.1. The existing requirements of ECG 18.7 regarding impaired fire barriers remain valid, and provide adequate fire protection.

98-156 RCP 1-3 Lower Radial Bearing TE-153 Removal

Reference Document No.: DCP M-49471
Rev. No: 0
Reference Document Title: RCP 1-3 Lower Radial Bearing TE-153 Removal

Safety Evaluation Description:
This design change eliminates a leak at the thermowell adapter fitting on RCP 1-3, TE-153 (lower radial bearing outlet temperature) by removing the RTD and replacing the adapter with a pipe plug installed in the thermowell. This will disable the RCP 1-3 lower radial bearing outlet temperature indication. The reason for this modification is a leak at the adapter fitting (Ref. A0473369).

Other measurements are available (such as No. 1 seal Leakoff flow and temperature, and axial and radial vibration monitoring, and CCW flow indication) which provide indication of RCP performance. The temperature indication provided by TE-153 is Class II and has no safety related function.

This modification interfaces with licensing basis document because the FSAR Update discusses indications of RCP malfunctions. Lower radial bearing temperature is one of the items discussed.

Safety Evaluation Summary:
RCP 1-3 lower radial bearing temperature indicator TE 153 will be disabled by this design change. Other indications are available to plant operators, such as No. 1 Seal Leakoff, flow and temperature, axial and radial vibration monitoring, and CCW flow indication, which may be used to monitor the condition of the RCP.
The key issues related to this design change are the removal of RCP 1-3 lower radial bearing temperature indication and replacement of a threaded adapter with a threaded pipe plug. The compete failure of the pipe plug is enveloped by the small break analysis.

Conclusion:
Alternate methods are available to monitor the condition of RCP 1-3. The small break analysis envelopes unlikely failure of the pipe plug. Therefore, this change does not represent an unreviewed safety question.

98-157 Replacement of Aux. Bldg. Ventilation System Supply Air Roughing Filters

Reference Document No.: H-49340
Rev. No: 0
Reference Document Title: Replace Aux. Bldg. Supply Air Filters
Safety Evaluation Description:
To potentially reduce the volume of radwaste generated by the plant, the roughing filters for the Aux. Bldg. Ventilation System Supply Air are to be replaced. The previous design was of a 'throw-away' type; the new design is incinerable. These pocket filters are installed in banks on the upstream side of the Supply Air fans to remove dust and other particulate matter from the air stream.

Safety Evaluation Summary:
These passive components in the plant require no operator interface. The filters are replaced as part of normal maintenance activities when the pressure drop across them indicate that they have become dirty. As Design Class II items, they serve no safe shutdown or accident mitigation function. They are not credited with any performance related to plant safety. Their effect on the Fire Protection Program, principally with respect to combustibles

loading, has been addressed with no resulting adverse affect on the ability to achieve and maintain safe shutdown in the event of a fire. The only interface with the licensing basis is the change associated with filter technical data and combustibles' description presented in the FSAR Update.

Conclusion:
The revised filter efficiency and design flow and pressure drops associated with these newer filters maintain the performance of the Aux. Bldg. supply air system. Removal of dust and other particulate debris from the incoming air stream is not a safety-related function. Further, a failure modes and effects analysis has shown that no credible failure of these filters is capable of adversely affecting the ability of the supply air system from providing the required design cooling for the Aux. Bldg., including the systems and components it contains.

98-158 Qualification of FCV-38

Reference Document No.: DCP J-49461
Rev. No: 0
Reference Document Title: Qualification of FCV-38
Safety Evaluation Description:
This change revises the design basis for the AFWS to ensure that sufficient AFW is available to at least two intact SGs in the event of postulated rupture between the non-return valves associated with FCV-37 and 38, and upstream of FCV-95, of the four inch steam supply line to the AFWP 1 turbine. In support of this change, FCV-38 is documented as being fully qualified for a HELB environment. All supporting documentation, including DCMs, EQ Files, and Design Calcs, is revised to reflect this qualification. This change also clarifies the licensing basis for AFW regarding the consequences and mitigating features for a postulated rupture of the four inch steam supply line to the AFWP 1 turbine (reference FSAR 3.6.1.2(3)(3)). It also clarifies that the assumption for operator action to isolate feedwater to a faulted steam generator within ten minutes applies to steam line breaks outside of containment as well as inside containment.

This is a documentation only change. No SSCs are physically impacted in any way. In the present configuration, it is possible to lose all feedwater in the event of a line break in the location described above. The line rupture disables the turbine driven AFWP 1. The line rupture also raises the temperature of the GE/GW area which will cause the LCVs associated with AFWP 3 to fail open. Although AFW can be fed through the failed open valves, the pump is subject to tripping on overcurrent due to pump runout conditions. A single failure of the remaining AFWP 2 would then disable all means of supplying feedwater. In the past, a break in a four inch line was assumed to be made up by normal feedwater.

Therefore, if no reactor trip occurred, the plant could be shut down in a normal fashion. However, following the BIT removal, an analysis by Westinghouse showed that a reactor trip may occur from such an event if plant operators have not enacted a manual reactor trip upon recognition of the steamline break. This change allows the crediting of remote manual operation (closure) of the steam supply valves, FCV-37 and 38, even in a resulting steam environment for FCV-38 (FCV-37 is located in a mild environment). Closure of these valves isolates the break and effectively ends the event (and ends the concern for runout of AFWP 3) as there are no longer any faulted steam generators. Also, since these valves can be credited to operate, a single failure of either AFWP 2 to operate, or either FCV-37 or 38 to close cannot disable all feedwater.

Safety Evaluation Summary:
This is a document only design change. It does not physically impact any SSC. However, it does clarify the basis for feedwater isolation to a faulted steam generator even for MSLBs outside of containment. This has impacted Operations' procedures to ensure that the necessary isolation occurs within the ten minute period assumed in various analyses relative to the mass energy release of a MSLB. The licensing and design bases for AFW relative to the event described have been unclear in the past. This change clarifies both the licensing and design bases to ensure proper understanding by plant personnel in the future. The revised licensing and design bases envelope any previously understood conditions. The licensing and design bases addressed by this LBIE are for the changes described in DCP J-49461. They envelope all previous equipment requirements. A separate review performed as an action of NCR N0002070 addresses past Operability concerns.

Conclusion:
This safety evaluation concludes that the described document change only modification does not result in an unreviewed safety question. The modification ensures that the required AFW flow to a minimum of two intact steam generators will always be available to allow and maintain safe shutdown during the limiting plant events (Loss of Feedwater; Minor Secondary System Pipe Rupture).

99-001 DCM T-11 Revision

Reference Document No.: DCM T-11
Rev. No: 8
Reference Document Title: Control of Heavy Loads
Safety Evaluation Description:
This LBIE is prepared for revisions to DCM T-11, Rev. 8 processed by the following AT-DCMC's:

1) AT-DCMC A0454218

DCM T-11, Table A4.4.1-2 identifies the loads, their weights, the lifting devices and applicable maintenance procedures. In this table, the original Reactor Vessel Head Tensioners are addressed. A similar table is included in SSER 27, as Appendix A (Technical Evaluation Report prepared by INEL for NRC). This SSER addresses the lifting device for RV Head Tensioner as a 2T monorail and defines the heavy load as a 1.3T tensioner. The new Reactor Vessel Head Tensioners are a different model and weigh more (2T) than the old models and is within the capacity of the lifting device. Accordingly, this DCM is revised to reflect the information to include the new tensioners, in addition to the old tensioners which could be used as spares.

2) AT-DCMC A0341748

DCM T-11, Section 5.7.d for Surveillance/Maintenance Requirements and PCD No. T03529, identify ASME Section XI (Inservice Inspection and Testing of Nuclear Power Plant Systems) as a governing code for weld repairs and inspections of the reactor vessel internals lifting device, reactor head lifting device and the reactor coolant pump motor lifting device. This referenced code is an inadvertent error and is being revised in both the DCM & PCD to reflect the correct requirements governing maintenance & repair in accordance with NUREG-0612 submittal, Section A2.12 of Appendix A. In accordance with this submittal, DLAP MA1.DC12, Sections 5.7 & 5.8 provide the requirements for Repairs and Maintenance.

Safety Evaluation Summary:
The revisions to DCM T-11 and PCD in accordance with the following AT-DCMC's, as addressed in this LBIE, have no impact on plant operation, including impacts on operator actions, or on the licensing commitments and requirements.

1) AT-DCMC A0454218

Table included in SSER 27, as Appendix A evaluated the acceptability of the lifting devices. The loads identified in this table with weights were for clarification of the adequacy of the lifting device. Changing of the load weight without exceeding the capacity of the lifting device has no impact on the capability of the device to provide its intended function as designed.

2) AT-DCMC A0341748

PCD No. T03529 & DCM are revised to correct an inadvertent error and to reflect the correct commitment in accordance with NUREG-0612 submittal to NRC.

Conclusion:
This update to DCM T-11 and change to PCD Commitments T03529 does not result in an unreviewed safety question.

Use of the new 2T tensioners instead of the 1.3T tensioners on the 2T Reactor Vessel Head Tensioner Monorail is within the design capacity and does not impact the safety function as designed.

Correction of an obvious error in the DCM & PCD with the correct commitment of the NUREG-0612 submittal does not impact the licensing basis.

99-002 Chemical Cleaning of the Generator Stator Cooling Water System

Reference Document No.: TP TO-9902
Rev. No: 0
Reference Document Title: Chemical Cleaning of the Generator Stator Cooling Water System

Safety Evaluation Description:
Perform chemical cleaning of unit 1 main generator stator winding on-line. This involves injecting ppm-level Na_2EDTA, hydrogen peroxide, and possibly traces of a reaction activator or inhibitor into the main generator stator cooling water system. The Na_2EDTA dissolves the copper oxide on the surface of the copper stator winding and forms a soluble ion which will be eventually removed in the stator cooling water system demineralizer. The peroxide acts to oxidize Cu_2O to CuO so it can be dissolved by the EDTA, and the activator/inhibitor are used to control reaction times. This method of cleaning is licensed to ABB Kraftwerke of Germany and is called the On-line Cuproplex process. An experienced specialist from ABB will direct the cleaning process at DCPP.

Safety Evaluation Summary:
Any possible accident due to this activity is bounded by the turbine trip which is a condition 2 fault evaluated in FSARU Chapter 15 Sections 15.2 and 15.5. The FSARU concludes that the reactor coolant system integrity and main steam system integrity are maintained and the core safety limit will not be violated. This results in insignificant radiation exposures as stated in the last paragraph of FSARU section 15.5.10.

Conclusion:
Chemical cleaning the main generator on-line will not create any unreviewed safety question.

Note: There is a statement in DCM S-61A stating that it is required to maintain stator cooling water conductivity below 1.5 microS/cm. This statement in the DCM is a configuration statement and not a design bases or

a licensing commitment. The max. conductivity limit is 9.5 microS/cm in Vendor Manual 663280-41. The 1.5 microS/cm is just an operational limit. Since the increase in stator water conductivity above 1.5 microS/cm will only last during the cleaning period, no change to the DCM is made.

99-003 Steam Generator Scale Conditioning Agents

Reference Document No.: OP F-5:II
Rev. No: 21
Reference Document Title: Chemistry Control Limits and Action Guidelines for the Secondary Systems

Safety Evaluation Description:
This LBIE is being performed because the use of scale conditioning agent chemicals at Diablo Canyon will be the first field application of scale conditioning agent technology at an operating nuclear power plant.

Safety Evaluation Summary:
The use of scale conditioning agents will not adversely affect the steam generator tubes, tube support plates, or secondary shell relative to their structural integrity and their safety functions. Moreover, the use of scale conditioning agents will not act as an initiator for any accidents previously evaluated in the FSAR.

The use of scale conditioning agents will not impact the accident analyses or plant accident scenarios. The response of the plant safety systems, when subjected to accident conditions, will not be affected so as to prevent the mitigation of accidents previously evaluated in the FSAR. The scale conditioning process will not affect safety system performance and, therefore, will not increase the consequences of an accident previously evaluated in the FSAR.

The scale conditioning agents are used when the plant is shut down. The configuration and operation of the steam generator are not altered such that a different type of accident could be created.

Conclusion:
The use of scale conditioning agents in the Diablo Canyon Units 1 and 2 steam generators has been evaluated using the criteria of 10 CFR 50.59(a)(2). Based on the evaluation it has been concluded that the plant components and safety systems will not be adversely affected during normal operation and accident conditions by the use of the cleaning system. As such, this situation does not represent an unreviewed safety question.

99-004 Fire Barrier Drawing Revisions

Reference Document No.: DCP M-049464
Rev. No: 0
Reference Document Title: Fire Barrier Drawing Revisions
Safety Evaluation Description:
This DCP changes the fire barrier drawings to clearly indicate the fire barriers credited in the DCPP Fire Protection Program. The need for the drawing revisions was noted during the barrier inspections performed as part of the penetration seal program. This change will also correct discrepancies in the differentiation between the Appendix A and the Appendix R barriers, and modify the HELB designator for others. The rating of the barriers will remain the same as the existing rating.

Safety Evaluation Summary:
This DCP changes the fire barrier drawings to clearly indicate the fire barriers credited in the DCPP Fire Protection Program. The need for the drawing revisions was noted during the barrier inspections performed as part of the penetration seal program. This change will also correct discrepancies in the differentiation between the Appendix A and the Appendix R barriers, and modify the HELB designator for others. The rating of the barriers will remain the same as the existing rating. Some of the drawing changes involved clarification of the as-built rating of the barriers.

Conclusion:
This change will not adversely impact the DCPP Fire Protection Program or the ability to achieve and maintain safe shutdown. This change will also not affect the ability to mitigate the effects of a HELB event.

99-005 DCP J-49444 Replace Meteorological Processors

Reference Document No.: DCP J-49444
Rev. No: 0
Reference Document Title: DCP J-49444 Replace Meteorological Processors
Safety Evaluation Description:
DCP J-49444 replaces the primary and backup meteorological processors. A FSAR change is required to add that the backup meteorological processor communicates with SPDS.

Safety Evaluation Summary:
FSAR sections 2.3.3 and 2.3.3.7 describe the equipment which the backup meteorological processor communicates with. A FSAR change is required to add SPDS to the equipment which the backup meteorological processor communicates with.

Meteorological data is required in both the control room and the emergency response facilities per NUREG-0654. Primary and backup meteorological computers (YM-559 and YM-569) process the meteorological data and transmit it to the required locations.

The existing computers are 286 DOS-based machines, make use of an 8 port serial card and are not year 2000 compliant. The demands being made on the current system are beginning to tax its performance (i.e. data links are experiencing sporadic time out problems). The operating system and hardware of these computers is not sufficient for the for current and future requirements.

Additionally, the Emergency Response Data System (ERDS) gathers meteorological data from the HP1000 network server that will be deleted when Banyan Vines is replaced.

Conclusion:
This is a minor change to equipment description in the FSAR. There is no licensing basis impact for this FSAR change.

99-006 Installation of Plug in CST Level Instrument Tap - Unit 1

Reference Document No.: Jumper Log
Rev. No: 0
Reference Document Title: Jumper Log for Installation of Plug in CST Level Instrument Tap - Unit 1
Safety Evaluation Description:

Safety Evaluation Summary:
Install a plug in Unit 1 Condensate Storage Tank (CST) instrument tap for LT-40. This evolution requires the use of a diver in an operable tank. This LBIE evaluates:

1. The installation of the plug.
2. The potential of the plug to dislodge and become foreign material.

The use of a diver in the operable CST and impact on the tank's safety function.

The installation of the plug will render the use of Nozzle 9 inoperable for use as an instrument tap. This is acceptable since LT-40 which is hydraulically connected to Nozzle 9 has been declared inoperable by the Shift Supervisor. The installation of the plug by the diver will be done while the CST is operable. All applicable personnel safety precautions will be followed. There is no credible failure mechanism of the diver or the plug which would render the CST inoperable. In the unlikely event of an actual demand on the system while the diver is in the tank, the velocity is sufficiently low and the vortex cage is large

enough to avoid a personnel and nuclear safety hazard.

Conclusion:
The implementation of this jumper log stops a nuisance water leak from the CST. The implementation of this jumper log is acceptable.

99-007 Filling the Refueling Cavity Using Containment Spray System

Reference Document No.: OP B-2:II
Rev. No: 23
Reference Document Title: RHR - Filling The Refueling Cavity
Safety Evaluation Description:
Each refueling outage the refueling cavity is flooded. A number of options in how this is accomplished are described in procedure OP B-2:II. One option is described in section 6.12 which allows the Containment Spray (CS) and Residual Heat Removal (RHR) to be cross tied through the 9003A/B valves. Previously, this option was only allowed with the reactor defueled.

This procedure change allows the CS system to be cross tied to RHR in this abnormal configuration when fuel is in the reactor vessel.

Safety Evaluation Summary:
Each refueling outage the refueling cavity is flooded. This is described in the FSAR at sub section 5.5.6.2.2.4. This description is general in nature, and includes reference to using the Containment Spray system as one source to flood the refueling cavity. The specific mechanisms and sequences on how the Containment Spray System accomplishes this task are not described in the FSAR.

This evolution is not new at DCPP. Previously, this evolution has been successfully performed for the purpose of testing the CS-1-9003A/B valves under procedure TP TB 92-18 (now expired). The TP TB 92-18 procedure, and its safety evaluation were approved by the PSRC and signed by the plant manager 3/2/93. Previous to that, this evolution was performed, also for the purposes of testing, as STP V-17 (now rescinded). The major difference between those previously approved tests and the evolution being evaluated for this procedure change is the intent of the procedures. The TP and the STP intentions were to test the CS-1-9003A/B valves, and as a result the refueling cavity filled. This procedure change does not test the CS-1-9003A/B valves. Its aim is to fill the refueling cavity.

The flow path for the method described in this procedure change is from the RWST, motive force from a CS pump, through the CS/RHR cross tie, and injecting into the RCS/Reactor cavity through one train (two loops) of the RHR piping. Although flow in the CS/RHR cross tie line is reverse from design, the

evolution operates both systems within their design parameters for flow, temperature, pressure, etc.

The RHR and CS systems are cross tied through the 9003A/B valves. Normally these valves are interlocked to allow operation only under Cold Leg Recirc conditions. For the purposes of this evolution, these interlocks are bypassed by either installation of jumpers (preferred method), or manual MOV operation (alternate method). The installation of jumpers on the 9003A/B valves is spelled out in STP V-2B2, and this procedure change uses that STP's appendices to accomplish the task.

RHR cooling is maintained through two loops throughout the evolution. The other two loops provide a fill path for raising refueling cavity level. RHR operability is not impacted. This is justified because of two reasons. First, minimum required Tech Spec RHR flow requirements to two loops is maintained. Second, because the 9003A/B MOV's are capable of being closed from the control room during this evolution, if cooling from the RHR train in which CS is injecting to fill is needed, then CS can be immediately isolated, and the standby RHR train placed in service.

Conclusion:
Filling the refueling cavity with fuel in the vessel using a Containment Spray Pump does not pose an unreviewed safety question. This is based on the following:

- The Containment Spray System is not required in Mode 6, therefore it is not required to perform any ECCS function,
- Core cooling is maintained by the in service RHR pump recirculating through two RCS loops,
- All systems are operated within their design parameters for flow, pressure, temperature, etc.
- The standby RHR train is not rendered inoperable by cross tying to the Containment Spray System through the 9003A/B valves,
  and
- If needed, the standby RHR train can be easily and quickly placed in service by closing the 9003A/B valve and starting the RHR pump.

99-008 Auxiliary Saltwater Continuous Chlorination System

Reference Document No.: ECG 17.2
Rev. No: 1
Reference Document Title: Auxiliary Saltwater Continuous Chlorination System

Safety Evaluation Description:
It is proposed that ECG 17.2 Bases section be revised to allow continuous auxiliary saltwater (ASW) chlorination of both Units as long as there are two or

more circulating water pumps (CWPs) in operation for dilution flow. It is common to have one CWP down during a curtailment and two CWPs down during a unit outage. During these periods it is very desirable to continue to chlorinate the ASW seawater side conduits and heat exchangers. The current ECG 17.2 revision has instructions to either drop one train of ASW chlorination or get approval from Chemistry to chlorinate both ASW trains when one CWP is down. Chemistry has found over the years that the dilution from three or even two CWPs is sufficient to allow both ASW trains to remain in continuous chlorination mode (1ppm) without exceeding the National Pollutant Discharge Elimination System (NPDES) discharge limit. The ECG is being changed to allow two train chlorination to continue during curtailments or outages (see Narrative Summary for specific requirements) without prior approval from Chemistry.

Safety Evaluation Summary:
ECG 17.2 Bases contain equipment configurations for ASW chlorinations that must be met to assure compliance with the NPDES permit. These configurations were established in 1994 based on data available at the time and were as follows:

1. One ASW train (total) may be chlorinated using the continuous chlorination system if two CWPs are in operation.
2. Two ASW trains (total) may be chlorinated using the continuous chlorination system if four CWPs are in operation.
3. Two ASW trains (total) may be chlorinated manually (30 minutes per day) if one CWP is in operation.

Subsequently it has been determined that fewer CWPs need to be operated to maintain the seawater discharge below the NPDES limits while continuously chlorinating the ASW piping. The evaluation justifying the new CWP configurations is documented in Reference 1. The proposed new equipment configurations are:

1. Two ASW trains (total for both Units) may be chlorinated using the continuous chlorination system if two or more CWPs are in operation. Two ASW trains (total for both Units) may be chlorinated manually (30 minutes per day) if one CWP is in operation. (No change has been made to manual chlorination configurations.)

Conclusion:
Weekly chemical analyses have shown that the number of operating CWPs proposed by this revision is sufficient to remain below the NPDES discharge limits. This change will allow more latitude for the chlorination program and still meet NPDES discharge limits.

99-009 Second Level Undervoltage Relay Protection Setpoint Change

Reference Document No.: DCP E-049451
Rev. No: 0
Reference Document Title: Second Level Undervoltage Relay Protection Setpoint Change

Safety Evaluation Description:
The Second Level Undervoltage Relay (SLUR) calibration history uncovered repeated occurrences of the relay's as-found setpoint being outside its Technical Specification Limit of 3785 Volts. Since the as-found setpoints remained above the limiting voltage needed for the operating equipment, these relays remained capable of performing their Licensing Basis Function. The referenced design change is modifying the setpoint for these relays such that DCPP will have the highest confidence that between calibration cycles, no Second Level Undervoltage Relay will exceed its Technical Specification limit.

Safety Evaluation Summary:
The raising of the SLUR setpoint results in earlier detection of a potentially degraded voltage condition. Based on computer simulated transient analyses (ETAP), the new SLUR setpoint may result in an emergency diesel generator start, but will not cause the initiation of a bus load shed, with an operable LTC. This outcome exists under the current SLUR setpoint. This new setpoint was established using ISA 67.04 methodology, which increased the design margin between the SLUR setpoint and the Technical Specification Limit. This methodology is extensively used at DCPP for establishing instrumentation setpoints, and accepted by the NRC. Based on this methodology, the new setpoint ensures that setpoint drift will not exceed the SLUR Technical Specification Value of 3785V.

Part of Design Calculation 359A-DC performed transient analyses of the Startup Transformer's LTC in manual operation for all of the 230kV system configurations shown in Table II of Dispatch Instruction O-23. This design calculation is valid for the current SLUR reset of 93% and the new SLUR reset of 93.75%. The effects of this SLUR setpoint change resulted in three (3) of the six operable line configurations failing to reset the SLURs. Therefore, under manual LTC operation, only the "all lines in" configuration is of Table II will now be capable of resetting the SLURs. Since the "all lines in" configuration is the normal configuration of the 230kV system, manual LTC operation remains viable.

Conclusion:
This setpoint change does not change the circuitry of the SLURs. The SLUR setpoint change results in earlier detection of a degraded voltage condition on the vital 4160V busses. Transient Analyses show that under normal and

accident conditions, the SLURs perform their function in accordance with the Design and Licensing Basis of DCPP. Therefore, the SLUR setpoint change will prevent the SLURs from exceeding the Technical Specification value between calibration cycles and has no affect on the functions the SLURs perform.

The operability of the offsite 230kV supply source is dependent on the operability of the Startup Transformer's LTC or all 230kV lines inservice. Any manual operational status of the LTC in conjunction with a 230kV line outage, causes the 230kV system to be inoperable. This condition is acceptable because, cross-ties of the 12kV and 4kV busses are the only plant evolution that places the LTC in manual, which occurs during normal 230kV system configurations (i.e., "all lines in").

99-010 Second Level Undervoltage Relay Protection Setpoint Change

Reference Document No.: DCP E-050451
Rev. No: 0
Reference Document Title: Second Level Undervoltage Relay Protection Setpoint Change

Safety Evaluation Description:
The Second Level Undervoltage Relay (SLUR) calibration history uncovered repeated occurrences of the relay's as-found setpoint being outside its Technical Specification Limit of 3785 Volts. Since the as-found setpoints remained above the limiting voltage needed for the operating equipment, these relays remained capable of performing their Licensing Basis Function. The referenced design change is modifying the setpoint for these relays such that DCPP will have the highest confidence that between calibration cycles, no Second Level Undervoltage Relay will exceed its Technical Specification limit.

Safety Evaluation Summary:
The raising of the SLUR setpoint results in earlier detection of a potentially degraded voltage condition. Based on computer simulated transient analyses (ETAP), the new SLUR setpoint may result in an emergency diesel generator start, but will not cause the initiation of a bus load shed, with an operable LTC. This outcome exists under the current SLUR setpoint. This new setpoint was established using ISA 67.04 methodology, which increased the design margin between the SLUR setpoint and the Technical Specification Limit. This methodology is extensively used at DCPP for establishing instrumentation setpoints, and accepted by the NRC. Based on this methodology, the new setpoint ensures that setpoint drift will not exceed the SLUR Technical Specification Value of 3785V.

Part of Design Calculation 359A-DC performed transient analyses of the Startup Transformer's LTC in manual operation for all of the 230kV system

configurations shown in Table II of Dispatch Instruction O-23. This design calculation is valid for the current SLUR reset of 93% and the new SLUR reset of 93.75%. The effects of this SLUR setpoint change resulted in three (3) of the six operable line configurations failing to reset the SLURs. Therefore, under manual LTC operation, only the "all lines in" configuration is of Table II will now be capable of resetting the SLURs. Since the "all lines in" configuration is the normal configuration of the 230kV system, manual LTC operation remains viable.

Conclusion:
This setpoint change does not change the circuitry of the SLURs. The SLUR setpoint change results in earlier detection of a degraded voltage condition on the vital 4160V busses. Transient Analyses show that under normal and accident conditions, the SLURs perform their function in accordance with the Design and Licensing Basis of DCPP. Therefore, the SLUR setpoint change will prevent the SLURs from exceeding the Technical Specification value between calibration cycles and has no affect on the functions the SLURs perform.

The operability of the offsite 230kV supply source is dependent on the operability of the Startup Transformer's LTC or all 230kV lines inservice. Any manual operational status of the LTC in conjunction with a 230kV line outage, causes the 230kV system to be inoperable. This condition is acceptable because, cross-ties of the 12kV and 4kV busses are the only plant evolution that places the LTC in manual, which occurs during normal 230kV system configurations (i.e., "all lines in").

99-011 Insitu Pressure Testing of Steam Generator Tubing

Reference Document No.: FTI 1268592A
Rev. No: 4
Reference Document Title: Field Procedure for Insitu Tube Pressure Test
Safety Evaluation Description:
Insitu pressure testing of degraded steam generator tubes (as detected by eddy current testing) may be performed to verify licensing basis leakage and structural integrity of the tubing, in support of meeting NEI 97-06 performance criteria, and in support of alternate repair criteria. Insitu testing has not been performed at DCPP prior to 1R9.

Safety Evaluation Summary:
Degraded tubes that are insitu pressure tested at structural integrity pressure limits must be removed from service by plugging. Tubes that are insitu pressure tested to steam line break pressure conditions and up to the maximum allowable hydrotest pressure may be returned to service provided that post-test NDE verifies that the Technical Specification repair limits are not

exceeded.

Insitu pressure testing during plant shutdown has no impact on plant operation nor impact on operator actions. Insitu testing is performed to demonstrate that a degraded tube is within the bounds of the licensing basis and, if appropriate, to support implementation of ARC. Insitu testing has no affect on adjacent tubes. Insitu testing helps quantify potential leakage from degraded tubes that could contribute to off site doses following a SLB, thus verifying that licensing basis leakage limits are not exceeded. Tubes that are plugged following insitu testing will no longer contribute to SLB leakage nor have burst potential. Tubes that are returned to service after insitu testing retain an adequate margin of safety to withstand normal and faulted operational loading.

Conclusion:
Insitu pressure testing of degraded steam generator tubes (as detected by eddy current testing) may be performed to verify leakage and structural integrity of the tubing, in support of meeting NEI 97-06 performance criteria, and in support of ARC. Tubes that are plugged following in situ testing will no longer contribute to SLB leakage nor have burst potential. Tubes that are returned to service after insitu testing, provided that post-test NDE verifies that the Technical Specification repair limits are not exceeded, retain an adequate margin of safety to withstand normal and faulted operational loading.

99-012 Steam Generator Upper Bundle Hydraulic Cleaning

Reference Document No.: SSS-2.2.2-GEN-14
Rev. No: 0
Reference Document Title: Steam Generator Upper Bundle Hydraulic Cleaning (UBHC) Procedure

Safety Evaluation Description:
This LBIE addresses upper SG tube bundle hydraulic cleaning. Upper bundle hydraulic cleaning is a maintenance process performed to clean the steam generator tube bundle. It is designed to remove deposits that may cause tube support plate blockage and to remove scale and contaminants on tubes. The cleaning is achieved when the reactor is shut down using a high pressure cleaning system installed in the tube lane of the tube bundle and positioned at various elevations in the steam generator by raising it through the flow slots in the tube support plates.

Safety Evaluation Summary:
Use of the Upper Bundle Hydraulic Cleaning system does not require permanent modification of the steam generator tubes, tube supports, or shell. The loads imposed by the deployment of the cleaning system on the steam generator tubes, blowdown pipe, and tube support plates have been evaluated and found to be acceptable (References 2 and 3). To prevent high

loads during off-nominal conditions a load cell is installed at the base of the vertical deployment system to measure the load being applied. Software that controls the operation of the hydraulic pump will shut down the system at a safe load well below the maximum allowable limit.

The tube support plate analysis is performed using a finite element model of the tube support plate. The load applied to the tube support plate is assumed to be due to a stuck tool. For a stuck tool an upward load of several hundred pounds may be applied to the bulk cleaning head, and subsequently transmitted to the tube support plate (TSP). Equivalent plate properties are used in the model to account for the presence of the tube and flow holes. Once stresses are calculated for the equivalent solid plate, stress factors are then applied to calculate the peak stresses in the plate ligaments. The effects of tight tube to plate intersections are conservatively ignored in determining plate stresses for a stuck tool. An analysis addressing the effects of locked tubes produced large stresses in certain TSP ligaments near the wedge locations and a limit on the number of operational cycles. The locations where this cycle limit was calculated are far removed from the locations of the load assumed to be applied by the UBHC tool. The UBHC tool induced stresses at the limiting wedge locations are very small, and can be considered negligible. This means that the stress in the TSP ligaments during cold shutdown will essentially remain unchanged should the tool contact the TSP. Therefore, the loads potentially induced by the UBHC tool will not contribute to a decrease in the allowable number of thermal cycles between cold shutdown and 100% power (NSD-E-SGDA-99-013, Rev. 1). The analysis of the plates also assesses the effect on the stresses of thinned ligaments between tube holes in the vicinity of the flow slots.

The effects of the water jets on the tubes and tube support plates have been evaluated using testing and analysis (NSD-E-SGDA-97-038, NSD-E-CD&ME-98-168m, and NSD-E-SGDA-99-13. The jet velocities expected with the cleaning head are comparable to the velocities permitted for standard sludge lance operations or other steam generator cleaning systems. An evaluation has been made of the effect of potential tube vibration. The evaluation of flow induced vibration considers a jet directly impacting a tube at the midspan between the tube support plates. Considering a direct impact provides a more conservative solution (higher loads) than the intended configuration of a jet directed between the columns of tubes. A jet directed between the columns of tubes will strike the tubes tangentially or obliquely. The maximum applied load to a single tube is calculated to be approximately 10 lbs. Also, this load is not constant, but varies with time both along the tube and in the magnitude of the applied load. Stress in the tube due to an applied load of this magnitude, frequency and location has been determined assuming that the tubes are well supported at the tube support plates due to crevice deposits and denting. The

effect of the dynamically applied load has been determined to be well below the maximum bending stress allowable of $1.5S_m$ (35 ksi) for an operating steam generator. Gross yielding of the tube material will not occur. In addition, the alternating stress in the tube due to UBHC loading is well within the endurance limit of the Alloy 600 material. Therefore, there is no potential for fatigue usage accumulation due to the applied loading. Tube cracks that may be left in service under the alternate repair criteria at the tube support plate locations are axially oriented. Loads from the upper bundle hydraulic cleaning do not tend to open axially oriented cracks. In addition, cracks located within the thickness of the tube support plate are not subject to direct impact of the cleaning jets due to the presence of the tube support plate and the deposits in the crevice between the plate and the tube.

Conclusion:
The use of the Upper Bundle Hydraulic Cleaning system in the Diablo Canyon Units 1 and 2 Steam generators has been evaluated using the criteria of 10 CFR 50.59(a)(2). Based on the evaluation it has been concluded that the plant components and safety systems will not be adversely affected during normal operation and accident conditions by the use of the cleaning system. As such, this situation does not represent an unreviewed safety question.

99-014 Steam Generator Inspections With Changeable Feet Bobbin Probes

Reference Document No.: FTI 51-5003036
Rev. No: 0
Reference Document Title: 50.59 Input For Using Changeable Feet Probes at DCPP

Safety Evaluation Description:
During 1R9 a new type of bobbin probe, the Zetec Changeable Foot Probe, will be used to perform the steam generator eddy current inspection. The eddy current inspection is conducted in accordance with the DCPP Operating License and Technical Specifications. A vendor (FTI) safety evaluation was prepared to document that use of the new probes does not present an unreviewed safety question and does not invalidate the inspection results with respect to the DCPP Operating License or Technical Specifications.

Safety Evaluation Summary:
As documented in vendor safety evaluation FTI 51-5003036-00, detailed testing and certification by the probe manufacturer (Zetec) shows that changing the wearable feet on this type of bobbin probe does not affect the performance of the probes when measured in a calibration standard at the manufacturers facility. Changeable Feet Probes are certified to pass the new probe variability criteria. Changing the feet in the field will restore the probe centering capability. A check of the signal-to-noise ratio performance of the probe in the field after the feet are changed will verify that the new probe/feet

combination has restored the probe to a "new" condition. Probe wear criteria will be monitored and met for each new probe/feet combination used for the inspection.

Conclusion:
Use of the new Zetec Changeable Foot Probe during DCPP steam generator eddy current inspections has been evaluated using the criteria of 10 CFR 50.59. Based on this evaluation, it is concluded that use of these probes will not affect the in-service inspection results, does not present an unreviewed safety question, and will not violate the DCPP Operating License or Technical Specifications. Implementation of the voltage based repair criteria for ODSCC as described in the Technical Specifications will not be affected.

99-015 Reactor Upper Internals Guide Tube Support Pin Replacement

Reference Document No.: DCP N-049449
Rev. No: 0
Reference Document Title: Reactor Upper Internals Guide Tube Support Pin Replacement

Safety Evaluation Description:
The reference document above proposes to replace Unit 1 second-generation "Modified Rev. A" (2GMRA), Inconel 750 reactor vessel upper internals guide tube support pins with new third-generation pins developed by Westinghouse. The new pins are made of a different material, cold-worked 316 stainless steel (CW316SS), and modified design and stress distribution such that they are highly resistant to stress corrosion cracking (SCC).

In the course of replacing the pins, it is necessary to replace all the guide tube hold down cap screws with a slightly different yet functionally equivalent design because there are no like-for-like replacement guide tube hold down cap screws available. In addition, if the locking cap which retains the guide tube hold down cap screw becomes damaged, a contingency tandem locking cap is employed to secure adjacent cap screws with a single device. Replacement of the pins is due to late 1990's industry operating experience showing crack indications and/or failures of the 2GMRA pins in both foreign and domestic plants. Replacement is a prudent measure to reduce the risk of a forced outage due to the potential for failed pin loose parts to damage the steam generator internals.

Safety Evaluation Summary:
There are no impacts on the safety and licensing basis of DCPP. The functioning of the new pin is transparent to plant operation. The new pins provide the same function as the existing pin with improvements in material and slight changes in configuration to increase the stress margin and extend

the life of the pin. Implementation of the proposed activity removes the uncertainty of a forced outage due to the potential for failed pin loose parts to damage the steam generator internals. There is more industry operating history on the performance of 316 stainless steels material in similar reactor environments. Thus, there is more confidence that the material and design features of the pin will reduce its susceptibility to SCC.

Conclusion:
This LBIE concludes that replacement of the guide tube support pin is acceptable and does not create an unreviewed safety question. The support pin hardware is composed of NRC approved materials, and its design is a third generation of the original which improves its resistance to SCC. All applicable ASME Code margins of safety and fatigue usage factors are met. All Original Equipment Manufacturer design criteria for the pin are satisfied. The replacement support pin, nut, locking cap, guide tube hold down cap screws and tandem locking caps are functionally equivalent to the original equipment and do not change the insertion characteristics of the rod cluster control assemblies.

99-016 FHARE 143 & FSAR Penetration and Fire Stop Commitment Change

Reference Document No.: FHARE 143 & FSAR
Rev. No: 0, 11 & 12
Reference Document Title: Cable Tray Fire Stops
Safety Evaluation Description:
FHARE 143 was written to provide clarification and basis for including only credited cable tray fire stops and excluding non-credited cable tray fire stops from the DCPP surveillance program. Specifically, FHARE 143 and this LBIE evaluates changes being made to Appendix 8.3C-2, revision 11 of the FSAR and Appendix 9.5H-6, revision 12 of the FSAR. Appendix 8.3C-2, revision 11 of the FSAR currently states that "visual inspection of fire stops and seals is performed during each refueling outage". A statement regarding inspection of fire barrier penetration seals and cable tray fire stops is being proposed to be added to Appendix 8.3C-2, revision 11 of the FSAR. The statement is as follows: "Fire barrier penetration seals and credited cable tray fire stops are periodically visually inspected." Also, the statement "credited cable tray fire stops" is being proposed to be added to the definition of fire rated assemblies in Appendix 9.5H-6, revision 12 of the FSAR.

This LBIE evaluation provides the basis for the proposed FSAR wording and commitment change.

Safety Evaluation Summary:
Appendix 8.3C-2, revision 11 of the FSAR states that "visual inspection of fire stops and seals is performed during each refueling outage". AR A0449756

was generated to identify that the cable tray fire stop visual inspection commitment described in the FSAR has not been implemented. An investigation into this commitment concluded that, cable tray fire stops are not inspected on a refueling outage frequency. There is no surveillance test procedure or record of a recurring task work order that implemented this visual inspection of cable tray fire stops. In addition, it was identified that penetration seals are not visually inspected during each refueling outage. Penetration seals have been and will continued to be routinely inspected on an 18 month surveillance frequency. A licensing basis review concluded that cable tray fire stops are specifically credited for separation of certain redundant instrumentation circuitry located in Unit 2 fire area 3-CC (U1 & U2 Containment Penetration areas). The applicable cable tray fire stops will be included in the scope of STP M-70D. Also, cable tray fire stops credited for separation of Unit 1 and 2 pressurizer PORV and auxiliary spray valves circuits located inside containment in fire areas 1-A and 9-A will be added to the scope of STP M-70D. Upon identification of the credited cable tray fire stops a prompt operability assessment was performed and a walkdown to confirm operability was performed on credited fire stops located outside of containment. No degradation was identified.

Conclusion:
Fire barrier penetration seals are inspected on an 18 month frequency and not each refueling outage in accordance with the requirements of ECG 18.7 and STP M-70A. Inspection of penetration seals on an 18 month frequency meets the intent of a refueling outage inspection frequency.

The fire protection licensing basis review concluded that fire stops are specifically credited for separation of certain redundant instrumentation circuitry located in Unit 2, fire area 3-CC. The applicable cable tray fire stops will be included in the scope of STP M-70D. Also, cable tray fire stops credited for separation of Unit 1 and 2 pressurizer PORV and auxiliary spray valves circuits located inside containment will be added to the scope of STP M-70D. Credited cable tray fire stops located outside of containment will also be inspected on an 18 month frequency in accordance with STP M-70D. Credited cable tray fire stops inside containment will be inspected on a minimum 24 month frequency.

All cable tray fire stops installed in the power plant will be continued to be maintained. Degraded cable tray fire stops will be identified in accordance with the DCPP problem identification and resolution program. Cable tray fire stops found to be degraded are repaired in accordance with current cable tray fire stop design requirements. Due to the visibility of cable tray fire stops throughout the power plant area, DCPP fire protection engineering believes that damaged fire stops are identified and repaired in a timely manner.

The existing fire protection program provides an acceptable level of fire safety equivalent to the fire protection level required by 10 CFR 50, Appendix R, Section III.G.2. The FSAR fire hazard analysis (FHA) evaluates the effects of a design basis fire for each safety related fire area/zone. The FHA concluded that the ability to achieve and maintain safe shutdown was not adversely impacted by a design basis fire in a single fire area. Given all of the fire protection features credited in each fire area, DCPP will withstand a design basis fire; therefore, and safe shutdown capability is not comprised. Also, systems required for safe shutdown either have adequate redundancy, adequate fire protection, or manual action can be taken to locally operate equipment required for safe shutdown. Since a design basis fire would be confined and limited to one fire area, redundant safe shutdown would remain available. Therefore, the DCPP fire protection program and safe shutdown capability is not adversely affected by this change.

99-017 Revision 17 of OP B-2:V

Reference Document No.: OP B-2:V
Rev. No: 17
Reference Document Title: RHR - Place In Service During Plant Cooldown
Safety Evaluation Description:
This procedure revision changes the method used to adjust chemistry parameters in the Residual Heat Removal (RHR) loops prior to placing RHR in service for normal shutdown cooling. The revision allows the chemical adjustment to be performed in Modes 1-3. The current revision allows adjustment in Modes 3 or 4, during plant cooldown.

The new method involves a feed and bleed operation during which the Refueling Water Storage Tank (RWST) feeds to the RHR system, which in turn lets down to the Chemical and Volume Control System (CVCS), specifically, the Liquid Holdup Tanks (LHUTs). During the feed and bleed, normal charging is reduced and excess letdown is placed in service to maintain Reactor Coolant Pump (RCP) seal flow and VCT level and temperature. Manual makeup to the RWST is provided by the Reactor Water Makeup System (RWMS) at a sufficient rate to maintain RWST level essentially constant. This feed and bleed will alter the chemical parameters as required by the Chemistry Section.

Safety Evaluation Summary:
This revision changes the method used to adjust chemistry parameters in the RHR loops prior to placing RHR in service, and allows the chemical adjustment to be performed in Modes 1-3.

The operation places the plant in an abnormal alignment for Modes 1-3 that involves a feed and bleed evolution with the RWST feeding to the RHR

system, which in turn lets down to the CVCS, specifically, the LHUTs. During the feed and bleed, normal letdown is isolated, excess letdown is placed in service, and charging flow is reduced to the minimum required for the RCP seals. Manual makeup to the RWST is provided by the RWMS at a sufficient rate to maintain RWST level above that required to ensure an adequate margin of inventory to meet design criteria. RWST level is raised prior to the evolution to ensure adequate inventory exists to meet Design Basis Accident (DBA) requirements. RWST level margin also accounts for operator action times to terminate the evolution in the event of a LOCA. The RHR loops will be flushed sequentially. When both loops meet Chemistry's requirements, makeup to the RWST is terminated and RWST level is allowed to drop to a point at or above its level prior to the evolution. Plant systems are then realigned as dictated by plant conditions or at the Shift Foreman's discretion. While manual makeup to the RWST is in service, the maximum RWST level could impact equipment inside containment due to the changes to flood level and sump pH. Currently, there is sufficient margin in the sump level calculations to provide the required assurance that the assumed addition of 3500 gallons of water is acceptable. The current conditions of the RCS (EOL, essentially no boron) make this issue moot. Sump pH calculations are performed for peak RCS boron concentrations (shortly after BOL). The introduction of approximately 3500 gallons of borated water into the system would be comparable to a low RCS boron concentration, definitely well within the assumptions in the current analysis. Therefore, the effects of an additional 3500 gallons of borated water into the RWST is enveloped by the current analysis.

While manually making up to the RWST, automatic makeup to the RCS is precluded. The procedure addresses maintenance of plant conditions during the evolution. RCS temperature control will be maintained within specifications by rod control.

During RHR pump operation, Component Cooling Water (CCW) is not cut in to the RHR heat exchangers. Engineering Calculation STA-096 determined that the RHR pumps could be operated for 3.3 hours without CCW flow to the RHR heat exchangers and still maintain train temperatures ≤210°F. STA-096 also calculated the feed and bleed flow required to remove the heat generated by an RHR pump, to maintain temperatures such that component integrity is not challenged. The minimum flow required to maintain RHR loop temperature ≤210°F was calculated to be 8.5gpm. Since RHR letdown flow will be approximately 95gpm, RHR temperature will be significantly lower than 210°F (per STA-096, temperature should not exceed 90°F).

Should a Safety Injection signal occur at any time during the evolution, the *dedicated operator* is directed in this procedure to terminate RHR letdown

immediately.

The normal letdown system (via CVCS-8149s and regenerative heat exchanger) will be isolated during this evolution. Normal letdown provides the means to bleed the RCS for the purposes of inventory and chemical control. The normal letdown system receives an automatic valve isolation (LCV-459 & 460 and CVCS-8149s) on low pressurizer level of 17%. This is to protect the pressurizer heaters from becoming uncovered. This protection is a control function and is not credited as a safeguards function.

With excess letdown in service to support this evolution, RCP seal water injection flow will continue to be provided. The seal water flows will be maintained above the minimum required to ensure proper seal lubrication and cooling. The RCS letdown flow through the excess letdown heat exchanger is cooled to a temperature below the boiling temperature of water at the reduced system pressure downstream of HCV-123. Letdown flow is further cooled in the seal water heat exchanger to normal charging pump suction temperature. Therefore, operation of the plant using excess letdown is an acceptable operating configuration and the portion of CVCS required for this mode of operation is neither safety-related nor required for the mitigation of any accident.

Conclusion:
Components being used for this evolution may be called upon during a DBA. An evaluation of the manual operator action required to terminate the evolution determined that the minimum RWST and sump levels required for a DBA would be maintained and would not be challenged by the flow to the LHUTs. Therefore, sump level calculations are not compromised. The time assumed for operator action to isolate RHR letdown flow, per the guidance in ANSI/ANS-58.8-1994, is extremely conservative. Emergency boration capability could be re-established. Therefore, all systems, structures, and components (SSCs) involved would be available to perform their intended safety functions.

The proposed changes do not result in an unreviewed safety question.

99-018 Revision of MP M-50.8, "Internals Lifting Device Handling"

Reference Document No.: MP M-50.8
Rev. No: 6
Reference Document Title: Installation And Removal Of The Upper And Lower Internals Lifting Device

Safety Evaluation Description:
The Procedure is being revised to provide detailed instructions for the safe handling of the unloaded Internals Lifting Device over the Rector Vessel

Exclusion Area's of Unit 1 and Unit 2 to facilitate reactor maintenance during unit outages.

Safety Evaluation Summary:
The only postulated accident associated with the handling of the unloaded lifting device is an accidental load drop impacting irradiated fuel in the Reactor Vessel. This accident is discussed in Section 15.4.5 of the Updated FSAR, "Fuel Handling Accident." In this section the accidental dropping of the upper internals package is postulated. This accident was analyzed and shown to cause no fuel damage due to the internals support ledge effectively breaking the fall of the of the upper internals.

Because of the physical size of the internals lifting device it would also impact the internals support ledge and be precluded from impacting irradiated fuel in the reactor vessel. However, it is highly unlikely that the unloaded internals lifting device would be dropped into the reactor vessel during the proposed lifting operation because the following specific requirements are included in the revised procedure:

1. Only the polar crane main hoist may be used for this operation. The factor of safety (to yield) for the unloaded lifting device is 57.1, compared to 5.5 for lifting device loaded with the upper internals package. This considers stresses in the polar crane reeving and the most highly stressed lifting device component.
2. The main hoist load brakes will be tested in a safe area prior to traversing over the reactor vessel exclusion area.
3. No raising or lowering of the load will be allowed while in the reactor vessel exclusion area.
4. Both upper travel limit switches will be tested and verified operable immediately prior to attaching the lifting device to the main hoist load block.
5. The intended load path will be visually inspected and any objects that may cause a load hang-up will be removed prior to proceeding with the lifting and transport of the internals lifting device.
6. A load cell will be used to continuously monitor the weight of the internals lifting device during lifting and transport.

Conclusion:
The above described specific and general requirements combine to provide a defense in depth approach to ensure that the internals lifting device is handled safely and to minimize the occurrence of the principal causes of load handling accidents. This activity does not result in an unreviewed safety question.

99-019 Westinghouse Procedures for 1R9 Split Pin Equipment Assembly, Installation, Testing and Removal

Reference Document No.: Multiple
Rev. No:
Reference Document Title: See Attached Multi-Procedure Change list
Safety Evaluation Description:
This LBIE supports three specific Westinghouse Procedures required for the 1R9 Guide Tube Support Pin (Split Pin) replacement:

1. MSI 2.8.1 PGE-1, Rev. 0 Installation and Removal of CRDM Drive Shaft Storage Rack at Diablo Canyon Unit 1
2. MSI 2.8.1 PGE-2, Rev. 1 Upper Internals Work Platform Assembly and Disassembly
3. MSI 2.8.1 PGE-3, Rev. 1 Jib Crane, Storage and Mini-Station Assembly and Disassembly at Diablo Canyon Unit 1

These procedures direct assembly, installation, and use of equipment required for the Split Pin replacement. This equipment includes large, heavy components, some of which are heavy loads (either separately or once assembled). This equipment will be installed in and around the refueling cavity. This equipment and will be in place during the reactor disassembly sequence, during the presence and movement of fuel in the reactor and refueling cavity, and during "No-Mode" conditions, with all fuel removed from containment.

The CRDM Drive Shaft Storage Rack will be assembled and installed in the Refueling Cavity prior to flooding the refueling cavity, and will be removed following draining of the cavity.

The Upper Internals Work Platform will be installed on the upper internals during the reactor disassembly and refueling sequence.

The Jib Crane and Mini-Stations will be mounted on the 140' deck at the cavity edge. The Mini-Stations are suspended from the 140' deck, and hang into the refueling cavity, and are used as personnel and tooling platforms over the flooded cavity. The Jib Crane and Mini-Stations will be installed following fuel off-load and will be removed prior to fuel reload.

The Guide Tube Support Pin Replacement will be performed during the refueling sequence, through the core offload window, and potentially into the core reload window. The Westinghouse equipment described in these procedures will be removed prior to Mode 4.

Safety Evaluation Summary:
Heavy Loads Issues: There are no specific discussions of heavy loads movements in the FSAR that would be affected by these procedures. The equipment will be rigged in accordance with existing plant procedures. The loads will be moved in accordance with MA1.ID14, and other plant procedures. All movement will be in accordance with procedures and the PG&E Heavy Loads program as reviewed and approved by the NRC, therefore there are no unreviewed safety issues.

Seismic Qualification: There are no specific analyses described in the FSAR that would be affected by the equipment covered in these procedures. The affected structures are the 140' deck, the refueling cavity and liner, the reactor head stand, the reactor upper internals, and the reactor lower internals stand. According to the Q List, the refueling cavity liner, head stand, the upper internals stand, and the lower internals stand are Quality Class II structures.

Each temporary addition was evaluated separately.

- Drive Shaft Storage Rack (DSSR) - Floor loading was evaluated and it was determined to be acceptable because the loaded DSSR is less load than the lower internals structure. Therefore there are no floor loading concerns. The DSSR will be present in the lower cavity when fuel is in the vessel and during fuel movement, therefore it needs to be seismically secured. Seismic tie downs have been specified (AR A0464796, AE #10), and the Westinghouse procedure requires their installation. The DSSR will be secured to the lower internals stand. Reference Calculation EQP-916.

- Upper Internals Work Platform - Floor loading has been evaluated. It was determined that the Work Platform is less load than the Upper Internals Lift Rig, which has been previously analyzed. (Calculation No. 1.284.2 Set 1). Since the Work Platform will be present during the time that fuel is present in the vessel, it needs to be seismically secured. Seismic tie downs have been specified and their installation is required by the Westinghouse procedure. Calculations EQP-919 and 1.284.2 Set 1 have been performed to specify adequate seismic restraints, and these restraints have been specified by AT MM A0469553.

- Jib Crane and Mini-Stations - This equipment will be installed following completion of fuel offload, and will be removed prior to reload. Therefore the Jib Crane and Mini-Stations are not required to be seismically restrained, and must be analyzed for floor loading only. Floor loading was reviewed and determined to be acceptable. The Westinghouse procedure specifies the required mounting, and this has been analyzed and determined to be acceptable (A0464796, AE #16,

Calculation EQP-920).

Fuel Handling Accident: The temporary equipment installed in these procedures does not handle or come into contact with the fuel. The DSSR and the Work Platform, which will be installed when fuel is present, will be secured in a manner such that they do not interfere with fuel handling equipment. The Work Platform has removable sections for transfer of guide tubes. These sections are procedurally prohibited from being removed when there is fuel present (MSI 2.8.1 PGE-2, Caution in Step 9.3). The Work Platform is capable being located such that the eastern-most corner of the work platform bridge can come into contact with the manipulator crane over the western-most fuel assemblies. In order to ensure that the Work Platform Bridge does not interfere with the manipulator crane, temporary rail stops will be placed to restrict the Work Platform when refueling is in progress (MSI 2.8.1 PGE-2, Steps 9.2.9 and 9.3, Note 5). The procedures and specified seismic restraints ensure that the equipment will not impact fuel or fuel handling equipment in a seismic event. The Jib Crane and Mini-Stations, which are not seismically restrained, cannot be installed until the reactor is defueled, since they interfere with the manipulator crane motion. The FSAR describes a fuel overheating event. The added equipment in the refueling cavity will displace some water, however the volume of displaced water is minimal, and the effect on cooling is judged to be insignificant. There is no equipment added to the reactor vessel, and therefore there is no effect on the "time to boil" calculated in plant procedures.

Conclusion:
These procedures direct installation of temporary equipment for the 1R9 Guide Tube Support Pin (Split Pin) Replacement. The temporary equipment to be installed has been evaluated for effects appropriate to the applicable modes. The equipment has been designed to minimize effects on fuel handling and fuel handling equipment. Where potential interferences exist, procedural controls and hardware stops are provided to ensure that equipment is safely controlled. Required seismic restraints have been designed and analyzed. The procedures for installation of the temporary equipment specify appropriate controls for the installation of the equipment and the installation of the required seismic restraints. There is no impact to any analysis specified in the FSAR. All rigging and suspended loads will be performed in accordance with the NRC approved PG&E Heavy Loads Program and implementing procedures.

99-020 Concrete Equipment Hatches

Reference Document No.: FHARE-14
Rev. No: 4
Reference Document Title: Concrete Equipment Hatches

Safety Evaluation Description:
This LBIE is being generated to document the acceptability of having unsealed gaps around concrete equipment hatches in rated fire barriers located in fire areas/zones 15, 14-D, 18, 19-D, 30-A-1, 30-A-2, 30-A-3, and 30-A-4. FHARE 14 provides the technical basis of the acceptability of the lesser rated configuration.

Safety Evaluation Summary:
The hatches for the fire areas/zones described above consist of one or more concrete sections used to close off the hatch opening. The gaps (2-3/8" maximum) around the removable hatches sections do not have astragals installed and are not sealed with an approved fire rated material to maintain the rating of the fire area boundaries. FHARE 14 evaluated the affects of the gaps on the DCPP fire protection program, and concluded the ability to achieve and maintain safe shutdown was not adversely impacted. This LBIE supports the FHARE and the associated FSAR change.

Conclusion:
The concrete hatches provide adequate separation between the subject fire areas and safe shutdown will not be compromised by the presence of these unsealed hatch gaps due to the automatic and manual fire protection features, lack of continuity of combustibles, tortuous path of travel for any hot gases. The lack of safe shutdown equipment in the vicinity or the absence of combustible material nearby further assure the capability to achieve safe shutdown is not affected.

99-021 Electrical Drawing 050029 Drawing Revision

Reference Document No.: DCP M-049476
Rev. No: 0
Reference Document Title: Cable Tray Fire Stop Drawing Revision
Safety Evaluation Description:
This is a drawing change only design change that adds a note to Electrical drawing 050029, Sheet 81 to allow the installation of the cable tray fire stops inside the containment crane wall, the RCP cubicles, and the pressurizer cubicle to be optional. The cable tray fire stops in the containment 91' annular area that are specifically credited will still be required to be installed. The note added by this DCP will also state the fire stops are not to be installed in any HELB impingement zones unless approved by engineering. The need for the drawing revisions was determined to be necessary due to the potential of cable tray fire stop materials to clog the containment recirculation sump (Reference INPO Operating Experience Report 9307). This change will also add a reference in the FSAR, Appendix 9.5B, Table B-1, Section 3, "Electric Cable Construction, Cable Tray, and Cable Penetrations" to this design change. This section of the FSAR describes the installation details of cable

tray fire stops.

Safety Evaluation Summary:
This drawing change only DCP makes the installation cable tray fire stops inside containment fire zones 1-B, 1-C, 9-B and 9-C (inside the containment crane wall, the pressurizer and RCP cubicles, and excluding the annular area) optional, and prohibits their installation in a HELB impingement zone, unless approved by engineering. The need for the drawing revisions was determined to be necessary due to the potential of cable tray fire stop materials to clog the containment recirculation sump (Reference AR A0470342 and INPO Operating Experience Report 9307). This change will also add a reference to this design change to the FSAR, Appendix 9.5B, Table B-1, Section 3, "Electric Cable Construction, Cable Tray, and Cable Penetrations." This section describes the installation details of cable tray fire stops.

Conclusion:
The fire protection licensing basis review concluded that the fire stops that are specifically credited for separation of certain redundant instrumentation circuitry are located in the Unit 1 containment annular area below the 117' elevation of Unit 1, and the containment penetration area and the annular area of the containment below the 117' elevation of Unit 2. These credited fire stops will remain in place and be inspected by STP M-70D. Allowing the installation and or removal of the cable tray fire stops inside the containment crane wall, the RCP cubicles, and the Pressurizer cubicle will not affect the ability to achieve and maintain safe shutdown.

The cable tray fire stops located in these areas are part of the DCPP defense-in-depth fire protection program and removal of cable tray fire stops will decrease the overall fire protection level currently in place. Nevertheless, the existing fire protection program provides an acceptable level of fire safety equivalent to the fire protection level required by 10 CFR 50, Appendix R, section III.G.2. The FSAR fire hazard analysis (FHA) evaluates the effects of a design basis fire for each safety related fire area/zone. The FHA concluded that the ability to achieve and maintain safe shutdown was not adversely impacted by a design basis fire in a single fire area. Given all of the fire protection features credited in each fire area, DCPP will be able to withstand a design basis fire, and therefore, safe shutdown capability is not comprised. Also, systems required for safe shutdown either have adequate redundancy, adequate fire protection, or manual action can be taken to locally operate equipment required for safe shutdown. Since a design basis fire would be confined and limited to one fire area, redundant safe shutdown would remain available. Therefore, the DCPP fire protection program and safe shutdown capability is not adversely affected by this change.

99-022 Change to Generic Letter Commitment to Plug PWSCC Cracks

Reference Document No.: AR A0475922
Rev. No: 0
Reference Document Title: Change to Generic Letter Commitment to Plug All Cracks

Safety Evaluation Description:
PG&E has developed and validated an eddy current sizing technique that will allow accurate measurement of PWSCC axial crack dimensions. PG&E has presented the results of this development to the NRC and has received NRC agreement that the technique can be used to size PWSCC cracks and leave them in service if maximum depth is <40% through wall. As part of this development and presentation to the NRC it was shown that the uncertainty and growth associated with <40% through wall PWSCC cracks will not challenge leakage or structural integrity over the next cycle. PG&E intends to implement this sizing technique in 1R9.

Safety Evaluation Summary:
As cracking type degradation has developed in the industry, the NRC has generally not agreed to leaving crack-like degradation in service under this TS because of a concern about the capability of eddy current testing to accurately size crack dimensions. There has been much discussion in the industry and with the NRC about the requirements to adequately qualify an eddy current sizing method which could be used to size cracks and allow them to remain in service. The industry developed the Appendix H Peer Review program to determine technique uncertainty for sizing cracks. During the same period, the NRC staff developed and issued draft Regulatory Guide (RG) DG 1074 which contains guidance on how to qualify (called "validate" by DG 1074) an eddy current technique in order to use it to leave cracks in service.

Working with TVA, PG&E developed a sizing technique for axial PWSCC cracks at dented tube/tube support plate (TSP) intersections in steam generators. This development program used 38 laboratory specimens with 51 crack indications and 3 DCPP pulled tubes with 4 crack indications. This technique was submitted to the EPRI PWR Seam Generator Examination Guidelines, Revision 5, Appendix H Peer Review process, to establish the technique uncertainty for length, average depth, and maximum depth. These were published by EPRI as Equipment Specification Technique Sheet (ETSS) 96703.

The sizing technique was also "validated" using a performance demonstration as outlined in NRC draft Regulatory Guide DG 1074. PG&E and TVA conducted a performance demonstration with teams of NDE analysts at the Westinghouse Waltz Mill site to "validate" the sizing technique system uncertainty (system uncertainty includes both technique and analyst

uncertainty) in accordance with the NRC draft Regulatory Guide DG 1074. The performance demonstration was conducted because of ongoing uncertainty within the industry about NRC requirements to validate NDE technology to leave indications in service. The performance demonstration was conducted in accordance with the NRC's own guidance. The NRC staff was invited to attend and observe the performance demonstration - which they did. As part of the performance demonstration, PG&E and TVA historical eddy current PWSCC data was analyzed to determine PWSCC growth rates.

Implementation of a sizing technique for PWSCC cracks in order to leave degradation in service if sized less than the Technical Specification plugging limit of 40% through wall is similar to use of sizing techniques to leave cold leg thinning (CLT) and anti-vibration bar (AVB) wear scars in service.

On September 1, 1998, PG&E, TVA and Westinghouse met with the NRC to present a technical justification for leaving PWSCC cracks in service up to 40% average depth using the validated PWSCC sizing technique. This presentation also gave the technical justification that structural and leakage integrity would be maintained throughout the cycle. The NRC did not agree with an average depth approach. To use average depth as a plugging limit will require submittal and approval of an alternate repair criteria license amendment (WCAP-15128 is being developed for this submittal). Rather, the NRC staff stated that they agreed with the method used to validate the sizing technique and also agreed that it could be applied to leave PWSCC cracks in service if <40% maximum depth under the current DCPP and Sequoyah Technical Specifications with no change to the operating license.

Draft WCAP 15128 and draft Westinghouse NDE Qualification Report SG-99-01-004 all support use of the validated sizing technique and a repair limit of 40% through wall for PWSCC axial cracks at dented tube/TSP intersections. At this repair limit a crack will not grow to challenge burst or leakage integrity even when applying a 95% confidence level growth and 95% confidence level NDE uncertainty to the next cycle.

Conclusion:
Implementing the 40% Technical Specification plugging limit for axial PWSCC cracks at TSP intersections is understood and has been agreed to by the NRC staff. It does not result in an unreviewed safety question and does not require any change to the Technical Specifications.

99-023 Unit 2 Condensate Storage Tank Level Taps for LT 40/44

Reference Document No.: DCP 50478
Rev. No: 0
Reference Document Title: Condensate Storage Tank Level Taps for LT-40/44

Safety Evaluation Description:
This DCP relocates and modifies the lower instrumentation tap for Condensate Storage Tank level transmitters LT-40 and LT-44. The process for installation of the new lower taps in new locations involves a machining process (Hot Tapping) to allow penetration of the tank wall without draining the tank. Abandoning the old tap location involves a drilling and tapping process so a threaded plug can be installed. These processes have the potential for introducing small amounts of drilling debris into the Condensate Storage Tank.

Safety Evaluation Summary:
The new level taps for LT-40 and 44 will provide the same function, characteristics, and level range as the original level taps. During the "Hot Tap" process there is the possibility of some small amount of drilling debris entering the Condensate Storage Tank. The DCP evaluated this consideration and concluded that because of the physical configuration of the tank nozzle for the AFW suction and the insignificant shear force from the flow, that any debris will remain on the tank floor and not enter the AFW system.

Conclusion:
The activity describe in this design change will not result an unreviewed safety question.

99-024 DCPP Unit 1 Cycle 10 Reactor Core Fuel Load and Core Operating Limits Report

Reference Document No.: DCP N-049443
Rev. No: 0
Reference Document Title: DCPP Unit 1 Cycle 10 Reactor Core Fuel Load and Core Operating Limits Report

Safety Evaluation Description:
This Design Change incorporates the new fuel loading pattern into the reactor core design for Cycle 10. The new core is required to maintain continued power operation for the next operating cycle. New Fuel is introduced into the core which has slight design changes from the previously installed fuel. The mechanical design of the fresh Region 12A and 12B fuel assemblies is the same as the Region 11A and 11B VANTAGE+ fuel assemblies with the addition of a new bottom grid insert, implementation of the 1.25X IFBA Coating, casting of the bottom nozzle, grid strap fabrication process change and IFM grid assembly inner strap notch and sleeve diameter increase.

Concurrent with this fuel loading pattern, the thermal-hydraulic performance design requires a revisions to the Core Operating Limits Report for Unit 1 Cycle 10 and Technical Specification Bases for DNBR design limits which accurately reflect Improved Thermal Design Procedure (ITDP) and WRB-2.

Safety Evaluation Summary:
The Cycle 10 reload core design meets all the design criteria for maintaining its design basis function. The potential for the pellet-clad gap to reopen later in cycle has been justified as acceptable and will not increase the probability of any accident. Furthermore, adherence to design criteria and manufacturing standards precludes other components and systems from increasing the probability of analyzed accidents.

The reload core design meets the requirements for applicable design criteria or has been justified as acceptable and ensures that no new risk to components or systems occurs which could prevent those systems from mitigating the consequences of evaluated accidents or adversely affect the integrity of the fuel cladding as a fission product barrier. The features implemented in this design are similar to those implemented in other DCPP core designs. These items have no impact on the chemical, physical or mechanical properties of the core which causes it to increase the consequences of any accident in the SAR. Also, the previously evaluated radiological source term bound the new reload core design.

Conclusion:
The evaluation performed in the Westinghouse RSE for Cycle 10 and the additional evaluations performed as apart of this design change insure that the Nuclear Fuel for Cycle 10 is designed in accordance with the proper licensing and design bases and that no impact on nuclear safe5ty results from it implementation. This activity does not result in an unreviewed safety question.

99-025 Steam Generator Mechanical Plugging (FTI)

Reference Document No.: MMP M000043-1
Rev. No: 3
Reference Document Title: Steam Generator Mechanical Plugging (FTI)
Safety Evaluation Description:
This LBIE addresses mechanical plugging of steam generator tubes which have been identified for removal from service. The change consists of installing mechanical rolled plugs at the steam generator tubesheet. The integrity of the RCS and of the Steam Generators is maintained; there is no change in design or function of the Steam Generators caused by the installation of these plugs.
Revision 3 to this document expands the use of the mechanical plugs to the SG 1-1 implant tubes made of Inconel 690 in addition to the standard SG tubes made of Inconel 600. Aside from an acceptable increase in leak rate caused by the less tight fit of the 690 plugs in 690 tubes at operating temperature, the design is unchanged from the previous.

Revision 2 to this document was issued to (1) incorporate use of long rolled plugs, (2) add reference to latest revision of the supplier 50.59, and (3) use latest approved LBIE screen which specifically addresses vendor supplied 50.59 evaluations.

Safety Evaluation Summary:
This design change authorizes the mechanical plugging of steam generator tubes which have been identified for removal from service. The change consists of installing mechanical rolled plugs at the steam generator tubesheet. The integrity of the RCS and of the Steam Generators is maintained; there is no change in design or function of the Steam Generators caused by the installation of these plugs.

The rolled plug is fabricated from thermally treated Inconel 690 material. This material is being used in preference to Inconel 600 type tube plugs since Inconel 690 exhibits superior resistance to PWSCC. The plug is inserted into the tube end and mechanically expanded into the tube by a rolling expansion process. The plug is designed to be removable, and it limits tube leakage at operating temperature to an average of 0.0048 cubic inches/hr. when inserted in Inconel 600 tubes and 0.0129 cubic inches/hour when installed in Inconel 690 tubes. In some instances, a "long" rolled plug may be required to ensure proper roll expansion, e.g. in the case of replugging after the removal of a previously-installed plug. The tube plug has also been evaluated in accordance with ASME Section III rules to verify the structural suitability for the applicable design and operating conditions of the plant.

Conclusion:
The installation of mechanical tube plugs in the Diablo Canyon Units 1 and 2 Steam generators has been evaluated using the criteria of 10 CFR 50.59(a)(2). Based on the evaluation it has been concluded that the plant components and safety systems will not be adversely affected during normal operation and accident conditions by the use of the cleaning system. As such, this situation does not represent an unreviewed safety question.

99-026 Steam Generator Tubesheet Stabilizer (FTI)

Reference Document No.: M000048-1
Rev. No: 3
Reference Document Title: Steam Generator Tubesheet Stabilizer (FTI)
Safety Evaluation Description:
This LBIE addresses a revision to MMP M000048-1 which broadens its scope to include two configurations of the tubesheet cable stabilizer previously addressed in Revision 2: 1) the original configuration to be used in conjunction with the rolled mechanical plug (MMP M000043-1), and 2) the new configuration to be used with the welded plug (MMP M000044-1).

Safety Evaluation Summary:
Removing a tube from service is accomplished via a mechanical rolled plug or welded plug. Installation of a tubesheet stabilizer is conjunction with a plug is determined by engineering assessment. The stabilizer is a stainless steel cable installed within the tube, designed primarily to control flow-induced vibration and prevent tube severance, and secondarily to limit movement of a severed tube should the tube defect propagate into a full sever. By limiting movement, the possibility of a severed tube impacting an adjacent in-service tube is eliminated, thus precluding an in-service leak via this mechanism. These statements are supported by fluid-elastic stability margin (FSM) calculations.

Two separate but similar stabilizer designs are addressed:

1. The tube cable stabilizer, FTI part number 124705-001, previously addressed in MMP M000048-1, Revisions 1 and 2, and
2. The implant tube cable stabilizer, FTI part number 124705-001M, added in this Revision 3.

Stabilizer Failure Modes
The stabilizer design is one that may be installed without causing further damage to degraded tubes. Failure of a stabilizer within a severed tube could ultimately lead to the failure of an adjacent tube and the release of reactor coolant fluid to the secondary system. In order for this to occur, the stabilizer cable would have to wear significantly—to about 1/3 of its installed diameter—within the severed tube for contact to occur with an adjacent tube. This contact would allow wearing of the contacted tube OD, which would be detectable in outage eddy current exams, or by operational leakage which is monitored.

Design Qualification
The stabilizer is classified as a non-pressure boundary, non-Code, safety-related component. Fluid-elastic stability margins (FSM) and turbulence responses were calculated. The loss of stiffness due to possible tube severing and a pinned joint forming at the top of the TS and at TSP 1 were considered for tubes at all locations, including peripheral tubes subject to the highest cross-flows. The reaction forces between the tube and TSP and between the tube and tubesheet are acceptable. The stabilizer design is qualified (i.e., prevents contact with adjacent tubes) during the DE, DDE and Hosgri seismic events. The added mass (stabilizing 10% of total steam generator tubes based on plugging margin) of the stabilizers does not have a significant effect on steam generator seismic loadings. The stabilized tube is qualified for the Main Steam Line Break (MSLB) depressurization event. Tube locking in TSPs has been considered in the qualification. Stabilizer materials are appropriate for use in the SG.

Conclusion:
The installation of the FTI tube cable stabilizers in the Diablo Canyon Units 1 and 2 Steam generators has been evaluated using the criteria of 10 CFR 50.59(a)(2). Based on this evaluation it has been concluded that the plant components and safety systems will not be adversely affected during normal operation and accident conditions by the use of either cable stabilizer described above. Thus there is no unreviewed safety question created.

99-027 ECG 45.3 Revision

Reference Document No.: ECG 45.3
Rev. No: 1
Reference Document Title: Electrical Power Systems - Containment Penetration Conductor Overcurrent Protection Devices

Safety Evaluation Description:
This revision to ECG 45.3 will make the following changes:

- Change the surveillance frequency for performing a functional test on each representative sample from 18 months to 24 months to support extended fuel cycles.

- Add a requirement that all devices shall be tested on a rotating basis of 48 months for all 12kV and "drawout" breakers, and 96 months for all molded case circuit breakers and thermal overload devices.

- Increase the sample size percentage to be surveilled at each refueling outage from 10% to 25% for 12 kV and "drawout" type breakers, and from 10% to 15% for molded case circuit breakers and thermal overloads.

- Change "CHANNEL CALIBRATION" to "function testing shall verify that the protection device operates within the withstand capability of the associated penetration conductor" to more accurately describe the purpose of the SR.

- Add a statement that protection devices found inoperable shall be restored to OPERABLE status prior to re-energizing its associated circuit.

Change specific testing methods for molded case circuit breakers from:

- "200% (for D.C. breakers) and 300% (for A.C. breakers) of the pickup of the time delay element and verifying that the circuit breaker operates within the time delay band for that current specified by the manufacturer. The instantaneous element of molded case circuit breakers shall be tested by injecting a current equal to -25%, +40% of the pickup value of the element and verifying that the circuit breaker trips with no intentional time delay." to "function testing shall verify that

the protection device operates within the withstand capability of the associated penetration conductor," to more accurately describe the purpose of the SR.

- Modify the 60 month SR for circuit breaker inspection and preventive maintenance to replace the term "circuit breaker" with "penetration protective device."

Safety Evaluation Summary:
The containment penetration conductor overcurrent protective devices for Class 1E and non Class IE electrical conductors penetrating the reactor containment, such as primary and backup breakers or fuses, provide redundant overcurrent protection to ensure containment integrity by keeping the fault current within the withstand capability of the associated penetration conductor. The penetration conductor overcurrent protective devices open the circuit to ensure a short circuit electrical fault will not damage the conductor within the penetration canister in such a manner to cause a breach in the containment seal.

Containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or by demonstrating the operability of primary and backup overcurrent protective devices through periodic surveillance.

The effect of the changes proposed for Revision 1 to ECG 45.3 are summarized as follows:

1. The surveillance frequency for testing the overcurrent devices is changed from 18 months to 24 months to accommodate extended refueling cycles. Reviews of operational, maintenance and surveillance test histories show no time dependent failure mechanisms that would impact the reliability of the overcurrent devices by changing the surveillance frequency from 18 months to 24 months.

2. To compensate for the extended surveillance frequency, the minimum sample size required to be tested at each refueling outage has been increased from 10% to 25% for 12 kV and "drawout" type breakers, and from 10% to 15% for molded case circuit breakers and thermal overloads.

3. A requirement was added to test all the overcurrent devices on a rotating basis over either a 48 month or 96 month period, depending on the type of device, to assure each device receives periodic testing. The increase in sample size and the implementation of a rotating test

schedule over either 48 or 96 months results in testing each overcurrent device more frequently than had been required under TS 3/4.8.4.2.

4. SRs 45.3.1, 45.3.2, 45.3.3, 45.3.4, and 45.3.5 have been revised to more accurately describe the purpose of the SRs and to update the testing methods and acceptance criteria based on current industry standards and practices. The revised SRs will maintain compliance with RG 1.63, Revision 1.

5. A requirement has been added that protective devices found inoperable shall be restored to operable status prior to reenergizing their associated circuits.

6. SR 45.3.5, has been revised to replace the term "circuit breaker" with "penetration protective device," because this SR involves more than just circuit breakers. This is an administrative change only.

Conclusion:
The overcurrent protective devices respond to fault conditions, they are not initiators for accidents. The devices operate in response to electrical fault conditions. The changes proposed for ECG 45.3, Revision 1, result in no change to the function or operation of the containment penetration overcurrent protective devices. The protective devices are expected to continue to perform reliably as they have in the past.

The changes to the SR frequencies, extending them from 18 months to 24 months, allow the surveillances to continue to be conducted during refueling outages, which are now occurring at longer cycles. The SR frequency extension has been offset by increasing the sample size tested at each refueling, and by placing a time limit of 48 months or 96 months on testing all overcurrent devices. These changes will result in each device being tested on a more frequent basis than had been required by the original TS.

SRs 45.3.1, 45.3.2, 45.3.3, 45.3.4, and 45.3.5 have been revised to more accurately describe their purpose. This results in updating the testing methods and acceptance criteria based on current industry standards and practices, while maintaining compliance with RG 1.63, Revision 1. The ECG reformatting using the ITS format is descriptive only, and does not affect the ECG technical requirements.

None of the changes proposed for Revision 1 to ECG 45.3 result in an unreviewed safety question.

99-028 Addition of High Emittance Paint on Pressurizer Safety Valve Loop Seals

Reference Document No.: DCP N-049479
Rev. No: 0
Reference Document Title: Addition of High Emittance Paint on Pressurizer Safety Valve Loop Seals

Safety Evaluation Description:
This design change coats the outer surface of the RV-8010B loop seal piping with 3.0 to 4.0 mils (dry film thickness) of high emittance coating. The coating will be applied to areas exposed to the pressurizer vessel on the upstream side of the loop seal and areas exposed to the upstream piping on the downstream side of the loop seal. In addition, the back side of the upstream pipe will also be coated to facilitate the radiant heat transfer from this pipe to the downstream leg.

Safety Evaluation Summary:
This modification in accordance with the subject DCP, as addressed in this LBIE, has no negative impact on plant operation, including impact on operator action, or on the licensing commitments and requirements.

Conclusion:
The modification being implemented by the subject DCP has no impact on the licensing basis and does not result in any unreviewed safety question.

99-029 Degraded Silicone Foam Seals in ECG 18.7 Fire Barriers

Reference Document No.: OE 94-07
Rev. No: 1
Reference Document Title: Degraded Silicone Foam Seals in ECG 18.7 Fire Barriers

Safety Evaluation Description:
NCR N0001789 identified that silicone foam penetration seals without damming do not satisfy the design requirements for vendor qualified seals. As such, these seals should not be relied upon to fully perform their intended design function. These seals are installed in various fire barriers within both Units. The specific seals used in fire barriers separating fire zones 14-A/14-B and 25/S-6 are addressed in Operability Evaluation (OE) 94-07. Unlike others in the plant, these barriers do not have installed fire detection on at least one side of the barrier (which would require a continuous fire watch in accordance with ECG 18.7 for an inoperable fire barrier). However, OE 94-07, evaluates the acceptability of reducing the compensatory measure to an hourly fire watch in lieu of a continuous fire watch based on the expected performance of the existing seals. OE 94-07 is being revised to remove fire zones 13E\14A, 24E\19A, and 19-A\19-C from the scope of this OE .

Fire zones 13E/14A and 24E/19A were included in OE 94-07 Revision 0. The penetration seals in these areas have been repaired and the acceptance criteria of STP M-70A, "U1&2, Inspection of Fire Barrier Penetration Seals," has been met. Although these barriers could be declared operable, the generic TS tracking sheet still includes these barriers in the list of inoperable fire barriers.

Fire zones 19-A/19-C were included in OE 94-07 Revision 0. This room is no longer used to store lube oil and therefore, the barriers no longer need to be maintained as an Appendix A to Branch Technical Position 9.5.1 barrier. Design Change Package M-049464 documents this change.

Safety Evaluation Summary:
NCR N0001789 identified that silicone foam penetration seals without damming do not satisfy the design requirements for vendor qualified seals. As such, these seals should not be relied upon to fully perform their intended design function. The foam penetration seals installed without damming boards are in the walls of the Clean and Dirty Lube Oil Room (Fire Zone 14-B) and the walls between stairwell S-6 and the Operations Ready Room (Fire Zone 25). The penetration seals in these barriers are part of the fire barriers which must be maintained to remain in compliance with Appendix A requirements.

DCPP is committed to comply with the intent of the guidelines of the following sections of Appendix A to BTP (APCSB) 9.5.1, Rev. 0:

Fire zones 13E/14A and 24E/19A were included in OE 94-07 Revision 0. The penetration seals in these areas have been repaired and the acceptance criteria of STP M-70A, "U1&2, Inspection of Fire Barrier Penetration Seals," has been met. Although these barriers could be declared operable, the generic TS tracking sheet still includes these barriers in the list of inoperable fire barriers.

Fire zones 19-A/19-C were included in OE 94-07 Revision 0. This room is no longer used to store lube oil and therefore, the barriers no longer need to be maintained as an Appendix A to Branch Technical Position 9.5.1 barrier. Design Change Package M-049464 documents this change.

Conclusion:
Based on the safety evaluation, safe shutdown capabilities would not be affected by a fire involving these fire barriers. Also, the degraded silicone foam seals can be expected to prevent the propagation of a fire for at least one hour in the areas affected. Given the administrative controls and the fire protection features provided, the existing foam seals will be sufficient to

mitigate the consequences of a fire and ensure safe shutdown capabilities would remain available.

99-030 Primary System Inventory Control and Plant Response Following an RCS Branch Line Break

Reference Document No.: SECL 98-084
Rev. No: 0
Reference Document Title: Primary System Inventory Control and Plant Response Following an RCS Branch Line Break

Safety Evaluation Description:
FSAR sections 5.2.1.6, 5.5.14.2, 6.3.3.2.2 and 15.3.1.1 indicate that the RCS makeup control system is capable of maintaining pressurizer pressure and level after a 3/8" ID line break. Based on an evaluation by Westinghouse, the RCS makeup system is not capable of maintaining pressurizer pressure and level, however, the plant is capable of performing an orderly shutdown and currently meets the applicable licensing and design bases.

These FSAR sections are being changed to reflect the fact that the CVCS is not capable of maintaining pressurizer level and pressure at normal levels following the failure of a 3/8" line if only the normal makeup system is credited (single CCP providing makeup). The CVCS is capable of maintaining primary system pressure and pressurizer level above the reactor trip and low pressurizer pressure safety injection setpoints from which a controlled and orderly shutdown may be performed.

Safety Evaluation Summary:
This issue was originally identified as part of a review of a Millstone Inspection Report (No. 97-206U3) which questioned the basis for a statement in the Millstone FSAR. The Millstone FSAR stated that one charging pump was able to provide makeup flow to maintain normal RCS pressure in the event of an RCS break equivalent to a 3/8" diameter hole in the RCS pressure boundary. A review of actual pump test data indicated that this might not be the case. Because this statement was undocumented, Millstone was issued a level 4 NOV.

As a result of the PG&E review of this issue, Westinghouse was requested to perform a review of the 3/8" line break for DCPP. DCPP-specific details were provided and Westinghouse analyzed the case for DCPP. This review concluded that, under worst-case assumptions (minimum charging pump performance), for a 3/8" I.D. hole, DCPP could not maintain pressurizer pressure and level as stated in the FSAR. Subsequent calculations based on the Westinghouse analysis have concluded that DCPP can maintain pressurizer level and pressure for failures of branch lines with an ID of 1/4".

In addition to the Westinghouse evaluation, an effort was undertaken to determine the actual diameters of the lines penetrating the RCS pressure boundary. The results of this effort concluded that there were three instances where the penetrating line had an inside diameter of 3/8". These cases are:

- Reactor Head Vent
- RVLIS Connection on Hotleg #3 (LT-1310)
- RVLIS Connection on Hotleg #4 (LT-1320)

All of the remaining instrumentation and sample line penetrations through the RCS pressure boundary are 1/4" inch ID tubing or orifices. The results of this effort are summarized in AR A0456727, AE02.

A detailed review of the FSAR and other licensing documentation was also performed to identify the licensing basis and any other commitments and descriptions of the branch lines. The results of this review concluded that the RCS pressure boundary must be considered as Class A per 10 CFR 50.55a and that DCPP is committed to meet ANSI Standard ANSI/ANS 51.1-1988. This standard requires that DCPP be able to shut the plant down in a controlled and orderly manner using the normal makeup system only, and that an orderly shutdown may be considered to include initiation of ECCS systems as part of the shutdown. The shutdown cannot be initiated by the actuation of an ECCS function.

Meeting the requirements imposed by this standard requires demonstration that the reactor operator has sufficient time to conduct the actions necessary for an orderly shutdown and cooldown of the reactor in the event of a break in an 3/8" orificed line attached to the RCS. Since plant operation remains consistent with definitions provided in ANSI/ANS-51.1-1988, there is no credit being taken for operator actions in place of automatic actions, or modification to operator actions, as discussed in NRC Information Notice 97-98, dated October 23, 1997. Therefore, consistent with the requirements of 10 CFR 50.55a and ANSI/ANS-51.1, the ability to shutdown and cooldown the reactor in an orderly manner in the event of a break in a 3/8" orificed line is maintained.

Conclusion:
A review of the population of RCSPB branch lines with diameters $\leq$ 3/8" indicates that almost all of these penetrations are 1/4" ID. The only branch lines that are actually 3/8" ID are two RVLIS penetrations and the reactor pressure vessel (RPV) head vent. For the events involving the 1/4" lines, there is no issue - the plant will stabilize at normal pressurizer pressures and levels as designed and plant recovery and/or shutdown will be able to proceed in a controlled and orderly manner in accordance with plant

procedures. All of the identified 3/8" leaks are characterized as water-side breaks and, as discussed above, result in breaks in the liquid portions of the RCS, leading to a slow loss of pressurizer level at ~1%/minute.

The capability of the primary system inventory control and the plant response following a RCS branch line break in either the steam side or water side has been evaluated. The conclusions of the evaluation have shown that, based on a conservative set of assumptions, for a steam side branch line break and the small diameter water-side lines, pressurizer pressure and level can be maintained. For a water side branch line break with net flow out of the break (3/8" ID lines), the pressurizer heaters will have sufficient capacity to hold the pressurizer pressure at the backup heater setpoint with the backup heaters cycling on and off. The water-side break net flow will cause a continuous slow drop in the pressurizer level of approximately 1%/minute.

In the case of the larger penetrations, there is a potential that the failure of lines downstream of the penetration will result in an event where the normal plant makeup system may not be able to maintain pressurizer level. As discussed above, the occurrence of a failure in these locations will result in conditions where the plant is no longer capable of maintaining pressurizer level utilizing a single CCP. The plant would stabilize at conditions where the makeup system is capable of maintaining conditions (at a reduced pressure and pressurizer level) if no operator action is taken, allowing the operators to perform a controlled and orderly shutdown from the reduced pressure conditions.

The evaluation of the 3/8" water-side break scenario has demonstrated that the reactor operators have sufficient time to conduct the actions necessary for an orderly shutdown and cooldown of the reactor in the event of a break in an 3/8" orificed line attached to the RCS. Since plant operation remains consistent with definitions provided in ANSI/ANS-51.1-1988, there is no credit being taken for operator actions in place of automatic actions, or modification to operator actions, as discussed in NRC Information Notice 97-98, dated October 23, 1997. The small break LOCA analysis results are not affected by the proposed change since the minimum safeguards high head safety injection flows are not impacted by this change. Therefore, consistent with the requirements of 10CFR50.55a and ANSI/ANS-51.1, the ability to shutdown and cooldown the reactor in an orderly manner in the event of a break in a 3/8" orificed line is maintained.

This evaluation changes the conclusions presented in the Diablo Canyon Power Plant Units 1 and 2 FSAR for a 3/8" line break, since normal charging flow may no longer maintain pressure and level for branch line breaks up to a size of 3/8-inch based on the assumptions used by this evaluation.

As discussed, this issue does not represent an unreviewed safety question and the required FSAR changes may be implemented under 10 CFR 50.59.

99-031 Nuisance Leakage from Reactor Coolant System During Shutdown

Reference Document No.: Multiple
Rev. No:
Reference Document Title: Multiple
Safety Evaluation Description:
The proposed change to the reference document is necessary to acknowledge the potential for nuisance leakage from nozzle dams used to isolate the steam generator channel heads from the reactor coolant system (RCS) loop piping for maintenance during plant outages.

The reference documents in their current form do not discuss the potential for nuisance leakage.

Some nozzle dams are experiencing nuisance leakage from their vent valve penetrations which can be accommodated by monitoring of RCS levels, inventory make-up and pumping of displaced inventory from the channel heads to the appropriate plant drain system for reuse, if possible. However, the loss of any inventory outside its intended boundary is serious and requires evaluation to assure there is no adverse impact to plant safety.

Refueling cavity activities affecting nuclear and radiation safety which depend on the RCS and refueling cavity inventory are fuel movement and shielding of plant personnel from the activated reactor upper internals assembly, respectively. In addition, vent paths in cold legs of the reactor coolant system have the potential to quickly uncover the reactor core during postulated loss of decay heat removal events if not properly balanced with equivalent or greater sized hot leg vent paths.

The vent modification was added to the nozzle dams in early 1998 to allow trapped air behind the nozzle dam to be vented to preclude spurious alarms from its leakage monitoring system (ref. A0419502, A0441172). During 2R8, nuisance leakage was observed from the penetration though the nozzle dam by the vent valve (ref. A0455648). Prior to 1R9, a vendor modification to the nozzle dams was performed in an attempt eliminate the leakage path (ref. A0465488). A backing plate was added to strengthen the sealing surfaces of the vent valve to preclude leakage. During 1R9 refilling of the RCS after defueled maintenance, leakage began from the vent valve stem penetration area (ref. A0478199).

The leakage to be managed in the reference document was not necessarily assumed with the introduction of the nozzle dams. Thus, the operation with

the nozzle dams is considered a change in operation or configuration as described in the FSAR update.

Safety Evaluation Summary:
There is no impact on plant operations beyond those currently specified in OP A-5:III. The management of nozzle dam leakage is within all DCPP licensing requirements and does not change the design or performance criteria of the nuclear steam supply systems.

Conclusion:
Nuisance leakage from the reactor coolant system during defueled and Mode 6 conditions, though not desirable, can be managed safely with no increased risk to the health and safety of the general public.

99-032 RHR Recirculation Sump Debris Loading Change

Reference Document No.: Calculation M-591
Rev. No: 15
Reference Document Title: Determination of the Uncovered Screen Area in the Recirculation Sump

Safety Evaluation Description:
The change evaluated is to revise the recirculation sump debris imposition inventory to account for the current configuration of various sources in the containment building. The support calculations and the design bases documents which are impacted are of the level of detail where the specification of debris sources requires the revision of all these documents. The periodic revision of the calculations (outage cycle revisions) necessitates that when potential sources are added or identified, the design bases and licensing bases documents are reviewed for impact. The FSAR is changed due to its level of detail regarding the definition of debris. Thus, the addition of the vapor barrier to the total debris inventory requires a change to the facility as described in the FSAR (Section 6.2.3).

Safety Evaluation Summary:
The change provided by the revision of the design calculation for recirculation sump screen head loss and operability have added the vapor barrier. The installed jacketing containing vapor barrier material within the calcium silicate insulation is located on a variety of components. The addition or change of insulation provided in various design change vehicles is also reflected in the supporting calculations which, like the recirculation sump screen head loss calculation, require, at a minimum, a LBIE screen be performed in accordance with TS3.ID2 every revision. As stated in the change description, this change is of sufficient scope to warrant an FSAR change.

The support calculations (N-42 and N-51) inventory the vapor barrier insulation material (inside and outside the containment crane wall, the pressurizer and RCP cubicles) incorporating changes as stated above. This inventory becomes the input into M-591 for determination of the impact on the sump screen. The addition of the vapor barrier was evaluated for originating as debris from jet impingement on calcium silicate insulation. The mode and rate of transport and the qualitative changes which the barrier material exhibits in a post design bases event environment was then evaluated. The major assumptions in the sump screen calculations are that this material acts like failed coating material in its potential to pile onto the paint pile. The paint chips, which are also specifically evaluated in M-591, form a pile which is then enlarged by the vapor barrier material. The effect is to raise the differential pressure across the recirculation sump screen. The magnitude of the increase is still below the design and safety limits that form the bases for Residual Heat Removal Operability during a design bases event. These limits include the design bases margin for NPSH available to the RHR pumps during various accident cases.

Conclusion:
The licensing basis review of the changes to the design bases calculations for Recirculation Sump demonstrate that the RHR Sump remains operable during the design bases events defined in the FSAR Update. The RHR system component design for reliability, redundancy and operation within design and safety limits is not affected by this change. No events which could impact the health and safety of the public are determined to be created by the change in deposited inventory of debris on the Recirculation Sump Screen. The continuity of measures by which the inventory of RHR sump impacting debris is maintained is considered prudent and in compliance with 10 CFR 50.59 for the evaluation of changes to the facility.

99-033 Throttling CCW Flow from RHR HXS in Modes 4, 5, 6 and Defueled

Reference Document No.: AR A0476737
Rev. No: 0
Reference Document Title: Justification AR A0476737
Safety Evaluation Description:
This CTE will justify and allow throttling CCW flow to the RHR HXS in Modes 4, 5, 6 and defueled without declaring the associated RHR HX inoperable. Throttling will prevent higher-than-design shell side flows and allow finer RCS temperature control during RCS cooldown and heat up.

Safety Evaluation Summary:
The design CCW flow through the RHR Hx shell side is 4950 gpm. The high flow alarm setpoint is 6250 gpm. With the CCWS aligned to supply the RHR Hxs, the shell side flows routinely exceed the high flow alarm which is locked in

for the duration of RHR cooldown. Westinghouse has provided guidance which limits flow to 140% of design or 6930 gpm. Per Westinghouse, flows between 6250 and 6930 gpm are acceptable for periods up to 12 hr. on a refueling frequency.

Additionally, near the end of each cooldown it becomes difficult to control the cooldown rate with a high CCW flow rate to the RHR Hxs.

To clear the high RHR Hx shell side flow alarm and more finely control RCS cooldown, Operations has in the past throttled the CCW flow rate through the in-service RHR Hx. Although CCW flow rate is not a condition of operability for the RHR Hx in the Tech Specs, Operations usually declares any RHR Hx whose discharge throttle valve is not sealed open, inoperable. This is a conservative interpretation, but unnecessarily constraining.

From a design basis standpoint, the requirement for full CCW flow to an RHR Hx exists only in modes in which we postulate a LOCA. The containment integrity analysis (WCAP -13907) assumes a CCW flow of 5000 gpm to the shell side of the RHR Hx to assure adequate heat removal from containment during the recirculation phase of the accident. In all non-LOCA modes, the RHR heat load determines which CCW flow rate is acceptable. Note that per DCM M-46 and M-71, the CCW return lines from the RHR Hx are analyzed to a maximum temperature of 148 deg F. AR A0452350 identified the need to control the CCW return line temperature since normal operation (CCW throttle valves fully opened) could result in exceeding the qualified maximum temperature. It is recognized that throttling the manual CCW return valves will result in higher return temperatures than un-throttled, however, procedural guidance ensures that the maximum qualified temperature will not be exceeded.

Throttling CCW flow to the RHR Hx will also result in elevated return temperatures on the tube side (RHR side) of the Hx. The RHR return lines have been analyzed to the same temperature and pressure conditions as the RHR supply lines. In other words, the return line temperature assumes no cooling through the Hx (reference DCM M-46 and M-71 RHR Mode 5NN). Consequently, throttling the CCW flow will not cause the RHR return lines to exceed their maximum qualified temperature.
From a Licensing perspective, there are four thing we have to acknowledge:

1. There is no explicit requirement for CCW flow rate to determine RHR Hx operability.
2. There is no CCWS requirement for the position for CCW-151 and CCW-459 in Modes 5, 6 and defueled.
3. Surveillance requirement 4.7.3.1.a states that in Modes 1-4 each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed or otherwise secured in position,

shall be in its correct position. For the CCWS, the correct valve positions were determined by PG&E and are controlled by operating procedures (OP K-10 series). Currently, in Mode 4, the manual throttle valves (CCW-151 and CCW-459) are sealed open per OP K-10E4 to comply with SR 4.7.3.1. The correct position for modes in which LOCAs are postulated (Modes 1-3) is full open. In modes for which LOCAs are not credible, the required position is determined by the heat load on RHR during cooldown.

Per A0476737 E01, there is no need to postulate a Mode 4 LOCA, consequently there is no requirement for the CCW return throttle valves to be fully open.

Conclusion:
The referenced AR will establish and justify that it is allowable to throttle the CCWS system flow to the RHR Hx in Modes 4, 5, 6 and defueled without declaring the associated RHR Hx inoperable. The CCWS flow rate to the RHR Hx is not specified in the TS. Only nominal and design flow rates are listed in FSAR Tables 5.5-10 and 9.2-5. These flow rates are not operational restrictions. The CCWS surveillance requirement to seal the throttle valves in their "correct position" is appropriate, however, the operating procedure which specifies the "correct position" is unnecessarily conservative.

99-034 O-23, Offsite Power

Reference Document No.: O-23
Rev. No: March 5, 1999
Reference Document Title: Operating Instructions for Reliable Transmission Service for Diablo Canyon

Safety Evaluation Description:
Operating Instruction O-23 is being revised to update Table II: *Operability and Compensatory Measures for DCPP Unit 1 and 2 for Manual Load Tap Changer (SUT 1-1 / SUT 2-1) Operation, and Capacitor Banks at Diablo and Mesa Either Available or Unavailable.* Table II is being revised to be consistent with the analysis assumptions used in Tables I, III, IV, and V. The changes in Table II are a result of the change in assumptions in the analysis performed in Calculation 359A-DC, Revision 0. The worst-case plant loading is now the Reactor Trip, Delayed Safety Injection Signal scenario, which assumes:

- A Reactor Trip causes all three Condensate / Condensate Booster Pumps to transfer to offsite power
- The Safety Injection Signal causes the LOCA loads to transfer.

Safety Evaluation Summary:
On January 29, 1999 the PSRC requested we prepare this LBIE on the

revised O-23 assumptions, validate the existing Tables in O-23, and evaluate past operability. During the discussion on January 29, the PSRC also asked about available margin for the new Table II in O-23. To perform the O-23 validation effort, we enlisted support from Duke Engineering & Services (DE&S). Many former PG&E employees and contractors who were involved with the analysis for O-23, are now employees of DE&S. We requested DE&S to document:

- The background that led to the discrepancies in Table II of O-23
- A referenced calculation for each line in each Table in O-23
- The available margin for each line in each Table in O-23.

In response to our request, DE&S issued Technical Report 0017-0024-RPT001, *O-23"Operating Instructions for Reliable Transmission Service to DCPP" History and Validation*, February 23, 1999. The DE&S Report confirmed that for the current issue of O-23, May 14, 1998, Tables I, III, IV, and V are correct.

O-23 Table II

The version of Table II referenced in the LBIE for the SLUR Setpoint change that was discussed with the PSRC on January 29, 1999 is correct. This version is based on new analysis performed in January, 1999. The version of Table II issued on December 22, 1997 was correct for all lines in service. Other discrepancies in this version are described below. The version of Table II issued as part of O-23 on May 14, 1998 used outdated assumptions for offsite power loading. These outdated assumptions resulted in non-conservative results for the issued Table. The corrected Table II is provided in the version of O-23 that is evaluated in this LBIE.

O-23 Table II Error Impact

The DE&S Report included an evaluation of Table II all lines in service and no compensatory actions, using the correct assumptions for DCPP loading (Delayed SI), and for 1998 Los Padres and PG&E system peak loading (500 MW and 21,613 MW, respectively). The results show that if the correct assumptions are applied, very narrow ranges of 230kV voltages are required for success.

The May 14, 1998 Table II provided a broader range of success voltages for each tap. However, since the capacitor banks were installed in April, 1998, and the capacitor banks attempt to hold 230kV voltage above 236kV, there is a reasonable possibility that using Table II would result in successful 230kV loading.

The DE&S Report also notes that there are a number of conservatisms in the calculations that could improve the successful outcomes if Table II were applied. First, the calculations are performed for the extreme peak summer PG&E and Los Padres load. Under lighter loaded grid conditions the grid is stiffer, has higher voltages, would support additional load, and would reset the Second Level Undervoltage Relay after transfer. Second, the ETAP software used to make these analyses overstates the load of each accelerating motor by about 15%. This has been reported by OTI the developer of ETAP in an error notice. Third, the analysis assumes the 230 kV voltage is fixed during the analysis. In reality the voltage increases, approximately 1% following a Unit Trip. This increase in voltage might allow for the SLUR relay to be reset. Fourth, the SLUR reset of 93% is a 95/95 confidence number. It is highly probably that the SLUR will reset at a lower voltage than used in the analysis. Fifth, the impedance used in the Thevenin equivalent circuit is based on a conservative voltage of 191kV. An impedance based on a higher voltage would produce a lower voltage drop than the 191kV presently used and increase the probability of resetting the SLUR relay.

Conclusion:

- While the Delayed SI scenario is not a design or licensing basis for DCPP, it is consistent with the accident analysis for DCPP
- The assumptions for plant loading, applying the Delayed SI scenario, are conservative and are based on credible accident sequences and plant responses.
- O-23, May 14, 1998 Tables I, III, IV and V are correct.
- O-23, December 22, 1997 and May 14, 1998, Table II had a sequence of errors. The cause of these errors is understood; corrective actions are in place.
- O-23, December 22, 1997, Table II all lines in service was correct at the time. This was the only application of Table II ever implemented.
- The O-23 Table II that is evaluated in this LBIE is correct. The results are referenced to an approved calculation and re-validated in the DE&S report. The revised Table II applies the correct DCPP loading based on a Delayed SI scenario, applies the correct Los Padres and PG&E System loading, and uses the correct acceptance criteria for the SLUR reset (93%).
- The evaluation of past operability shows that Table II for the case of all lines service was correct at the time it was used; and that Table II was not used for any case shown to be incorrect.

99-035 RHR Recirculation Sump Debris Loading Change Revisions To RHR Design Basis Document

Reference Document No.: Calculation M-591
Rev. No:

Reference Document Title: Determination of the Uncovered Screen Area in the recirculation Sump

Safety Evaluation Description:
The current calculations and design basis documents that determine Recirculation Sump Screen adequacy include consideration of blockage debris generated from failed paint, LOCA jet-impinged fibrous insulation, and failed vapor barrier material that coats the inside of calcium silicate insulation jacketing. This change adds a different type of fibrous debris (cable tray fire stop material) that can be generated by LOCA jet-impingement.

Safety Evaluation Summary:
NRC Bulletin 93-02 instructs licensees to "identify fibrous air filters or other temporary sources of fibrous material, not designed to withstand a LOCA, which are installed in your primary containment." In PG&E's response (DCL 93-153), PG&E identified CFCU filter units, iodine removal units, piping insulation, and steam generator 2-4 insulation. PG&E further demonstrated that only the piping insulation needed to be considered as a credible LOCA jet target. In NRC GL 97-04, the NRC requested a description of how licensees calculate the head loss associated with ECCS suction strainers. PG&E described the methodology in DCL 98-002.

Recently, D.C. Cook identified cable tray fire stop material as a potential source of fibrous materials. Walkdowns in the Unit 1 containment building during the Unit 1 ninth refueling outage identified cable tray fiber stops that contained various types of fibrous materials. The following actions were taken in response:

1. The cable trays and the various types of fire stops were reviewed. Typical material includes Marinite board, Kaowool M board, Kaowool blanket, Mineral wool, and RTV foam. Stops were placed at maximum intervals of 5 feet on vertical trays and 12 feet on horizontal trays. It was determined that fire stops were credible sources of post-LOCA debris.

2. Samples of material were sent to Innovative Technology Solutions Corporation (ITS) for examination and characterization. ITS is active in nuclear industry sump screen debris issues. ITS has provided PG&E with a report describing key parameters for the screen head loss calculation M-591. ITS identified that an important parameter, the fiber specific surface Sv which is fiber surface area/fiber volume, is higher for this material than for fiberglass fibers. Higher Sv causes a higher head loss for a similar thickness of fiber bed.

3. Design basis documents were reviewed for the potential blockage of the sump screen by fibrous debris.

The FSAR and DCMs identified fibers from insulation sources (fiberglass), and noted that some fraction of these would migrate to the screen and contribute to head losses across the screen (DCMs were recently updated to address this issue per AR A0467355). The new fiber source is consistent with existing design basis document statements regarding fibrous debris, except that the source is fire stops rather than fibrous insulation, and the fibers have higher S, values.

4. The fire stop material was inventoried within the jet zone of influence for all LOCA HELB locations in the Unit 1 containment for those breaks requiring the consideration of dynamic effects of postulated ruptures (i.e., main coolant loop piping is excluded by DCPP Leak-Before-Break exemption). The jet impingement criteria (for size of jet and relative energy necessary to disassociate the fire stops into fibers) were consistent with conservative engineering practices and the DCPP licensing basis. The break locations are postulated in accordance with NRC Branch Technical Position MEB 3-1, Rev. 2. The modeling of jet impingement forces and the associated zone of influence is consistent with the methodology of SRP 3.6.2. section 111.3. Any cable tray that was in the line of sight of the break and within the zone of influence was assumed to fail. In addition, any cable tray fire stop within 2 cable tray supports were also assumed to fail due to tray deformation (a civil engineering review in AE#21 to A0470342 justified the two support distance). Per direction given in DCM T-1 2, no credit was taken for blockage of the break or jet interaction from the other side of the guillotine, impingement from jets deflected off solid barriers was not considered, and the jet flow is assumed parallel to the axis of the pipe.

5. The need for the fire stop material was reviewed. It was determined that the fire stops are optional. The cable trays are not safety related. Fire stop removal causes a reduction in the DCPP Defense-in-Depth Fire Protection Program, but on balance, it was determined that the material removal was an advisable overall safety improvement. The material removal review, its impact on the Fire Safety program, and the Licensing Basis Impact Evaluation are contained in Design Change Package (DCP) M-049476.

6. Most fire stop material within the jet zones of influence of affected LOCA HELBs was removed. Some fire stop material proved to be in such a difficult location that its removal was not practical or advisable under ALARA considerations. It was determined that sufficient margin existed in the screen head loss calculation M-591, that the remaining fire stop material would be left in place and considered a new post-LOCA debris generation source. Since the source is identified as

a change to the debris sources accumulating on the sump screen, the screen for this LBIE required a 50.59 evaluation.

7. During this process, it was identified that fiberglass insulation debris generation would be increased by consideration of pipe whip and subsequent moving jet impingement zones. This is not a consequence of the fire stop issue, but it does impact the debris generation calculation N-42 and the screen head loss calculation M-591.

8. A series of calculations were performed to determine that the additional fibrous debris (fire stop and extra fiberglass insulation) does not impact the ability of the screen to perform its function. This calculation process is identical to the process described in DCL 98-002. The Unit 1 debris generation calculation N-42 has been revised to include the fire stop material and additional fibrous insulation material resulting from pipe whip considerations. (Unit 2's calculation N-51 will be updated after the next Unit 2 refueling - OE 99-01 is in place to address Unit 2's operability until that time.) This inventory becomes the input into M-591 for determination of the impact on the sump screen. The effect is to raise the differential pressure across the recirculation sump screen. The magnitude of the increase does not impact the conclusion of calculation M-591, that sufficient head exists for Residual Heat Removal Operability during a design bases event. These calculations assure that the design basis requirements are met for NPSH available to the RHR pumps during various accident cases.

Conclusion:
The licensing basis review of the changes to the design bases calculations for Recirculation Sump demonstrate that the RHR Sump remains operable during the design bases events defined in the FSAR Update. The RHR system component design for reliability, redundancy and operation within design and safety limits is not affected by this change. No events which could impact the health and safety of the public are determined to be created by the change in deposited inventory of debris on the Recirculation Sump Screen. The continuity of measures by which the inventory of RHR sump impacting debris is maintained is considered prudent and in compliance with 10 CFR 50.59 for the evaluation of changes to the facility.

99-036 Steam Generator Tubes Locked in Tube Support Plates

Reference Document No.: FSAR
Rev. No: 13
Reference Document Title: FSAR Sections 5.5.2.3.4 and 3.7.1.3
Safety Evaluation Description:
Normal corrosion processes associated with operation of the DCPP steam

generators (SG) have caused the as-designed gaps between the SG tubes and the tube support plates (TSP) to become filled with corrosion products. This creates a fixed boundary condition at the tube-TSP interface that was considered neither in the original analysis, nor in the original design. Recent analysis of the SGs in this locked-tube condition demonstrate compliance with ASME Code requirements, hence acceptability for continued operation. The FSAR is to be updated to reference this analysis.

The operability analysis uses the damping values recommended in Reg Guide 1.61 as approved for use at DCPP. As NRC approval of these recommended damping values is implied, but not clearly reflected in FSAR section 3.7.1.3 "Critical Damping Values" , the FSAR entry must be clarified.

Safety Evaluation Summary:
This LBIE addresses FSAR changes only: 1) Including reference to the operability analysis for the locked-tube condition, and 2) Clarification of the acceptable use of Reg Guide 1.61 recommended damping values. This LBIE is somewhat out of the ordinary in that the physical change that makes necessary the first FSAR change above has taken place as a result of normal equipment operation—it is not a change to be implemented by the modification process. There is no physical modification to be implemented.

While investigating NRC Information Notice 96-09 "Damage in Foreign Steam Generator Internals," PG&E recognized that the design analysis of record for the DCPP SGs does not properly reflect the fixed boundary conditions caused by the closing of the tube-TSP crevices. PG&E initiated action to evaluate the effects of tube-locking, and engaged Westinghouse Electric Company to perform analysis to demonstrate Code compliance of the operating steam generators. The analysis is to be referenced in section 5.5.2.3.4 "Tube and Tubesheet Stress Analysis" as a supplement to the (original) analysis of record.

FSAR section 3.7.1.3 "Critical Damping Values" contains in note "c" an ambiguous reference to acceptable use of the Reg Guide 1.61 damping values rather than the tabular damping values (i.e., 0.5% DE, 1.0% DDE, 4.0% Hosgri) for the reactor coolant loop. SSER 7 states in part, "The damping values recommended by Reg Guide 1.61 constitute our current criteria and have been acceptable on all applications for several years. Therefore, they are acceptable for use in the seismic reevaluation." referring to the Hosgri analysis. SSER 7 continues with a statement concerning NRC acceptance of the full-size plant tests and a request that PG&E demonstrate the similarity of DCPP SGs to those of Indian Point 2. This demonstration was accepted in SSER 8. Thus, the note in FSAR section 3.7.1.3 will be expanded to clearly indicate the acceptability of Reg Guide 1.61 damping

values (i.e., 2% DE, 4% DDE, 4% Hosgri).

Conclusion:
Both proposed changes to the FSAR are acceptable.

99-037 OE 98-09 to Remain Open Past 1R9

Reference Document No.: OE 98-09
Rev. No: 3
Reference Document Title: Oper. Of Seis. Qual. Equip. in Turb. Bldg. with Undersized Ped./Bldg. Gaps

Safety Evaluation Description:
OE 98-09, which is applicable to both units, addresses the operability of the turbine building, the Unit 1 and 2 turbine pedestals, and seismically qualified equipment located in the turbine building which may be affected by potential interaction between the Unit 1 and 2 turbine pedestals and the turbine building structure during a design basis Hosgri earthquake. The OE addresses conditions that do not meet the minimum separations credited in FSARU table 3.8-27A, "Turbine Building and Turbine Pedestal (Units 1and 2), Comparison of Displacements and separations at Elevation 140 Feet Hosgri Cost of Work Analysis."

No changes are proposed by this LBIE. The LBIE only evaluates extending OE 98-09, R3 past 1R9.

Safety Evaluation Summary:
During the 1R9 refueling outage, all Unit 1 modifications requiring shutdown conditions were completed. The remaining Unit 1 actions can be completed on line. This condition is acceptable because the remaining noncompliances are minor in nature, and even in the event of a Hosgri event will lead to only localized damage which will not affect the safety function of Class I equipment in the turbine building. The safety evaluation performed for OE 98-09, Section I.C, "Effect of Condition on Safety Function," states that, in general, the criteria that a gap be established between the turbine building and pedestal such that no interaction occurs during a limiting earthquake significantly simplifies the analysis of building response. The OE evaluates a limited number of areas where this criteria is not maintained. However, the resulting impacts are significantly limited in size compared to the overall size of the two structures. The Unit 2 evaluation envelopes Unit 1. The major discrepancies for Unit 1 were corrected during 1R9, so the Unit 2 evaluation continues to envelope Unit 1. As explained in the Unit 2 evaluation, the nonconforming conditions could result in impacts, but they are limited to small localized areas, and only localized damage is expected to occur. Also, maximum displacement does not occur initially. The motion of the pedestal and building increases over the duration of the earthquake. Therefore, it is

likely that any resulting damage will occur progressively, widening the gap (e.g. crushing the concrete) with each cyclic impact and each successive impact will be less than the calculated maximum. The load acts in a relatively small area and the localized damage will not affect the capability of safety related equipment to perform its function.

Unit 2 modifications will be completed during 2R9. The OE completion date for both Unit 1 and Unit 2 is November 30, 1999 (following completion of 2R9). This schedule was approved by the PSRC on January 29, 1999 (Meeting No. 99-006) in conjunction with the PSRC review of NCR N0002058 (Reference 6).

OM7.ID8, Revision 6, "Operability Evaluation," step 5.3.4.a, "Plant Start-Up from a Refueling Outage with an OE Remaining Active," requires preparation of an LBIE screen and, as appropriate, an LBIE, for all OEs applicable to a Unit, prior to the restart of the Unit from a refueling outage.

Since OE 98-09, Rev. 3 will extend past the 1R9 refueling outage, an LBIE screen is required. LBIE screen question 4.1.a) screened "yes" requiring completion of an LBIE.

Conclusion:
Extension of OE 98-09, Rev. 3 past the 1R9 refueling outage does not involve an unreviewed safety question.

99-038 Equipment Control Guideline Change

Reference Document No.: ECG 4.3
Rev. No: 2
Reference Document Title: Equipment Control Guideline for Steam Generator

Safety Evaluation Description:
The change evaluated is to revise the Equipment Control Guideline 4.3 to reflect the original intent of the low temperature over pressure control administratively applied to the steam generators. The change will allow the secondary side of the steam generators to continue to be limited to a minimum temperature prior to pressurization. The only change is that the primary pressure, or RCS pressure, and temperature will no longer be limited by this Equipment Control Guideline. No change to the FSAR is required since the minimum temperatures for pressurization of either side of the steam generator are not specified, though the ECG 4.3 is listed in Chapter 16.

Safety Evaluation Summary:
The change provided by the revision of the Equipment Control Guideline for Steam Generator pressure limitation clarifies the original intent of the ECG.

Originally, the ECG was a Technical Specification 3/4.7.2 which defined a minimum steam generator secondary side temperature for the purpose of testing. There was a potential for having limits for a full range of pressure tests for the secondary side of the steam generators. The limits would require access to the primary side of the tubesheet, thereby, allowing the primary side to be vented to atmosphere. To preclude exceeding fracture toughness stress limits, minimum temperatures were established. The pressure temperature limits governing differential pressure between primary and secondary side of the steam generator were also established unaffected. This change will in no way affect such testing limits. Thus, there is no potential for impacting the fracture toughness limits by the secondary system.

This change to the ECG 4.3 requires an Licensing Basis Impact Evaluation per procedure OP1.DC16.

The control of the reactor coolant temperature in this ECG is not necessary to protect the RCS pressure boundary since the RCS pressure and temperature limitations are designed to protect the pressure boundary including the primary side of the tubes and the tubesheet. The WCAP pressure and temperature limitations for the secondary hydro of the SGs protects the integrity of the secondary sides of the tubes and the tubesheet. Since both sides of the pressure boundary are maintained, there is no increase in the probability of an accident (SGTR), a malfunction (failure of a tube or the tubesheet), the possibility of a new malfunction or accident (failure of the tubesheet or multiple SG tubes), or increase in consequences (since the assumptions regarding tube failures are maintained, and the hydro limitations demonstrate that the secondary side of the SG pressure boundary won't be exceeded).

The SG manufacturer (Westinghouse) was contacted to obtain information regarding the basis for controlling both the primary and secondary side pressure and temperature, and that they could find no basis for the control of the primary side temperature. It was indicated that they believed the ECG could be revised to remove the reactor coolant reference without affecting the integrity of the SGs.

The supporting Westinghouse analysis (WCAP 13141) provides the minimum secondary system pressure and temperature requirements. The limitation of the Equipment Control Guideline 4.3 is revised to those parameters.

Conclusion:
The control of primary system pressure and temperature is provided by LTOP and the supporting analyses and licensing documents for LTOP. The licensing basis review of the changes to the Equipment Control Guideline 4.3

indicate a different bases for the control of pressure for the secondary side of the steam generator. This change will reduce confusion between the two pressure and temperature control sets of limits and ease the administrative burden on the operating personnel to achieve them. The result is that the adherence of the reactor coolant and secondary side of the steam generators to the appropriate fracture toughness limits will continue with out the multiple variables imposed by the current ECG 4.3.

99-043 Allow Alternate LTOP XMTRs PT-403A/405A for Use in Mode 4

Reference Document No.: STP I-7-P403A/405A
Rev. No: 4
Reference Document Title: RCS Loop 4/3 Alternate Wide Range Press XMTR PT-403A/405A Calibration

Safety Evaluation Description:
Allow use of Alternate LTOP Transmitters PT-403A, -405A to be used in Mode 4.

Safety Evaluation Summary:
The alternate LTOP transmitters meet the requirements for Safe Shutdown. Table 9.5G-1 and Appendix 9.5A have been reviewed. One of three wide range PTs is required: PT-403, -405 and -406. Electrical Engineering has reviewed the cable routing of PT-403A, PT-405A, and PT-406 and has confirmed that a fire in any 1 area will not disable all three signals. Appendix 'R' criteria for safe shutdown has been met with alternate LTOP transmitters in service in Mode 4.

The SAR in Section 5.2.2 describes the Low Temperature Over Pressure (LTOP) protection feature of the Plant. This feature is not changed. The alternate transmitters, PT-403A/405A, used to provide the necessary inputs to the LTOP circuitry are left in service during Mode 4. The LTOP feature described in the SAR does not specify the input pressure transmitters by number. It is acceptable to leave the alternate LTOP transmitters in service in Mode 4 because they meet all engineering functional requirements for this service. The transmitters are redundant and do not share a common tap. The taps are from the same RHR suction line from Loop 3. Sharing different taps from the same line is acceptable for this service. The alternate transmitters, PT-403A/405A, are fed from the same redundant power supplies as the normal transmitters, PT-403/405. The difference in location between the alternate and normal transmitters is that PT-403A/405A are located in Containment while PT-403/405 are located in the Auxiliary Building.

Although PT-403A/405A are located below the flood elevation, there are no Mode 4 accidents that could result in a harsh environment (including flooding) that would result in the loss of these transmitters. However, if you assume a

high energy line break at the connection to the RCS, these transmitters are not required as the breach would provide the necessary pressure relieving pathway (per T.S. 3/4.4.9.3 if PORVs not available RCS vent greater than or equal to 2.07 in^2 is required). PT-403A/405A are qualified at the installed location in Mode 4 and do not represent a common mode failure mechanism.

Conclusion:
The use of alternate LTOP transmitters PT-403A, -405A in Mode 4 meets all engineering functional requirements. This activity does not result in an unreviewed safety question.

99-047 Relocation of Operational Support Center (OSC)

Reference Document No.: Vol. 11 of Plant Manual
Rev. No: Rev 3/Ch 18
Reference Document Title: Emergency Plan
Safety Evaluation Description:
This LBIE (Section 5 only) addresses the Operational Support Center (OSC) primary facility relocation from the 140' NSSS office to the 104' FIN Team area. In addition, the first back-up facility is changed from the 119' NSSS office to the 140' NSSS office.

Safety Evaluation Summary:
The relocation of the OSC primary facility from the 140' NSSS office to the 104' FIN Team area requires a change to the Emergency Plan (sections 7.1.5.1 and 7.2.1.6). The specific location is described in detail in the E-Plan and thus a revision is necessary.

Conclusion:
The change is not technical and does not result in a change in intent. There is no decrease in effectiveness of the Emergency Plan based on the relocation of the OSC.

99-048 O-23 Interim for Morro Bay Bus E Replacement Project

Reference Document No.: O-23
Rev. No: March 19, 1999 Interim
Reference Document Title: Operating Instructions for Reliable Transmission Service for Diablo Canyon
Safety Evaluation Description:
The changes evaluated in this LBIE are the licensing impacts of the temporary transmission line re-configurations and the interim replacement of the O-23 operability Tables I, II, III, IV and V, during the Morro Bay Bus E Replacement Project. The 230kV offsite power system supplying the Diablo Canyon Power Plant includes the following transmission lines:

- Diablo - Mesa
- Morro Bay - Diablo
- Morro Bay - Mesa
- Morro Bay - Gates #1 & #2
- Morro Bay - Midway #1 & #2

Six of these transmission lines connect into the Morro Bay 230kV Switchyard. PG&E Transmission Projects is planning a replacement of 230kV Bus E in the Morro Bay Switching Station. To take a clearance on Bus E, the connected transmission lines and generating units must be disconnected and rerouted around the Bus. To clear Bus E, four temporary 230kV connections (Shooflies) will be made at Morro Bay. The major changes in the 230kV system include:

- The Gates #1 line will terminate at Templeton and will not connect to Morro Bay
- The Midway #1 line will connect directly to the Mesa line. Neither line will connect to Morro Bay.

The Bus E Replacement Project is scheduled form March 22, 1999, to June 18, 1999. During this period of time the operability tables in O-23 will be replaced with one interim table.

Safety Evaluation Summary:
To analyze the effects of the MB Bus E Replacement Project, on the DCPP 230 /12kV Startup system, we contracted with Duke Engineering & Services. Many former PG&E employees and contractors are now employees of DE&S. These personnel were responsible for the previous analysis of offsite power and are familiar with the PG&E 230kV system and the DCPP requirements. The scope of work included the analysis of the intermediate states of the 230kV system (while each of the shooflies are installed), and the final configuration for the duration of the MB Bus E outage. DE&S evaluated single and credible double contingency outages of the 230kV circuits serving DCPP, during the MB Bus E outage. DE&S used the Electrical Transient Analyzer Program (ETAP) software. This software has been applied in the analysis of the DCPP offsite system, for the last three years. The software is nuclear grade and has been verified and validated for use at DCPP.

The results are documented in DE&S Technical Report 0017-0033-RPT001, *Morro Bay Switchyard Outage Interim Version of O-23*, dated March 19, 1999. The important assumptions in the analysis are:

- The Second Level Undervoltage Relay reset value is 93.75% per DCP E-049451
- The DCPP loading is based on the worst case Unit trip or LOCA loading

- The maximum load in the Los Padres area is 425 MW
- The capacitor banks at DCPP and Mesa are operable
- The DCPP 230 /12kV Startup Transformer automatic Load Tap Changer is operable
- The double contingency outage of both Midway lines assumes a MB generator is in service for voltage support

The results of the DE&S Report are captured in the Interim O-23 dated March 19, 1999, that is the subject of this LBIE. The results show that the DCPP design and licensing basis requirements are maintained during the MB Bus E Replacement Project. The capacity and capability of the 230kV system is confirmed to be acceptable for DCPP loading. The DCPP loading is based on the worst case Unit Trip or LOCA loading. As documented in LBIE 99-034, dated March 4, 1999, the worst case DCPP loading is based on the reactor trip, delayed safety injection signal scenario. This scenario transfers a Unit's secondary loads, including a standby condensate/condensate booster pump, and the LOCA loads to the 230kV Startup system. Even for this worst case scenario, the 230kV offsite power system is operable, with compensatory measures, for any single and credible double contingency outages of 230kV transmission lines serving DCPP.

There are three double contingency line outages evaluated in the DE&S Report: 1) loss of both Gates lines, 2) loss of both Midway lines, and 3) loss of both Mesa lines. Since the Gates #1 line is not connected to Morro Bay switchyard during the MB Bus E Project, the loss of both Gates lines is a single contingency outage of the Gates #2 line. The loss of both Midway lines requires additional voltage support from an operating Morro Bay Power Plant Unit. This is the only case studied that would require a MB Unit to operate. A Morro Bay Unit could be brought on-line in approximately 36 hours to reinstate operability of DCPP offsite power. This 36 hours is within the 72-hour Technical Specification allowed outage time for DCPP offsite power. The loss of both Mesa lines causes the loss of customers in the Nipomo/Santa Maria area. Although this is an undesirable result for PG&E customers, the effect does not impact DCPP. DCPP is operable; no compensatory measures are required.

The maximum loading in the Los Padres area is assumed to be 425MW. This loading is less than the peak loading assumed in the summer 1998 of 500MW. The 425 MW peak is based on the MB Bus E outage being completed prior to the hot weather and peak air conditioning loads. The 425 MW peak has been reviewed by the CalSO staff, and found to be acceptable.

Each of the transition configurations (shooflies) has been analyzed. During installation of shoofly #1 and shoofly #4, our ETAP transient analysis shows

that with an operable automatic LTC, the 230kV system remains operable. These transition configurations present the worst case for DCPP. During each of these transitions, DCPP is served with only one 230kV circuit.

The analysis for normal 230kV system configuration is documented in design calculations 357-DC Rev 1, 359-DC Rev 1, 359-DC Rev 1-2I, and 359A-DC Rev. 0. The transient analyses using the worst case line outage conditions during this temporary installation, is documented in the DE&S Report. This Report shows that the SLURs will reset with no more than two steps of the startup transformer LTC. The LTC can move three steps in the time available to reset the SLURs. Thus margin remains the same as the original analysis.

There are a number of conservatisms in the calculations that could improve a successful transfer to the 230kV System. The ETAP software used to make these analyses overstates the load of each accelerating motor by about 15%. This has been reported in an error notice by the developer of ETAP. The analysis assumes the 230 kV voltage is fixed during the analysis. In reality the voltage increases, approximately 1% following a Unit Trip. This increase in voltage would improve the SLUR relay reset. The SLUR reset of 93.75% is a 95/95 confidence number. It is highly probable that the SLUR will reset at a lower voltage than used in the analysis. The assumed peak load in the Los Padres area is limited to a few hours from Monday through Friday. For the remainder of the time, the area load is reduced and the 230kV has reserve capability for DCPP load transfer.

DCPP Precautions During the MB Bus E Replacement Project

Operation crews will receive special training (tailboards) on the offsite power situation. During the one week of transition into and out of the temporary 230kV configurations (construction of the shooflies), DCPP Operations will take prudent precautions to minimize high-risk Plant activities. First, the work may not start if one of the DCPP units is on Startup power. Second, during the installation of the shooflies (jumpers) #1 and #4, on March 23rd and March 28th respectively, there will be a single power line providing 230kV power to DCPP. These evolutions will start on days and have an estimated duration between 12 and 16 hours. During those times, DCPP will minimize (through shift orders) any high risk maintenance (from both a PRA perspective, i.e. D/Gs and ASW maintenance, and a trip perspective). Third, when the schedule is finalized for the removal of the shooflies and only one power line will feed DCPP with 230kV power, the shift orders will be re- issued (date to be determined, expect in early June).

For the approximate 2-month period while the MB Bus E reconstruction takes place, there is a high assurance that the power will not be interrupted to

DCPP. This is because of design and maintenance practices. The design of shoofly #1 is such that it completely bypasses MB. The jumper is installed from the Midway #1 line to the Mesa to DCPP line about 50 yards away from MB. Hence, construction activities in the Bus E area have a high probability of not impacting that power source to DCPP. Additionally, maintenance is restricted on the southern system (that could affect Midway reliability) during this period. Also, maintenance in the Bus E area will be conducted under the watchful eye of a DC Switching Center operator, who will have instant communication capability with the DC Switching Center. Maintenance and construction will be performed by contractors and PG&E personnel.

There are several precautions established to prevent interference with shoofly #4, located about 30 feet away from the Bus E construction activity , such as having a qualified electrical observer for crane work. Shoofly #4 provides a more direct electrical path of 230kV power from the MB switchyard to DCPP. The design and maintenance practices planned should assure a high reliability of 230kV power to DCPP during the duration of the MB Bus E Replacement Project. During the Project, normal DCPP plant testing will continue, including EDG surveillance testing. During this testing , a single EDG may be out of service for 8 hours or less. The other five EDGs remain capable of accepting LOCA loads, should it be necessary.

Conclusion:
In summary:

- The 230kV system configuration during the MB Bus E Replacement Project is different than the configuration described in the FSAR. However, the temporary connections provide the same functionality described in the FSAR.
- The temporary transmission connections ensure the design and licensing basis configurations are functionally maintained. There are two sources of 230kV power to the DCPP switchyard, each capable of supplying the DCPP emergency load. The transition configurations and the 2-month temporary configurations are analyzed in a calculation. The results of the calculation show the 230kV system is operable for all lines in service and for single line outages and for credible double contingency outages.
- The O-23 that is evaluated in this LBIE is based on an approved calculation. The interim Table in O-23 applies a conservative loading of 425MW for the Los Padres area, and uses the correct DCPP loading assumptions and acceptance criteria for the SLUR reset (93.75%).
- The conservatisms in the analysis increase the probability that the SLURs will reset, even under the worst case loading.
- The analysis confirms the 230kV system is operable during the Morro Bay Bus E Replacement Project.

The Interim O-23 dated March 19, 1999, will ensure the 230kV system operability is controlled and maintained. The activity does not result in an unreviewed safety question.

99-049 Changes made to FSAR Update, Chapter 13, and Chapter 17 in preparation for the issuance of FSAR, revision 12.

Reference Document No.: FSAR Update
Rev. No: 12
Reference Document Title: Final Safety Analysis Report Update
Safety Evaluation Description:
The following title changes have been incorporated into the text of Chapter 13 and 17 of the FSARU. Many of these changes were made as a part of a general reorganization within NPG and in particular at DCPP:

1) Chairman of the Board and Chief Executive Officer was changed to Chairman of the Board.
2) President and Chief Operating Officer was changed to the President and Chief Executive Officer.
3) Senior Vice President, NPG was changed to Senior Vice President and General Manager, NPG.
4) The title "Executive Vice President, Electric Generation," and descriptive text, have been removed.
5) The titles "Manager, Outage Services," Director, Contract Services," and "Supervisor, NPG Support Services," have been removed and the title "Manager, Site Services" has been added.
6) The title "Manager, Design Engineering Services," and descriptive text have been removed from the text.
7) The title Director, Vendor Quality Control was added to the text to incorporate an existing function not previously noted in the text or on Figure 17.1-1.
8) Additional editorial changes are also noted:
   a) The word corporation was replaced by the word utility.
   b) Acronyms were left to replace spelled out titles for, as an example, DCPP.
   c) Redundant words were removed from Section 17.18

These changes are noted on pages 13.1-1, 13.1-5, 13.1-6, 13.4-1, 13.4-2, 17. i, 17.v, 17.1-1, 17.1-2, 17.1-3 and 17.1-3a, 17.1-4, 17.1-5, 17.1-6, 17.1-7, 17.2-2, 17.18-2 and Figures 17.1-1 and 17.1-2. The actual changes are noted on the attached 19 pages.

Safety Evaluation Summary:
During July and August, 1998, in preparation for the issue of revision 12 of the FSAR Update, a number of changes where incorporated into Chapter 13, "Conduct of Operations," Section 13.1, "Organizational Structure," Section

13.4, "Review and Audit" and Chapter 17, Quality Assurance, Section 17.1 "Organization" and Section 17.2 "Quality Assurance Program," and Figures 17.1-1, "Pacific Gas and Electric Company Utility Organization" and 17.1-2, "Nuclear Quality Services in the Utility Organization," in the Living FSARU by NSAL. These changes were necessary to reflect the organization and position title changes already in place, or soon to be in place, within NPG. NQS was contacted by NSAL in July and requested to review the noted changes. During the review process a number of modifications to those changes, as well as some additional changes to the text and Figures were discussed and agreed upon. As a follow-up action, and to document the NQS approvals required by XI3.ID2, NQS submitted Update Change Requests on July 22, 1998, and August 20, 1998, to document that approval. Those changes are being reviewed on this LBIE to verify that there are no other issues which must be addressed. The actual changes are noted on the attached 18 pages. The need to perform this LBIE was document during an NQS Assessment performed in December 1998 and was reported as a quality problem on AR-A0473258.

Conclusion:
This reviewer is in agreement with the preliminary USQ assessment documented on AR-A0473258, and has also concluded that the organization changes did not change the functions performed within NPG. The reporting relationships of certain organizations, Managers, and Directors were changed but those changes did not negatively impact the quality assurance program. There was no transfer of quality assurance responsibility from NQS to other organizations. Organizational detail was added regarding the Director, Vendor Quality Control which was not a change, but which had not previously been described in Chapter 17. In conclusion, the changes are not judged to represent either a USQ or a reduction of a QA program commitment.

99-050 FHARE 122, 33, 44

Reference Document No.: FHARE 122, 33, 44
Rev. No: 0, 3, 3
Reference Document Title: FHARE 122, Staircase S-7 Fire Area Boundary
FHARE 33, Undampered Ventilation

Safety Evaluation Description:
This LBIE is being generated to document the acceptability of having an unfinished plaster wall at the west side of staircase S-7 (elevation 1 19-ft of the Turbine Building). FHARE 122 also evaluates the acceptability of a 2-hour barrier communicating to adjacent fire areas 20, 22-C, TB-12/23-C and 24-D. FHARE 122 is attached to this LBIE and provides the technical basis of the acceptability of the lesser rated configuration.

Inclusion of Fire Zone S-7 into fire area TB-13 required a revision to FHARE

33 to include the new fire area designation for S-7. FHARE 44 was also deleted because the evaluation of the barrier between Fire Zones 24-E and 25 is no longer required. This LBIE will also address those FHARE revisions.

Safety Evaluation Summary:
FHARE 122 evaluated the affects of an unfinished plaster wall of Staircase S-7 (west wall at elev. 119-ft of the Turbine Building) in the event of a fire. FHARE 122 also evaluates the acceptability of a 2-hour barrier communicating to adjacent fire areas 20, TB-12/23-C and 24-D. With the low combustible loading, and the existing fire protection features in the affected fire zones, the unfinished wall and the 2-hour fire rated stairwell boundary would be able to withstand the hazards in the area. The ability to achieve and maintain safe shutdown is not adversely affected in the event S-7 was combined into fire area TB-13 (fire zones 24-E, 23-C-1 and 25) and having a 2-hour rated fire area boundary for the stairwell and adjacent fire areas. This LBIE supports FHARE 122, revisions to FHARE 33 and 44, and the associated FSAR change.

Conclusion:
The unfinished wall would be able to withstand the hazards associated within the area. The 2-hour rating for the stairwell wall from elevation 85-ft through 140-ft would be able to withstand the hazards associated with the affected fire areas. The ability to achieve and maintain safe shutdown would not be affected by the existence of this unfinished plaster barrier and the 2-hour fire area boundary rating.

99-051 Isolate the SFPHX 2-1 Shell Side Drain from the Open Drain System

Reference Document No.: DCP P-050459
Rev. No: 0
Reference Document Title: Isolate the SFPHX 2-1 Shell Side Drain from the Open Drain System

Safety Evaluation Description:
The Unit 2 SFPHX shell side (CCW) drain is currently hard piped to the Auxiliary Building Sump (ABS) via the open drain system which is part of the Liquid Radwaste System (LRS). Any CCW leakage past the shell side drain valve represents an unknown input of molybdated CCW into the LRS. The presence of molybdates in the LRS accelerates the depletion of the cation resins in the LRS demineralizers (A0459618). This change adds flanges and a blind plate to isolate this drain line from the open drain system while maintaining the seismic qualification of the safety-related portion of the line. This change also adds a tee and pipe cap to allow collection of the shell side liquid for re-use or disposal if needed.

Safety Evaluation Summary:
This drain line would only be used with the SFPHX out of service. This change affects the text of the FSAR Update, but the function of the SFPHX, CCW and the LRS are unaffected.

Conclusion:
This drain line is normally isolated. This change provides a more positive isolation from the open drain system. The function or failure of this drain line is not described in any accident analysis or the Technical Specifications. The function or malfunction of the CCW system, SFPHX and the LRS is not affected by this change. This change does not represent an Unreviewed Safety Question and does not require prior approval of the NRC.

99-052 Replace Feedwater Pump Speed Control System, Unit 2

Reference Document No.: DCP J-050419
Rev. No: 0
Reference Document Title: Replace Feedwater Pump Speed Control System, Unit 2

Safety Evaluation Description:
This change replaces the existing hybrid analog/digital main feedwater pump control system with a high-integrity, triple-redundant system that is almost exclusively digital. In the existing system, control algorithms are executed in analog modules, while monitoring for off-normal conditions is performed in digital subsystems. The upgrade modifies the servos to add new servo pilot positioners and feedback linkages, and installs a separate control oil supply skid for each pump to supply clean oil to the new servo pilot positioner. The upgrade includes linear voltage differential transformers (LVDT) to indicate governor valve position, a third speed probe, and an interface to the plant process computer. New alarms are added to the plant main annunciator system to indicate various abnormal system conditions.

Safety Evaluation Summary:
Critical electronic components in the Main Feedwater Pump Control System (MFPCS) are obsolete and no longer supported by the manufacturer. If the system is not replaced, eventually a failure will occur within the feedwater pump controls that cannot be repaired. Such a failure would most likely cause lost generation or stationing a reactor operator at the pump full time for local manual control. The new system is supplied by a large U.S. national control system supplier who is expected to support the system for the reasonable future.

The control hydraulics have been a continuing source of problems and unit trips due to entrained moisture and particulates. Failures and upsets within the feedwater pump control system have resulted in significant unit transients or

trips. In the existing MFPCS, control oil is derived from the MFP lubricating oil system. Ongoing hydraulic system problems result from water and particulates in the control oil system. This change installs a separate, redundant control oil supply skid for each pump to supply clean oil to the new servo pilot positioner. The dedicated control oil system eliminates components that are susceptible to control oil contamination. However, MFP lubricating oil will still provide the motive force for the main servo pistons.

The MFPCS upgrade is fault-tolerant, mitigating the condition of the existing system, where failure of a single device can trip the unit. The new system is a proven commercial design that employs triple-redundant electronic processors. If a single processor fails, the redundant processors will allow the system to continue automatic operation. The process 1/0 is also trip le-redundant. Failure of a single 1/0 module, channel, or processor will not cause loss of control.

The new system does not entirely eliminate single points of failure in the feedwater pump control system. The new servo pilot positioners and the existing servos and governor valves are not redundant. Based on field experience, the servo pilot positioners will not contribute to a failure rate greater than the existing cup valves that are replaced by the upgrade.

The upgrade includes linear voltage differential transfortners (LVDT) to indicate governor valve position, a third speed probe, and an interface to the plant process computer. New alarms will be added to the plant main annunciator system to indicate various abnormal system conditions. The added instrumentation will enhance reliability and improve the operator's ability to perform feedwater system diagnostics.

Conclusion:
The DCPP MFPCS is a nonsafety-related control system that is not required to mitigate any accidents or events evaluated in the FSARU. However, malfunctions in the main feedwater pump control system can contribute to the probability of events involving supply (or lack) of main feedwater. Although the MFPCS is not safety-related, components of the Main Feedwater System (MFWS) that comprise the Feedwater Isolation (FWI) function are required to mitigate certain FSARU accidents or events. The MFPCS upgrade makes a minor modification to the trip oil system; i.e., the check valve between the trip oil and control oil systems is removed. Several new trip functions are added, but the existing mechanical trips are not modified or affected. The changes will not adversely affect any Engineered Safeguards (ESF) systems or components associated with detection or mitigation of events associated with the FWI function. The upgrade specifically addresses aspects of the existing system that contribute to lack of reliability and lost generation. The upgrade

will not increase probability, frequency, or consequences of evaluated events or equipment malfunctions.

99-053 DCM T-11 Revision, AT DCMC AR A0437621

Reference Document No.: DCM T- 11
Rev. No: 8
Reference Document Title: Control of Heavy Loads
Safety Evaluation Description:
DCM T-11 refers to ANSI B30.2-1976 as the standard for inspection and testing of Category I monorails and hoists. ANSI B30.2 is the standard for cranes. The correct standards for monorails and hoists are ANSI B30.11 and B30.16. DCM T-11 is revised to reflect the correct ANSI Standards for inspection and testing of monorails and hoists.

Safety Evaluation Summary:
DCM T-11 Sections 4.4.1.6.a.2 and 5.3.A are revised to show that the Category I monorails and hoists are inspected and tested per requirements of ANSI B30.11-1973 and ANSI B30.16-1973 (instead of ANSI B30.2-1976).

Conclusion:
This update to DCM T-11 and change to PCD Commitment T03538 does not result in an unreviewed safety question. This change does not require prior NRC approval since it is a correction to an obvious error in referring to the wrong ANSI standard that is not applicable for the equipment in question.

99-054 S-1 stairwell non rated hatch.

Reference Document No.: FSAR
Rev. No: 12
Reference Document Title: Final Safety Analysis Report
Safety Evaluation Description:
This LBIE is being generated to document the acceptability of revising section 9.5 of the FSAR to indicate the presence of a non rated hatch in the barrier separating the S-1 stairwell from the adjoining ventilation shaft.

Safety Evaluation Summary:
The hatch located in the S-1 stairwell is a manufactured steel access door used to allow access through the barrier, and to close off the hatch opening. This LBIE evaluated the affects of the non rated hatch on the DCPP fire protection program, and concluded the ability to utilize the stairwell was not adversely impacted. This LBIE supports the revision to the FSAR.

Conclusion:
The manufactured steel hatch provides adequate separation between the S-1 stairwell and the ventilation shaft such that the stairwell will be useable in the event of a fire. The lack of combustible material in the ventilation shaft, and the tortuous path of travel for any hot gases ensures the stairwell will not be affected by a fire in the shaft. This activity does not result in an unreviewed safety question.

99-055 Radiation Monitor Range and Equipment Location Clarifications

Reference Document No.: A0459852
Rev. No:
Reference Document Title: Review Comments for FSAR Update Section 11.4

Safety Evaluation Description:
As part of the Licensing and Design Basis Alignment Program (LDBAP), a review of the FSAR, DCMs, and Training Guides was performed. As a result of this review corrections were identified for the FSAR Table 11.4-1 radiation monitor locations and ranges. These are clarifications only, no changes to plant physical hardware or software were made.

Safety Evaluation Summary:
R-6, the NSSS Sampling Room Area Monitor, provides continuous indication of ambient radiation levels in the NSSS Sampling Room. The associated Unit 1 display and control module is located in Control Room Cabinet RNRMA.

The upper range limits for the Digital Radiation Monitoring System monitors (for R-14 and 14R, R-15 and 15R, R-24 and 24R, and R-44A and 44B) were clarified to account for the implemented plant overrange or extended range swap over settings at 5,000,000 cpm. The software settings either shutdown the system or transfer it to an extended range to ensure appropriate indications are displayed. Without this feature, pulse pile-up conditions could cause indications to fall erroneously or detector stress could result from the overrange condition. Clarification of the lower range limit for R-24 and 24R and the upper range limit for all the DRMS channels in this FSAR table will be consistent with other documentation, including the Emergency Plan.
The elevation corrections for R-60 to R-69, R-82, and R-83 will allow consistency between the FSAR and the Plant Information Management System (PIMS). Corrections noted are minor, but were identified during the LDBAP review.

Conclusion:
The corrections to the FSAR have been provided and should be implemented. This activity does not result in an unreviewed safety question.

99-056 Valve Operator Shafts Through Barrier 5001

Reference Document No.: FHARE 3
Rev. No: 3
Reference Document Title: Valve Operator Shafts Through Barrier 5001
Safety Evaluation Description:
This LBIE is being generated to document the acceptability of having nonrated manufactured Roto-Hammer valve handwheel extension assemblies in rated fire barriers located in fire areas/zones 14-D/15, and 19-D/18. FHARE 3 is attached to this LBIE and provides the technical basis of the acceptability of the lesser rated configuration.

Safety Evaluation Summary:
The penetrations for the fire areas/zones described above consist of manufactured steel Roto-Hammer valve handwheel extension assemblies used to manipulate the valves in turbine lube oil rooms from the turbine deck. FHARE 3 evaluated the affects of the nonrated penetrations on the DCPP fire protection program, and concluded the ability to achieve and maintain safe shutdown was not adversely impacted. This LBIE supports the FHARE and the associated FSAR change.

Conclusion:
The Roto-Hammer valve handwheel extension assemblies provide adequate separation between the subject fire areas and safe shutdown will not be compromised by the presence of these penetrations due to the automatic and manual fire protection features, lack of continuity of combustibles, and tortuous path of travel for any hot gases. The absence of combustible material nearby will further assures the capability to achieve safe shutdown is not affected. This activity does not result in an unreviewed safety question.

99-057 FHARE 116, Revision 0

Reference Document No.: FHARE 116
Rev. No: 0
Reference Document Title: Testing and Maintenance of Ionization Smoke Detectors
Safety Evaluation Description:
FHARE 116 was written to evaluate the frequency of performing sensitivity testing and maintenance of ionization smoke detectors. The 5-year frequency of performing the tests was *an audit finding* identified in QE Q0011958, "Fire/Smoke Detector Sensitivity Testing and Inaccessibility". In 1993 the frequency of performing the sensitivity testing was changed from annually to every 5 years as a result of a licensing review. Because this was not consistent with the manufacturer's recommendation and NFPA 72D (1975 edition), the basis for the 5-year sensitivity test interval was *an audit finding*. This FHARE re-evaluates the basis for the 5-year sensitivity testing, and it

was judged that the failure rates were consistent with NFPA goals, and that the detector performance would not adversely impact the Fire Protection Program defense-in-depth.

Safety Evaluation Summary:
This FHARE re-evaluates the basis for the 5-year testing, and it was determined that the frequency for several detector zones, *as well as replacement detectors,* will need to be performed at a more frequent interval than 5-years. The functional test (smoke test) have been performed every 6 months in accordance with ECG 18.3, and provide reliability that the smoke detectors will function in the event of a fire. The frequency of cleaning and performing a sensitivity test is based on the environment that the detector is exposed. The review of sensitivity test results from1987 through 1999 have shown that the failure rates were consistent with NFPA goals, and that the detector performance would not adversely impact the Fire Protection Program defense-in-depth.

Conclusion:
A review of the sensitivity test results for smoke detectors have shown that the current 5-year test interval is consistent with NFPA goals, and that the detector performance would not adversely impact the Fire Protection Program defense-in-depth. Performance of the 6-month functional test (smoke test) is consistent with NFPA 72D requirements and maintenance recommendations and ECG 18.3 requirements. By increasing the frequency of sensitivity testing for selected zones and replaced detectors in accordance with performanced-based program requirements, there is reasonable assurance that smoke detection systems at DCPP will function as designed and will be able to provide early detection of a fire during its incipient stage. Therefore, the changes to the sensitivity test frequency will not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. This evaluation does not result in an unreviewed safety question.

99-058 FHARE 06, Revision 3

Reference Document No.: FHARE 06
Rev. No: 3
Reference Document Title: Seismic Gap at Concrete Block Walls

Safety Evaluation Description:
FHARE 06 was written to evaluate the adequacy of the unrated seismic gaps in the 480V Vital Switchgear Rooms, elevation 1 00-ft, in the Units 1 and 2 Auxiliary Buildings at the Diablo Canyon Power Plant (Fire Areas 5A1, 5A2, 5A3, 5A4, 5B1, 5B2, 5B3, and 5B4). FHARE 06 is now being revised to include the evaluation of non-rated penetrants in these gaps.

Safety Evaluation Summary:
The gaps that exist in the barriers of the 480V Vital Switchgear Rooms and the non-rated penetrants in these gaps will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configurations will not adversely affect redundant safe shutdown components, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in the 480V Vital Switchgear Rooms, the fire detection and suppression systems, the location of redundant components, and the availability of offsite power a design basis fire is not expected to adversely affect the ability to achieve safe shutdown. This evaluation does not result in an unreviewed safety question.

99-059 FHARE 103, Rev. 2

Reference Document No.: FHARE 103
Rev. No: 2
Reference Document Title: Fire Barrier Configurations in the Emergency Diesel Generator Areas

Safety Evaluation Description:
FHARE 103 was revised to include the evaluation of non-rated penetrants in the diesel generator fire area and fire zone boundaries (i.e., diesel exhaust stack penetrants, non-rated penetrant in seismic gap, unsealed penetrants). The FHARE was also revised to include the design modifications related to the ventilation exhaust openings for the Unit 1 side (DCP H-49117). The FHARE title was changed to accommodate the added barrier configurations, and the discussion of the configurations was reformatted to simplify the description of the fire barrier configurations.

Safety Evaluation Summary:
The non-rated openings and penetrants which exist in the barriers separating redundant emergency diesel generators will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configurations will not adversely affect redundant emergency diesel generators, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in the diesel intake and exhaust rooms (silencer rooms) and the diesel generator corridor, and the fire detection and suppression systems, the location of redundant components, and the availability of offsite power, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown. This evaluation does not result in

an unreviewed safety question.

99-060 FHARE 133, Revision 0

Reference Document No.: FHARE 133
Rev. No: 0
Reference Document Title: Seismic/Construction Gaps in the 12kV Switchgear Rooms

Safety Evaluation Description:
FHARE 133 was written to evaluate the gaps at the top of the barrier separating the 12kV Switchgear Rooms (Fire Areas 10&20) from the Diesel Generator Corridors (Fire Areas 11D&22C) and the gaps in the barriers separating the 12kV Switchgear Rooms (Fire Areas 10&20) from the Main Transformer Yards (Fire Areas 28&29) in both Units 1 and 2.

Safety Evaluation Summary:
The gaps, which exist in the barriers of the 12kV Switchgear Rooms, will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configurations will not adversely affect redundant safe shutdown components, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in the Diesel Generator Corridors and the 12kV Switchgear Rooms, the fire detection and suppression systems, the location of redundant components, the torturous path between redundant components, the availability of offsite power, and the spatial separation in the Main Transformer Yards, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown. This evaluation does not result in an unreviewed safety question.

99-061 FHARE 30, Revision 3

Reference Document No.: FHARE 30
Rev. No: 3
Reference Document Title: Diesel Intake & Exhaust Silencer Rooms, Common Gaps and Barriers

Safety Evaluation Description:
FHARE 30 was written to evaluate the gaps at the end of the barriers separating the Diesel Intake and Exhaust Silencer Rooms and the non-rated exterior wall of the Turbine Building (Units 1 & 2). It is now being revised to reflect the current plant configuration, include the evaluation of the gaps that run along the top of these barriers, that communicate with the ceiling, and the evaluation of the barrier that separates Fire Area TB- 17 from Fire Zones 23-C- 1, 23-B and 23-A. The FHARE title was changed to accommodate the added barrier configurations, and the discussion of the configurations was

reformatted to simplify the description of the fire barrier configurations

Safety Evaluation Summary:
The non-rated gaps at the ends of the walls separating the diesel intake and exhaust silencer rooms, the gaps communicating between these same barriers and the ceiling above, and the configuration of the barrier communicating between fire zone 22-C-2 and 23-C-1, 23-B, and 23-A will not affect the integrity of these fire barriers to compartmentalize a design basis fire. The configurations will not adversely affect redundant safe shutdown components, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown. This evaluation does not result in an unreviewed safety question.

Conclusion:
Based on the low combustible loading in the diesel intake and exhaust rooms (silencer rooms) and Fire Areas 23-A, 23-B, and 23-C-1, the fire detection and suppression systems available in these areas, the location of redundant components, and the availability of offsite power, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown.

99-062 500 kV Differential Tie Line Protection

Reference Document No.: DCP E-50374
Rev. No: 0
Reference Document Title: 500 kV Current Differential Line Protection
Safety Evaluation Description:
The transmission lines located between the 500 kV switchyard and DCPP's Generator Step-up Transformers were designed with two "Channel A" relay schemes to protect this transmission line section from damage resulting from local electrical faults:

One protection scheme consists of a current differential system comprised of two microprocessor-based differential current relays - one located at the 500 kV switchyard and the other one in the power block. The 500 kV switchyard and the power block differential relays communicate via a fiber optic link (pilot wire). These existing relays were suspect of erroneous actuation and were cut out of service in 1996 based on a recommendation from PG&E's Grid Maintenance and Construction, Protection Engineering Department. The vendor of these devices has not been able to find a specific failure mechanism root cause. The request for removal from service and associated Safety Evaluation screen are documented in AR A0407162.

The other protection scheme consists of line and ground directional over-current relays which are located at the 500 kV switchyard. These simple electro-mechanical relays do not have the intelligence to trip the unit in a

controlled and coordinated fashion. Continued dependence on this protection scheme could result in main generator and / or 500 kV breaker damage during severe transmission system disturbances.

In order to address the existing protection scheme problems mentioned above, the following modifications are necessary:

1. Replace the existing current differential relays (one located in the 500 kV switchyard and the other one located in the power block) with state-of-the art, microprocessor based relays which are not prone to spurious actuation. This LBIE supersedes the LBIE screen previously performed under AR A0407162 which authorized the disabling of the existing current differential relays.

2. Add a new redundant current differential scheme consisting of two new state- of-the-art, microprocessor-based current differential relays (one located in the 500 kV switchyard and the other one located in the power block) and associated communications pilot wire between the power block and the 500 kV switchyard. Existing current transformers (CTs) will be used to feed the new relays. This modification also entails the addition of a small cut-out toggle switch and associated wiring in the Control Room Vertical Board.

3. Add a time delay device to the existing (electro-mechanical) line directional over-current relays. The line and ground directional over-current relays will be normally cut-out of service and not relied upon. These relays will only be placed in service in the event that one of the new redundant current differential schemes were to be out of service. This protection scheme is only a back-up to the new redundant current differential protection schemes. The time delay device will slow down the actuation of the line directional over-current relays in order to optimize their unit trip signal timing so as to minimize possible damage to the main generator and / or 500 kV breakers in the event that these relays are ever relied upon and actuate.

Safety Evaluation Summary:
The new protection schemes perform the same function as the old protection schemes which is to trip the unit in the event that an electrical fault is sensed on the 500 kV transmission lines.

Due to its nature (differential current protection), this project entails modifications both in the power block as well as in the 500 kV switchyard. The power block modifications are addressed under DCP E-50374. The 500 kV switchyard modifications are covered under a separate package issued by PG&E's Grid Maintenance and Construction (GM&C) Engineering

Department. The modifications at the 500 kV switchyard are limited to the installation of two new current differential relays and associated cabinet, alarms and interlocks. The design associated with the 500 kV switchyard current differential relays is similar to the design of their two current differential relay counterparts located in the power block. This LBIE addresses all the modifications associated with this project, including the 500 kV switchyard modifications.

In preparation of the answers to this LBIE, the Safety Evaluation Report (SER) was searched (in addition to the FSAR Update and other supporting documents) using various word searches related to the proposed modifications. No sections were found in the S/SERs which are indicative of any detailed review by the NRC of the 500 kV line fault protection schemes which are affected by the proposed modifications.

Conclusion:
Based upon the safety evaluation performed on the 500 kV Tie Line Differential Protection project, and the detailed review of the criteria used and justifications offered, it was determined that an unreviewed safety question (USQ) is not involved.

99-063 Replace CCP 2-2 and convert pump shaft seal to 3rd generation

Reference Document No.: DCP N-050348
Rev. No: 0
Reference Document Title: Replace CCP 2-2 and convert shaft seal to 3rd generation seal configuration

Safety Evaluation Description:
This design change is to replace the pump casing and internal assembly for CCP 2-2 with like-for-like parts that has been equipped with the 3rd generation seal configuration.

The old seal design (1st generation) was a multi-component assembly requiring external cooling by CCW. CCPs 1-1, 1-2, and 2-1 have already been converted to 3rd generation seals. The 3rd generation seal does not require external cooling. This design change will remove the seal coolers, seal plate coolers and associated CCW piping. This design change will also remove the emergency fire water supply to the Unit 2 seal coolers as it is no longer required for either charging pump. The Q-List will be updated to delete the coolers.

Safety Evaluation Summary:
Replacement of the pump casing and internal assembly for CCP 2-2 are considered like-for-like. Changing the CCP shaft seal to 3rd generation seal only eliminates the external cooling requirement. The pump operation,

performance, and its ability to mitigate an accident are not affected. Elimination of external seal cooling improves the reliability of the CVCS system. Removal of the CCW lines and fire water lines has little impact on the CCW and Fire water systems.

Conclusion:
This design change has no adverse impact the licensing base of the plant. Operation and performance of CCP 2-2 and the CCW and fire water systems are not affected. This activity does not result in an unreviewed safety question.

99-064 Revision 1 to STP M-89A, Void Volume Measurement in SIP/CCP Suction Crosstie Piping

Reference Document No.: STP M-89A
Rev. No: 1
Reference Document Title: Void Volume Measurement in SIP/CCP Suction Crosstie Piping

Safety Evaluation Description:
The procedure revision involves actions to be taken if a void in the ECCS crosstie piping is discovered that is larger than the analyzed allowable volume. The revision directs immediate entry into TS 3.0.3, followed by CAUTION tagging SI-8807A/B, declaring SIP #1 inoperable, entering TS 3.5.2.a for an inoperable ECCS train, exiting TS 3.0.3, venting the void, and exiting TS 3.5.2.a.

Normally closed, remote manual motor-operated valves (MOVs) SI-8807A/B are opened as directed by EOP E-1.3, "Transfer to Cold Leg Recirculation." This procedure change administratively controls these valves such that they would not be opened during the time it takes to vent the void. These actions would (1) prevent a gas void in the SIP/CCP suction crosstie piping from entering the suctions to the SIPs or CCPs, potentially rendering them inoperable and incapable of fulfilling their ECCS function, and (2) result in restoring one ECCS train to operable status. Entry into TS 3.5.2.a also allows additional time to vent the void.

Safety Evaluation Summary:
The proposed revision changes the configuration and controls of valves SI-8807A/B, which are discussed in FSAR Update Table 6.3-5 and manipulated in EOP E-1.3, "Transfer to Cold Leg Recirculation." There is no impact on the injection phase following a LOCA.

Following discovery of a void volume that exceeds the analyzed allowable, tagging the valves such that they are not opened during EOP E-1.3 restores one train of ECCS to operable status. Specifically, upon discovery of a void

volume that exceeds the analyzed allowable, the following actions are required:

- Immediately enter TS 3.0.3,
- CAUTION tag the SI-8807A/B control switches such that they are not opened during switchover from safety injection to cold leg recirculation, thereby restoring one ECCS train to operable status,
- Declare SIP #1 inoperable and entering TS 3.5.2.a for an inoperable ECCS train,
- Exit TS 3.0.3,
- Vent per STP M-89 and successfully perform STP M-89A, and
- Remove CAUTION tags and exit TS 3.5.2.a once STP M-89A test results are acceptable.

These actions would (1) prevent a gas void in the SIP/CCP suction crosstie piping from entering the suctions to the SIPs or CCPs and potentially rendering them inoperable and incapable of fulfilling their ECCS function, and (2) restore one ECCS train to operable status. Once TS 3.5.2.a is entered, a single failure does not have to be postulated; therefore, both trains of ECCS would be capable of injecting into the RCS. In addition, the Action Statement limits the time allowed to restore the second ECCS train, while allowing additional time to vent the void.

EOP E-1.3 directs the operator to open SI-8807A/B and EOP E-1.4 assumes the valves were opened during E-1.3. EOP operator action time will not be impacted by tagging SI-8807A/B, since the operator will proceed to the next step of E-1.3. By keeping SI-8807A/B closed during switchover to cold leg recirculation, one train of ECCS is restored and maintained for recirculation, and required flows will be provided by the operable ECCS pumps. Once in EOP E-1.4, suction to SIP #1 would be isolated, but the pump has already been declared inoperable and is therefore not being relied upon to perform any safety function. The operable RHR, SI, and Charging pumps will be operating normally and providing the required flows; therefore, RWST draindown will be normal and there will be no impact on sump level when recirculation is initiated.

In summary, if a void volume is discovered that exceeds the analyzed allowable, this procedure will (1) prevent the void from rendering both CCPs or both SIPs inoperable, (2) restore one train of ECCS, and (3) vent the void.

Conclusion:
Based on the entire SAR, which includes the living FSAR Update, SER and SSERs, Tech Specs, and associated correspondence, the configuration proposed in the procedure revision is acceptable since it ensures that at least one train of ECCS will be available to fulfill its safety-related function during

recirculation. By entering Action Statement 3.5.2.a, single failure of other components is not required to be postulated, and the ECCS function will remain available for recirculation.

The proposed configuration is allowed by Tech Specs, even though it is different than the sequence outlined in FSAR Update Table 6.3-5 and EOPs E-1.3 and E-1.4. The Tech Specs recognize that many plant configurations exist which may render a single train of ECCS inoperable, and that rendering a single train of ECCS inoperable for a limited period of time is acceptable even though the condition would temporarily not be consistent with the FSAR. The procedure revision will not prevent alignment of the ECCS for cold leg or hot leg recirculation per EOPs E-1.3 and E-1.4. Therefore, no unreviewed safety question is involved.

99-065 Fire Water Supply to the Mechanical Maintenance Trailers

Reference Document No.: DCP M-049483
Rev. No: 0
Reference Document Title: Fire Water Supply to the Mechanical Maintenance Trailers
Safety Evaluation Description:
This DCP provides a source of fire water to the permanent mechanical maintenance trailers located on the 140' turbine deck.

Safety Evaluation Summary:
The mechanical maintenance trailers located on the turbine deck are existing facilities that are used as offices. These buildings do not have a source of fire water supplied to the suppression systems located inside the offices.

Conclusion:
Providing a source of fire water to the mechanical maintenance trailers will protect the personnel inside the offices, and also help prevent a fire originating inside these offices from spreading to other areas of the turbine building. This change does not adversely impact the DCPP fire protection program or the Appendix R analysis.

99-066 Allow Alternate LTOP XMTRs PT-403A/405A for Use in Mode 4

Reference Document No.: DCP J-049480
Rev. No: 0
Reference Document Title: Design Change Package
Safety Evaluation Description:
Allow use of Alternate LTOP Transmitters PT-403A, -405A during Mode 4.
Safety Evaluation Summary:
The alternate LTOP transmitters meet the requirements for Safe Shutdown.

Table 9.5G-1 and Appendix 9.5A have been reviewed. One of three wide range PTs is required: PT-403, -405 and -406. Electrical Engineering has reviewed the cable routing of PT-403A, PT-405A, and PT-406 and has confirmed that a fire in any 1 area will not disable all three signals. Appendix 'R' criteria for safe shutdown can be met with alternate LTOP transmitters in service in Mode 4.

The SAR in Section 5.2.2.3 describes the Low Temperature Over Pressure (LTOP) protection feature of the Plant. This feature is not changed. The alternate transmitters, PT-403A/405A, used to provide the necessary inputs to the LTOP circuitry are left in service during Mode 4. The LTOP feature described in the SAR does not specify the input pressure transmitters by number. It is acceptable to leave the alternate LTOP transmitters in service in Mode 4 because they meet all engineering functional requirements for this service. The transmitters are redundant and do not share a common tap. The taps are from the same RHR suction line from Loop 4. Sharing different taps from the same line is acceptable for this service. The alternate transmitters, PT-403A/405A, are fed from the same redundant power supplies as the normal transmitters, PT-403/405. The difference in location between the alternate and normal transmitters is that PT-403A/405A are located in Containment while PT-403/405 are located in the Auxiliary Building.

Conclusion:
Although PT-403A/405A are located below the flood elevation, there are no Mode 4 accidents that could result in a harsh environment that would result in the loss of these transmitters while the LTOP function is required. This activity does not result in an unreviewed safety question.

99-067 FHARE 119, Revision 0

Reference Document No.: FHARE 119
Rev. No: 0
Reference Document Title: Appendix A to BTP (APCSB) 9.5-1 Plaster Barriers

Safety Evaluation Description:
FHARE 119 was written to evaluate the acceptability of 2-hour rated plaster barriers that had previously been described as 3-hour rated barriers in licensing submittals and the Fire Hazards Analysis.

Safety Evaluation Summary:
The change in rating of the fire barriers from 3-hour to 2-hour will not affect the integrity of the fire barriers to compartmentalize a design basis fire. The affected barriers separate Fire Zone 24E from Fire Zone S6 and Fire Zone 23A from Fire Zone 23E. These configurations will not adversely affect redundant safe shutdown components and the configurations will not involve

a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in all of the fire areas/zones, the available fire detection and suppression systems, the location of redundant components, and the availability of offsite power, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown. This activity does not result in an unreviewed safety question.

99-068 Plaster Blockout Panels in 3-Hour Barriers

Reference Document No.: FHARE 50
Rev. No: 4
Reference Document Title: Plaster Blockout Panels in 3-Hour Barriers
Safety Evaluation Description:
FHARE 50 was revised to limit the scope to just blockout panels. The previous revision (Rev. 3) evaluated configurations which may have been upgraded from past design modifications and removed from the scope of the FHARE. Configurations which involved the entire wall being 2-hours were also removed from the scope of the FHARE because of the licensing impact with lesser rated plaster walls (for Appendix R nad Appendix A walls). These 2-hr rated walls were deleted from FHARE 50 and addressed in new FHAREs 118 and 119.

Because the plaster blockouts would only provide at most a 2-hour fire rating, FHARE 50 evaluates the impact on the 3-hour rated fire barrier.

Safety Evaluation Summary:
This FHARE evaluates fire barrier configurations related to 2-hour rated plaster blockouts located in 3-hour fire barriers. The fire barriers are described in the fire hazards analysis of the FSAR. These configurations do not involve security barriers, or the emergency plan and environmental plan. The fire barriers are also not credited in any of the Technical Specifications.

Conclusion:
This activity does not result in an unreviewed safety question.

99-069 Nonrated Penetrations in the Turbine Building Lube Oil Rooms

Reference Document No.: FHARE 131
Rev. No: 0
Reference Document Title: Non-rated Penetrations in the Turbine Building Lube Oil Rooms
Safety Evaluation Description:
This LBIE is being generated to document the acceptability of having non-

rated pipe penetrations in rated fire barriers located in fire areas/zones 14B/14A, 15/14A, and 18/19A. FHARE 131 is attached to this LBIE and provides the technical basis of the acceptability of the lesser rated configuration.

Safety Evaluation Summary:
The penetrations for the fire areas/zones described above consist of large pipes through sleeved penetrations in the fire barriers separating the in turbine lube oil rooms from the turbine building. FHARE 131 evaluated the affects of the non-rated penetrations on the DCPP fire protection program, and concluded the ability to achieve and maintain safe shutdown was not adversely impacted. This LBIE supports the FHARE and the associated FSAR change.

Conclusion:
The existing non rated piping penetrations provide adequate separation between the subject fire areas and safe shutdown will not be compromised by the presence of these penetrations due to the automatic and manual fire protection features, minimal quantities of combustibles in the Appendix R areas, and the configuration of the existing seals. The absence of Appendix R components and circuits nearby will further assures the capability to achieve safe shutdown is not affected. This activity does not result in an unreviewed safety question.

99-070 Use Of Bobbin Coil To Detect PWSCC n Less Than 2 Volt Dents

Reference Document No.: AR A0485126
Rev. No: 0
Reference Document Title: Change NRC Commitment Regarding S/G Tube Inspection Of < 2 Volt Dents

Safety Evaluation Description:
This change proposes to allow the option to replace the requirement to perform Plus Point inspections of ≤ 2 volt dented intersections in steam generator tubes, and delete associated expansion criteria. Rather, PG&E would like the option to employ and credit the bobbin probe, using EPRI ETSS 96012 analysis techniques, for detection of PWSCC in ≤ 2 volt dented intersections. Plus Point inspections of ≤ 2 volt dented intersections would continue to be conducted when bobbin identifies distorted support plate indications, in order to confirm and characterize the degradation.

The reason for this change is to reduce the time, cost, and radiation dose associated with performing extensive Plus Point inspections during a refueling outage. Approximately 3000 less than 2 volt dented intersections have required inspection under the current criteria in recent refueling outages. These inspections are costly due to inspection time and probe costs, and

subsequently increase radiation exposure to SG workers. Use of the bobbin detection technique should significantly reduce the number of required Plus Point inspections, yet detect significant PWSCC indications in less than 2 volt dents.

Safety Evaluation Summary:
This change proposes to delete the requirement to perform Plus Point inspections of ≤ 2 volt dented intersections, and associated expansion criteria. Rather, bobbin inspections using EPRI ETSS 96012 techniques will be credited for detection of PWSCC in ≤ 2 volt dented intersections. Plus Point inspections of ≤ 2 volt dented intersections would continue to be conducted when bobbin identifies distorted support plate indications, in order to confirm and characterize the degradation.

The current NRC-approved dent inspection program requires the following Plus Point inspections, on a SG basis:

≥ 5 Volt Dent Inspection Criteria

Unit 2: 100% of ≥ 5 volts dents, including hot leg and cold leg.

Unit 1, described below:

- Inspection criteria: 100 percent of ≥ 5 volt hot leg dented intersections up to and including the highest TSP elevation where PWSCC, circumferential indications, or axial ODSCC not detected by bobbin (AONDB) at ≥ 5 volt dents have been previously detected in that SG, plus 20 percent at each higher hot leg TSP elevation. The inspection will always include a 20 percent sample of ≥ 5 volt dented intersections at each hot leg TSP elevation.

- Expansion criteria: If PWSCC, circumferential indications, or axial ODSCC not detected by bobbin at ≥ 5 volt dents are identified in any 20 percent sample, then, in the affected SG, inspect 100 percent of ≥ 5 volt dented intersections at that TSP elevation. If PWSCC, circumferential indications, or AONDB at ≥ 5 volt dents are identified at 7H, then the 20 percent expansion will be applied to the cold leg (7C), and will continue down to the lowest cold leg TSP elevation until a 20 percent sample is obtained that is free from PWSCC, circumferential indications, or AONDB at ≥ 5 volt dents.

- Note: For any 20 percent sample, a minimum of 50 ≥ 5 volt dented intersections in that SG at that TSP elevation will be inspected, or all the ≥ 5 volt dented intersections at that TSP elevation will be inspected if the ≥ 5 volt dent population is less than 50.

2 to 5 Volt Dent Inspection Criteria (same for both units):

- Inspection Criteria: 100 percent of 2 to 5 volt dented intersections (specifically, greater than or equal to 2 volts and less than 5 volts) up to and including the highest TSP elevation where PWSCC, circumferential indications, or AONDB at 2 to 5 volt dented intersections have been previously detected in that SG, plus 20 percent at the next highest TSP elevation.

- Expansion Criteria: If PWSCC, circumferential indications, or AONDB at 2 to 5 volt dented intersections are identified in the 20 percent sample, then, in the affected SG, inspect 100 percent of the 2 to 5 volt dented intersections at that TSP elevation, plus 20 percent at the next highest TSP elevation. This step-wise expansion will be continued down to the lowest cold leg TSP elevation until a 20 percent sample is obtained that is free from PWSCC, circumferential indications, or AONDB at 2 to 5 volt dented intersections.

< 2 Volt Dent Inspection Criteria (same for both units): Criteria same as the 2 to 5 volt dent inspection program, that is:

- Inspection Criteria: 100 percent of < 2 volt dented intersections up to and including the highest TSP elevation where PWSCC, circumferential indications, or AONDB at < 2 volt dented intersections have been previously detected in that SG, plus 20 percent at the next highest TSP elevation.

- Expansion Criteria: If PWSCC, circumferential indications, or AONDB at < 2 volt dented intersections are identified in the 20 percent sample, then, in the affected SG, inspect 100 percent of the < 2 volt dented intersections at that TSP elevation, plus 20 percent at the next highest TSP elevation. This step-wise expansion will be continued down to the lowest cold leg TSP elevation until a 20 percent sample is obtained that is free from PWSCC, circumferential indications, or AONDB at < 2 volt dented intersections.

Inspection of dented intersections using the Plus Point probe was initiated in 1R7 and has been used in 5 subsequent outages. Starting in 2R7, the dent inspection program was expanded to include dents less than 2 volts (i.e., no lower voltage threshold). Approximately 2,000 intersections with less than 2 volt dents were inspected in 2R7 and approximately 3,000 intersections with less than 2 volt dents were inspected in each subsequent outage. The inspection criteria listed above was strictly adhered to in 2R8 and 1R9, when voltage-based ARC was implemented. In 1R7, only dents greater than 2 volts

were inspected. Because of the complexity in calling dents less than 2 volts, low voltage dent populations can fluctuate.

The number of PWSCC indications at dented TSP intersections detected by Plus Point since 1R7 is summarized in Table 1. Table 2 provides a detailed listing of the indications in 1R8, 2R8, and 1R9.

TABLE 1
PWSCC AT DENTED TSP INTERSECTIONS

Dent voltage	1R7	2R7	1R8	2R8	1R9	Total
< 2	34 SAI (34)	28 SAI (15)	95 SAI (32)	12 SAI (1)	47 SAI (7)	216 SAI (89)
2 - 5	30 SAI	23 SAI, 1 SCI	38 SAI	6 SAI, 1 SCI	17 SAI, 1 SCI	114 SAI, 3 SCI
> 5	13 SAI, 4 SCI	22 SAI, 20 SCI	13 SAI, 1 SCI	10 SAI, 5 SCI	5 SAI, 2 SCI	63 SAI, 32 SCI
Total	77 SAI, 4 SCI	73 SAI, 21 SCI	146 SAI, 1 SCI	28 SAI, 6 SCI	69 SAI, 3 SCI	393 SAI, 35 SCI

Notes:
1. SAI = single axial indication, SCI = single circumferential indication
2. Number in parenthesis are number of SAI detected by bobbin in ≤ 2 volt dents.
3. In 1R9, 33 SAI were left in service because they were less than 40% maximum depth using qualified Plus Point sizing techniques.

216 axial flaws (SAI) have been detected in ≤ 2 volt dents, of which bobbin detected 89 (bobbin calling criteria did not use ETSS 96012 analysis techniques). 35 circumferential flaws (SCI) have been detected in > 2 volt dents. Since circumferential flaws initiate at intersections which have larger dent voltages, potential SCIs will be detected by Plus Point during the course of the > 2 volt dent inspection.

Condition Monitoring (1R8, 2R8, 1R9)

Eddy current inspection data for PWSCC at dented TSP intersections over the last three outages (1R8, 2R8, 1R9) are compiled in Table 2. Plus Point profiles from 1R8, 2R8, and 1R9 were adjusted using the procedure described in draft WCAP-15128, so that validated Plus Point sizing results (maximum depth, average depth, length) can be compared. Table 2 lists the locations of the PWSCC indications detected in these outages, orientation (SAI or SCI), dent

voltage, bobbin result, Plus Point voltage, and Plus Point maximum depth, average depth, and length/extent for each SAI.

243 PWSCC SAIs were detected in 1R8, 2R8, and 1R9. The sizes of the SAIs were not significant. Applying the deterministic condition monitoring (CM) methodology in draft WCAP-15128, none of the PWSCC SAIs have leakage potential because all were less than 98% maximum depth, accounting for 19.2% NDE uncertainty. Length and average depths were compared against Figure 7-1 of draft WCAP-15128, which provides the EOC 95%/95% structural limit curve for partial depth axial PWSCC, accounting for uncertainty in NDE and material properties. All indications have large margins against burst relative to 1.4DpSLB. Therefore, all axial PWSCC passed CM requirements.

Assessment of Bobbin Detection of PWSCC at ≤ 2 Volt Dents (1R8, 2R8, 1R9)

Bobbin had limited success in detection of PWSCC in ≤ 2 volt dents, detecting 40 out of 154 SAIs in 1R8, 2R8, and 1R9. Based on Table 2, the structurally limiting SAIs in ≤ 2 volt dents that were not detected by bobbin are summarized in Table 3 below:

TABLE 3
STRUCTURALLY LIMITING SAI DETECTED IN 1R8, 2R8, 1R9 THAT WERE NDD BY BOBBIN

Outage	SG	R	C	Loc	+P	Crack No.	Dent volt	BC	+P Max. Volts	AD (%)	Length (in.)	EOC AD%	EOC L in.
1R8	11	23	64	1H	SAI	1	1.06	NDD	1.40	38	0.31	60	0.39
1R8	12	27	69	1H	SAI	1	1.15	NDD	1.93	44	0.39	66	0.47
1R8	12	5	87	5H	SAI	1	1.23	NDD	1.21	39	0.48	61	0.56
2R8	22	12	50	1H	SAI	1	0.99	NDD	1.9	31	0.40	50	0.47
1R9	12	11	27	1H	SAI	1	1.89	NDD	0.74	36	0.43	56	0.50

Using ETSS 96012 analysis techniques, it is likely that these limiting flaws would have been detected by bobbin. Nonetheless, if it is assumed that these flaws were not inspected by Plus Point, would not have been detected by bobbin and, therefore, would have been left in service, a theoretical deterministic operational assessment can be performed. The theoretical EOC depths and lengths are shown in Table 3 under "EOC" columns, applying 95% cumulative growth rates per EFPY for average depth and length (defined as 13.3% and 0.047 inch, respectively, per draft WCAP-15128). The EFPY for U1C9, U1C10, and U2C9 are 1.62, 1.51, and 1.46, respectively. Based on these EOC lengths and average depths, Figure 7-1 of draft WCAP-15128

shows that the theoretical EOC flaws have large structural margin against 1.4DpSLB.

Assessment of Bobbin Detection of PWSCC at ≤ 2 Volt Dents (1R9 Re-analysis using ETSS 96012)

To determine the potential impact of using ETSS 96012 bobbin analysis techniques, the 1R9 bobbin data for the ≤ 2 volt dents that were inspected by Plus Point (over 3000 intersections) were re-reviewed by blind analysis, i.e., the Plus Point results were not known. Four analysts (two production and two resolution analysts, representing primary and secondary) successfully completed an ETSS 96012 site specific performance demonstration (SSPD) test using the data set employed in the performance test documented in draft WCAP-15128. Based on the results of the bobbin re-analysis, the limiting 1R9 flaw listed in Table 3 was detectable by bobbin. The largest SAI in a < 2 volt dent that was undetected by bobbin was located in SG 11 R22C76 at 3H. This SAI had a maximum depth of 47%, an average depth of 33%, and a length of 0.15 inch. If it is assumed that this flaw was not inspected by Plus Point, would not have been detected by bobbin using ETSS 96012 techniques, and, therefore, would have been left in service, a theoretical deterministic operational assessment indicates that the average depth and length at EOC could be 53% and 0.22 inch, accounting for growth. Figure 7-1 of draft WCAP-15128 shows that this theoretical EOC flaw has large structural margin against 1.4DpSLB. The flaw could have a theoretical EOC 10 maximum depth of 89.6% based on 95% cumulative growth rate of 15.5% per EFPY (for maximum depth) and NDE uncertainty of 19.2% (for maximum depth). No SLB leakage would be expected from this flaw at EOC because flaws less than 98% deep are not expected to leak per draft WCAP-15128.

Based on the results of the re-analysis, a bobbin probability of detection (POD) curve was generated using Plus Point maximum and average depths (adjusted to the mean NDE uncertainty) as truth. Because the ETSS 96012 re-analysis data set did not include several larger Plus Point voltage SAIs (not all data was re-analyzed due to time constraints), and to provide statistical relevance at larger depths, the POD curve assumed that several larger voltage SAIs (> 0.6 volts by Plus Point) would have been detected by bobbin based on the results of expert opinion. The basis for including > 0.6 volt indications in the POD curve is because all re-analyzed > 0.6 volt indications were detected by bobbin as part of the ETSS 96012 re-analysis. The resulting POD curve was compared against the POD curve from the performance test results documented in draft WCAP-15128, which uses destructive exam depths as truth. The curves yield similar results. Therefore, the draft WCAP POD curve may be applied at DCPP Units 1 and 2. The draft WCAP POD curve shows that the 95% POD values for average depth and maximum depth are 35% and 45%, respectively. Projecting

the 45% maximum depth to an EOC depth will not result in exceeding 98% maximum depth, accounting for 95% CDF growth rate and NDE uncertainty, and a projected cycle length up to 2.18 EFPY. Projecting the 35% average depth to an EOC depth will not result in exceeding the 95/95 structural limits for 1.4DpSLB for flaw lengths up to 1.0 inch, accounting for 95% CDF growth rate and NDE uncertainty, and a projected cycle length up to 2.18 EFPY.

Revised Inspection Criteria

Based on application of ETSS 96012 bobbin detection criteria for axial PWSCC in ≤ 2 volt dented intersections, the revised eddy current inspection criteria that may be implemented starting in 2R9 is as follows:

- 100% bobbin coil inspection of all TSP intersections (no change)
- Plus Point coil inspection of all bobbin coil indications at dented TSP intersections (no change)
- For Unit 2, Plus Point inspection of 100% of ≥ 5 volt dents (no change)
- For Unit 1, Plus Point inspection of ≥ 5 volt dents as described earlier (no change)
- On a SG basis, Plus Point coil inspection of all TSP intersections having > 2 volt dents up to the highest TSP elevation where PWSCC, circumferential indications, or AONDB has been detected, plus 20% of > 2 volt dents at the next highest TSP.

If axial ODSCC not detected by bobbin (AONDB) is found in ≤ 2 volt dents, then Plus Point inspection of ≤ 2 volt dents is not required. Basis: 60% POD that is applied in the Monte Carlo is sufficient to cover undetected axial ODSCC flaws that could be masked in small dents. Note: AONDB determination may be based on either (a) Lead Level III review of the bobbin data or (b) tube integrity engineering review, if a correlation exists between Plus Point and bobbin voltages.

Conclusion:
This change proposes to eliminate performance of Plus Point inspections of ≤ 2 volt dented intersections in steam generator tubes, and associated expansion criteria. Rather, bobbin inspections using EPRI ETSS 96012 techniques will be employed and credited for detection of PWSCC in ≤ 2 volt dented intersections. Plus Point inspections of ≤ 2 volt dented intersections would continue to be conducted when bobbin identifies distorted support plate indications, in order to confirm and characterize the degradation.

Westinghouse report SG-99-01-004 provides a detailed EPRI Appendix H qualification package supporting the bobbin detection technique. The technique has also been validated based on a performance test conducted in 1998. Draft WCAP-15128 provides the results of the performance test and

resulting POD evaluation. Based on a review of past inspection results at DCPP Units 1 and 2, the bobbin POD in draft WCAP-15128 is applicable to DCPP. The draft WCAP POD curve shows that the 95% POD values for average depth and maximum depth are 35% and 45%, respectively . The POD will support future DCPP tube integrity assessments accounting for PWSCC in $\leq$ 2 volt dents.

Use of the bobbin detection technique should significantly reduce the number of required Plus Point inspections, reducing time, cost, and radiation exposure to workers.

99-071 Elimination of Unnecessary QEs for ECGs

Reference Document No.: ECG 0.0
Rev. No: 4XPR
Reference Document Title: Equipment Control Guidelines Applicability
Safety Evaluation Description:
Equipment Control Guideline (ECG) 0.3 is being revised to replace the requirement to initiate a Quality Evaluation (QE) with a requirement to initiate a quality problem action request (AR), when an ECG is not met.

Safety Evaluation Summary:
ECG 0.3 is being revised to delete the requirement to automatically initiate a QE when an ECG is not met. Instead, a quality problem AR will be initiated as appropriate. Quality problem ARs include "A" type ARs which require verification of completion of corrective actions, "Y" type ARs which require initiation of a QE, and "R" type ARs which require initiation of an non-conformance report (NCR). This revision is being made to eliminate initiation of QEs for problems that do not warrant this level of evaluation. There is no change in the requirements for initiating valid QEs. If a problem warrants a QE, a QE will be initiated in accordance with the requirements of 0M7.ID1 and OM7.ID2.

Conclusion:
Revision of ECG 0.3 to eliminate the requirement to automatically initiate a QE for non-compliance with ECGs is acceptable from a licensing basis impact, and does not involve an unreviewed safety question.

99-073 LBIE for DCM S-17B Revision, Brine Line Discharge to ASW

Reference Document No.: A0425609
Rev. No: N/A
Reference Document Title: Brine Line Discharge
Safety Evaluation Description:
A0425609 documents that the "Brine Line" discharge to the ASW 1-1 forebay

is not included in DCM S-17B. The brine line discharges brine and filter backwash from the Seawater Reverse Osmosis System (Makeup Water System) and treated sewage overflow from lift station 11 into the ASW forebay, downstream of the traveling water screens, upstream of the ASW gates and pumps.

Safety Evaluation Summary:
Per the evaluation in AR A0425609 there is no material in the brine line flow or the sewage lift station overflow that would impact the operability or function of the ASW pumps or the CCW heat exchangers. The brine and the chemicals discharged with SWRO filter backwash will not impact ASW operability either directly through plugging or indirectly through chemical or biological action. The sewage lift station overflow is not normally in service, but the secondary effluent has been evaluated as not impacting ASW system operability or causing equipment degradation.

Conclusion:
The existing configuration where the brine line discharges to the unit 1 ASW pump forebays does not impact the system design or licensing basis. This activity does not result in an unreviewed safety question.

99-074 Fire Protection System Performanced Based Surveillance Program

Reference Document No.: ECGs 18.1 thru 18.5 & AD13.DC6
Rev. No: Various
Reference Document Title: Fire Protection ECGs and Fire Protection System Performanced Based Surveillance Program

Safety Evaluation Description:
The ECGs for Fire Protection systems (not to include 18.7 Fire Rated Assemblies) are being changed to allow the surveillance interval to be modified under the control of the Fire Protection System Performanced Based Surveillance Program (AD13.DC6). This program requires evaluation of past performance results, limits the magnitude of any increase of the surveillance interval and requires evaluation of the subsequent results of the equipment/system performance under the new surveillance frequency by the new procedure. Adverse trend in performance requires adjustment of the interval to ensure system availability and readiness is maintained.

Safety Evaluation Summary:
The Fire Protection System Performanced Based Surveillance Program provides for evaluating the results of the current surveillance program as it affects a given component, class of components or system to determine the failure rate of those surveillances. This failure rate can be considered a measure of the reliability or availability of the equipment when called upon to

perform its intended function. In some cases, the measured failure rates are so low, that unwarranted effort is being spent testing or maintaining the equipment with no gain in reliability or availability. In fact the testing and handling of the equipment may contribute negatively to the overall failure rate and unavailability. This program allows for adjusting the surveillance frequency after proper evaluation and justification, to maximize the results of the surveillance program. The surveillance results subsequently obtained after adjusting the frequency, are formally monitored to ensure that equipment failure rates are not negatively impacted. If the failure rates start to increase, then the program requires readjustment of the frequency to prevent component or system degradation.

Conclusion:
This Program when implemented should enhance system maintenance and availability. Fire Protection System performance should remain at levels equivalent to industry expectations and equivalent to prescriptive program requirements. This activity does not result in an unreviewed safety question.

99-075 FHARE 139, Revision 0

Reference Document No.: FHARE 139
Rev. No: 0
Reference Document Title: Unrated HVAC Duct Seal Penetrant in Cable Spreading Room

Safety Evaluation Description:
FHARE 139 was written to evaluate the adequacy of the HVAC duct penetration, in the barrier separating the Unit 2 Cable Spreading Room (Fire Areas 7B) from Stairwell S-5 (Fire Zone S-5), to prevent the propagation of a design basis fire and not adversely affect the ability to achieve and maintain safe shutdown of the plant.

Safety Evaluation Summary:
The HVAC duct penetration, which exists in the barrier separating the Unit 2 Cable Spreading Room from Stairwell S-5, will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configuration will not adversely affect redundant safe shutdown components or circuits and will not adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the lack of combustibles in Fire Zone S-5, the fire detection and suppression systems in Fire Area 7-B to suppress the moderate combustible loading, the alternative shutdown methodology employed for a fire in the cable spreading room, the non-rated duct penetrant is not expected to adversely affect the ability to achieve and maintain safe shutdown conditions. This activity does not result in an unreviewed safety question.

99-076 FHARE 118, Revision 0

Reference Document No.: FHARE 118
Rev. No: 0
Reference Document Title: Appendix R Fire Area Plaster Barriers
Safety Evaluation Description:
FHARE 118 was written to evaluate the acceptability of 2-hour rated plaster barriers that had previously been identified as 3-hour rated barriers and credited as Fire Area boundaries satisfying 10CFR50, Appendix R fire area separation.

Safety Evaluation Summary:
The change in rating of the Appendix R fire barriers from 3-hour to 2-hour will not affect the integrity of the fire barriers to compartmentalize a design basis fire. The affected barriers separate: (Fire Area 3-T-1 from Fire Zones 3x and 3; Fire Zone 24E from Fire Area 24D and Fire Zones 23A and 23E; Fire Zone 23C from Fire Zone 23-C-1; Fire Area 4-A from 4-A-1; and Fire Area 6-A-5 from 6-B-5). These configurations will not adversely affect redundant safe shutdown components and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown. The low combustible loading and available fire protection features provide assurance that the 2-hr rated plaster walls will be able to maintain the fire within the affected fire area.

Conclusion:
Based on the low combustible loading in all of the fire areas/zones, the fire detection and suppression systems, the location of redundant components, and the availability of offsite power, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown. This activity does not result in an unreviewed safety question.

99-077 Revise GL 88-05 Commitment Regarding Boric Acid Walkdowns

Reference Document No.: STP R-8C
Rev. No: 4
Reference Document Title: Containment walkdown for evidence of boric acid leakage.
Safety Evaluation Description:

1) Changed reference to "forced outage of sufficient duration" to "near the beginning of every refueling outage and during other outages of sufficient duration if at least 92 days have elapsed since the last performance of STP R-8A or R-8C".
2) Added precautions on limited access controls and entries due to radiological conditions.
3) Added reminder to inspect pressurizer cubicle and Reactor Coolant

Pump seal areas.
4) Updated organizational references.

Safety Evaluation Summary:
The probability of an accident will not be increased by limiting boric acid inspections to no sooner than 92 days after the last outage or implementation of STP R-8A. The present procedure is indefinite with regard how often the walkdowns/inspections for boric acid leaks need to be performed. From past experience the most likely failures due to boric acid corrosion would be mechanical component failures. Complete failure of these types of components due to corrosion would be enveloped by the LOCA scenario. A more likely scenario would be a partial failure of the mechanical components. The resulting leakage would be noted and investigated. Other items of a critical nature have redundancy built into their systems. It is unlikely localized corrosion would make more than one of the components inoperable.

Changing the frequency of the required entries will not increase the probability of an analyzed accident from occurring. Changing the frequency to no sooner than 92 days after the preceding outage will eliminate the extra entries and radiation exposure should we have back to back outages.

Any inspection performed during the fuel cycle will decrease the probability of a malfunction occurring because leaks will be found sooner than they would had the unit gone the entire cycle without any interruptions.

Conclusion:
The worst case for length of time between inspections is still an entire fuel cycle. This has not changed. Therefore the worst case possible is unchanged. This activity does not result in an unreviewed safety question.

99-078 Revised Containment Integrity Analysis (CIA)
Reference Document No.: DCP N-049432
Rev. No: 0
Reference Document Title: DESIGN CHANGE PACKAGE
Safety Evaluation Description:
This DCP revises the DCPP Containment Integrity Analysis (CIA) results in FSAR 6.2C due to a Westinghouse analysis error as documented in PGE-96-587, and it also implements additional design document revisions which were not performed when the revised Containment Analyses were first incorporated into the DCPP licensing basis. DCPP informed the NRC of the original CIA licensing basis update per PG&E letter DCL-95-265, which included replacing FSAR Appendix 6.2B with a new Appendix 6.2C per a LBIE 50.59 approved by the PSRC 1/16/96. However, in the process of incorporating the Westinghouse analysis error correction, additional design documents were

identified as requiring revisions to ensure consistency with the latest Containment Integrity Analysis. Since the affected documents included several DCM revisions, two design calculation revisions, an FSAR update, and a Tech Spec bases revision, it was determined that a DCP along with a LBIE 50.59 should document these changes.

Safety Evaluation Summary:
The 50.59 evaluation establishes that the revised Containment Integrity Analyses (CIA) including the error correction, actually increase the available safety analysis margin for the peak pressure limit as well the post accident temperature and pressure profiles for EQ component evaluation. The revised CIA use existing approved methodology but gain margin through more physically accurate modeling enhancements. The analysis assumptions are consistently conservative with the previous analyses, except for a few cases where a more realistic equipment performance has been credited based on documented justification. None of the analysis revisions involve a physical change in plant equipment or operations, and no new equipment performance or safety function requirements have been imposed. The design document revisions accurately reflect the latest analysis methodology and results and have been verified to be consistent with the existing DCPP design basis and the ITS submittal.

Conclusion:
The revised CIA demonstrate acceptable containment integrity and post accident EQ pressure and temperature results. The analysis methodology and assumptions are consistent with the DCPP design basis and licensing basis and the results are conservatively bounding for DCPP. The revised CIA do not introduce an Unreviewed Safety Question and are acceptable to be incorporated into the DCPP design basis.

99-079 Remove PASS Containment Air Sample Carts

Reference Document No.: DCP M-049465
Rev. No: 0
Reference Document Title: Remove PASS Containment Air Sample Carts
Safety Evaluation Description:
The Post Accident Sampling System (PASS) containment air sample carts were originally installed as a part of the Sentry system to automatically take three undiluted containment air samples in shielded carts with a fourth cart as backup. The carts are identified as SF-1, SF-2, SF-3, and SF-4. In 1983 a diluted grab sample loop was installed. The loop consists of removable filters for containment air particulates and radio iodines and a flask for noble gasses. The sample flask is identified as SF-5. Since a method could not be determined for analyzing the samples in the sample carts, sample flask SF-5 was identified as the preferred method of obtaining containment air samples.

Since containment air sampling using SF-5 fully meets the NUREG-0737 requirements for post accident sampling, the carts will be removed to eliminate any confusion about the current post accident sampling capabilities.

This change removes the PASS containment air sample carts and abandons the solenoid valves and piping to the carts. The carts will be physically removed and discarded. The quick disconnect fittings on the Containment Air Sample Panel (CASP) will be removed and the lines will be capped. The electrical power to the solenoid valves that feed the sample carts will be de-energized and the valves and piping to the carts will be abandoned in place.

Safety Evaluation Summary:
The original design of the Sentry PASS CASP included the four sample carts with the capability to automatically take undiluted containment air samples in removable carts. Because of the inability to analyze the samples in the carts, a diluted containment air sample loop was designed and installed. This diluted containment air sample loop became the primary sample location for PASS sampling of containment air. Since the diluted containment air sample loop fully meets the containment air sampling requirements of NUREG-0737, the sample carts will be removed. This will eliminate the need to maintain the carts and develop the capability to analyze the samples.

<u>Compliance to NUREG-0737, Section II.B.3 (Only sections applicable to the containment air sampling requirements are included):</u>

(1) *The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.*

Compliance with this criteria has been successfully demonstrated through periodic implementation of STP G-14, "Operability Determination of Post Accident Sampling Program" using the diluted containment air sample loop.

(2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within 3-hour time frame established above, quantification of the following:

(a) *certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gasses; iodines and cesiums, and nonvolatile isotopes);*

Compliance with this criteria has been successfully demonstrated through periodic implementation of STP G-14, "Operability Determination of Post Accident Sampling Program" using the diluted containment air sample loop.

(b) *hydrogen levels in the containment atmosphere;*

Measurement of hydrogen concentration in containment using the PASS is no longer required (ref. Letter from the NRC dated February 5, 1998, Chron No. 234942).

(3) *Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.*

The containment air sampling system is self contained system with no isolated auxiliary system ties (ref. DCM S-11, drawings 102011, sheet 6 and 108011, sheet 6).

(6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR 50) (i.e., 5 rem whole body, 75 rem extremities).

Compliance to this criteria has been demonstrated by dose calculation STA-097.

(9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) *Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure.*

Regulatory Guide 1.3 applies to BWR's, 1.7 addresses hydrogen monitoring which, as stated above is no longer required by the PASS. The levels corresponding to the source terms given in Regulatory Guide 1.4 were addressed in the design of the diluted sample loop (ref. IOM to Mr. G. A. Tidrick, dated October

9, 1981 rloc 02624-2895).

(b) *Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.*

The ventilation system for the PASS keeps the sample panels at negative pressure relative to the PASS room to contain any potential leaks. The exhaust for this ventilation and the sample station are filtered through charcoal and HEPA filters.

(11) In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

(a) *Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.*
The diluted containment air sample loop was connected to the existing containment air sample line upstream of FCV-1102 which supplies sample cart SF-1. The sample system provided by Sentry Equipment Corp. is used and has the same provisions for purging sample lines, minimizing sample loss and distortion, and returning the residues of sample collection back to the containment (ref. CAP P-3, "Sentry Post Accident Sampling System Containment Air Sampling and Analysis").

The sample collection point and configuration was designed to obtain a representative sample of the containment atmosphere following a transient or accident and to minimize potential debris clogging of the sample lines (ref. Drawings 102023, sheet 4 and 108023, sheet 4).

Provisions for minimizing sample line length and for reducing plateout in sample lines were considered in the design of the diluted sample loop (ref. DCN DC0-E-M-1104).

After analysis, the containment air samples are disposed of as low-level radioactive waste per procedure RCP RW-1, "Collection and Packaging of Low-Level Radioactive Waste."

(b) *The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.*

The sampling system station contains both charcoal and HEPA filters (ref. Drawings 102023, sheet 7 and 108023, sheet 7).

Conclusion:
This change does not involve an unreviewed safety question. Removal of the sample carts has no potential to initiate or affect the initiation of an accident. This is a nonsafety-related system that has no accident mitigating function. The PASS containment air sampling is used as an aid to determine extent of core damage following a LOCA as is required by NUREG-0737. The technical specifications require sampling of containment air which is met by the diluted containment air sample loop. The removal of the carts have no affect on the technical specifications or their bases.

99-080 Revise Q-List to Add Two New Line Items

Reference Document No.: Q-List
Rev. No: 19
Reference Document Title: Classification of Structures, Systems and Components for DCPP

Safety Evaluation Description:
Two new line items are to be added to the Q-List as follow:

1. Auxiliary Saltwater (ASW) Vacuum Breaker Vaults, which are adjacent to the non-safety related QA Class S Circulating Water (CW) Conduits and extended from the ground surface to below the buried safety-related ASW pipes and tie into the CW Conduits. The buried safety-related ASW pipes penetrate the nonsafety -related Vacuum Breaker Vaults walls. The vault configurations are shown on Drawings 4005022 and 438098.
2. Concrete tunnel pipe vaults that span between the outdoor water storage tanks (OWSTs) located adjacent to, and east of the Auxiliary Building, to the Auxiliary Building below grade. These pipe vaults are shown on Drawings 438033, 463988 and 463989.

These vaults shall be classified as Design Class II, QA Class S, to ensure that their seismic configuration are maintained and controlled.

Safety Evaluation Summary:
Since portions of the ASW Vacuum Breaker Vaults and the piping vaults associated with Design Class I OWSTs house, support, or protect Design Class I SSCs (Ref. Civil Maintenance Rule Data Files CMR 3.5 and 2.8), control of the vault seismic configuration via QA Class S (Ref. IDAP CF3.ID11) is necessary to ensure operability of the Design Class I SSCs during abnormal conditions, including design bases seismic or Hosgri events. In addition, since the ASW Vacuum Breaker Vaults connect to the QA Class S CW Conduits, control of the vault seismic configuration is necessary to ensure operability of CW Conduit during abnormal conditions. Calculation 52.21.2.1 documents the seismic qualification for the ASW Vacuum Breaker Vaults.

Also, for the piping vaults adjacent to the Auxiliary Building, a critical function of the nonsafety-related eastern PWST vault is to maintain its structural integrity during a Hosgri event to ensure that it can provide adequate bearing support to the Design Class II PWST and the Design Class I Auxiliary Building (Ref. Calculation 52.21.11.2). Consequently, the PWST pipe vault should also be placed under seismic configuration management control via IDAP CF3.ID11.

Conclusion:
The Q-List will be updated to include two new line items:

- I.D.1.7 - ASW Vacuum Breaker Vaults
- I.M.14 - Piping Vaults for Outdoor Water Storage Tanks East of Auxiliary Building

These vaults shall be classified as Design Class II, QA Class S, to ensure that their seismic configuration are maintained and controlled. This activity does not result in an unreviewed safety question.

99-081 Non-rated HVAC Penetrations

Reference Document No.: FHARE 136
Rev. No: 1
Reference Document Title: Non-rated HVAC Penetrations
Safety Evaluation Description:
FHARE 136 was written to evaluate the adequacy of the HVAC duct penetration, in the barriers separating the Fire Areas/Zones 3B2/3C, 13D/12A, 13E/12B, and 24D/23A to prevent the propagation of a design basis fire. FHARE 136 also evaluates the impact of the non rated penetrations on the ability to achieve and maintain safe shutdown of the plant, and the affect on

safety related equipment.

Safety Evaluation Summary:
The HVAC duct penetrations which exists in the barriers will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configuration will not adversely affect redundant safe shutdown components or circuits and will not adversely affect the ability to achieve and maintain safe shutdown. The existence of these penetrations will also not adversely affect the operation of any safety related components or systems.

Conclusion:
Based on the lack of combustibles in the Fire Areas/ Zones, the existence of fire detection and suppression systems, and the configuration of the penetrations, the non-rated duct penetrations will not adversely affect the ability to achieve and maintain safe shutdown conditions. This activity does not result in an unreviewed safety question.

99-082 FHARE 30, Revision 4

Reference Document No.: FHARE 30
Rev. No: 4
Reference Document Title: Diesel Intake & Exhaust Silencer Rooms, Common Gaps and Barriers

Safety Evaluation Description:
FHARE 30 was revised to evaluate the gaps and non-rated features located in the barrier that communicates between fire zone 24-E (Elev 107-ft of the Ventilation Equipment Room) and fire zone 22-C-2 (EDG 2-3 diesel silencer area). The barrier configurations are similar to other non-rated barrier features along the walls of fire zone 22-C-2 which have been evaluated in previous revisions.

Safety Evaluation Summary:
The non-rated penetration configurations along the wall separating fire zones 24-E and 22-C-2 will not affect the ability of the wall to maintain a 3-hr rating and effectively compartmentalize a design basis fire. The configurations will not adversely affect redundant safe shutdown components, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in both fire zones 22-C-2 and 24-E, the available fire protection features, the location of redundant components and availability of offsite power, the non-rated penetration seal configurations would not affect the 3-hr rating of the barrier. The ability to achieve and maintain safe shutdown would not be adversely affected by the non-rated fire

barrier configurations. This activity does not result in an unreviewed safety question.

99-083 STP M-89B, Venting RHR to SI Pumps in Mode 1-4

Reference Document No.: STP M-89B
Rev. No: 0
Reference Document Title: Venting RHR to SI Pumps in Mode 1-4
Safety Evaluation Description:
This procedure vents the portion of the RHR/SI piping immediately upstream of SI-8804B, which may contain voids due to the lack of venting capability when SI-8804B is closed. As a result of gas forming upstream of SI-8804B, an SIP could be rendered inoperable if the gas void exceed 0.44 ft^3.

The proposed procedure bypasses the interlocks on valve SI-8804B by manually stroking SI-8804B open. PSRC Interpretation 87-04 Rev. 2 states that opening valve SI-8804B is allowed in Modes 1-3 without entering TS 3.0.3. SI-8804B is discussed in FSAR Update Table 6.3-5 and manipulated in EOP E-1.3, and SI-8923B is manipulated in EOP E-1.4.

Safety Evaluation Summary:
NCRs N0002076 and N0002095 discuss problems with ECCS venting and voids in the ECCS piping. This procedure involves actions taken to preclude or alleviate voiding in the RHR/SI piping immediately upstream of valve SI-8804B. AR A0487099 was written to document the possibility of voids due to an inadequate piping arrangement for venting upstream of SI-8804B. PSRC Tech Spec Interpretation 87-04 was issued to identify valves which opening or closing could place the unit in TS 3.0.3. SI-8804B was listed as being required to remain in the closed position during Modes 1-3. However, Rev 2 to 87-04 has been approved to allow this valve to be opened under administrative controls during Modes 1-3. This STP provides those administrative controls. By removing SI Pump 2-2 from service and isolating SI-8923B , SI-8920B, SI-8921B and the isolation valve for PI-939, the concern with opening SI-8804B is removed since concerns with affecting the RHR flowpath are alleviated. Having the SI-126 vent valve open in Modes 1-4 is no different than any other STP M-89 ECCS venting action. The SI piping is designed and analyzed for pressures exceeding RHR pressure. This method, manually stroking open and powering closed SI-8804B, is used for opening the valve in STP M-86A1. In addition, 8804B is included in OP O-9, Attachment 9.2, as a Limitorque operator, torque closed valve. OP O-9 specifically says that the concerns of short stroking are not applicable if the valve is stroked closed electrically from the open position, which this procedure does. Opening the valve manually will not affect the interlocks on the valve if it is stroked closed electrically as provided for in this STP.

Isolating SIP 2-2 for the performance of this procedure still provides for operation of SIP 2-1 during the injection phase and during Cold Leg and Hot Leg Recirculation, EOP E-1.3 and E-1.4, by means of the flow from the RHRP 2-1 through SI-8804A. Adequate flow to satisfy the accident analyses is provided by operation of either ECCS train during injection and during Hot leg and Cold Leg Recirculation.

RHRP 2-2 will not be running during the performance of this procedure so the head on the system is provided by the RWST. This head is adequate to push any void out through the opened vent valve, SI-126. The piping and components in the SI system are rated to allow pressurization from the RHR pump, should it start. The one component in question because of calibration concerns of rapid pressurization, PI-939, is isolated by the procedure. The high point vent downstream of SI-8804B will allow the void to be vented without challenging the SI piping or the safety of the plant or public. This activity does not result in an unreviewed safety question.

Conclusion:
Based on a review which includes the living FSAR Update (including text, tables, and figures), SER and SSERs, Tech Specs, and associated correspondence, the configuration proposed in this procedure is acceptable since it ensures that at least one train of ECCS will be available to fulfill its safety-related function. By entering Action Statement 3.5.2.a, single failure of other components is not required to be postulated. The plant is analyzed for a single failure of an entire ECCS safety train. The isolation of a single SI pump is bounded by that analysis (UFSAR 15.4.1.1.1.1). Therefore, ECCS function will remain available during the time required to restore the degraded condition.

The proposed configuration is allowed by Tech Specs, even though it is contrary to FSAR Update Table 6.3-5 and EOPs E-1.3 and E-1.4. The Tech Specs recognize that many plant configurations exist which may render a single train of ECCS inoperable, and that rendering a single train of ECCS inoperable for a limited period of time is acceptable even though the condition would temporarily not be consistent with the FSAR.
This procedure will not prevent alignment of the ECCS for cold leg or hot leg recirculation per EOPs E-1.3 and E-1.4.

99-084 DCP J-49373, Change ERFDS/SPDS MUX 1-10 Input Cards Classification

Reference Document No.: J-49373.
Rev. No: 0
Reference Document Title: Change ERFDS/SPDS MUX Input Cards Classification.

Safety Evaluation Description:
The signal input cards in the ERFDS/SPDS MUX'S 1-10 have several different Classifications. A clarifying note will be added to the first page of Instrument Schematics 102031, 102032, 102033, 102034,102035, 102036 and 102931stating that the ERFDS/SPDS input cards are Electrical isolation devices and are Class IE. The ERFDS input cards are shown for convenience with the associated instruments. No changes will be made to the individual Instrument Schematics.

The Instrument Classification for ERFDS/SPDS input cards in the Q-List and PIMS can be eliminated. These devices are QA Class Q , Electrical Class IE devices and do not have a Reg. Guide. 1.97 function. The MUX 11 receives non safety-related inputs, but will be managed as Class IE equipment. All input cards were purchased Class IE and will be maintained as Class IE devices.

Safety Evaluation Summary:
The signal input cards in the ERFDS/SPDS MUX'S 1 THRU 10 have several different Instrument Classifications. The Q-List shows the input cards for MUX'S 1 THRU 10 Instrument Class IB and the Instrument Schematics show the MUX'S 1 THRU 10 input cards classified variously as Class IA, IB, and ID. In PIMS, the MUX'S 1 THRU 10 have various Instrument Classifications. MUX 11 receives non safety-related inputs but will be managed as Class I equipment. All MUX input cards were bought Class IE and will be maintained Class IE.

A clarifying note will be added to the first page of Instrument Schematics 102031, 102032, 102033, 102034,102035, 102036 and 102931 stating that the ERFDS/SPDS input cards are Class IE Electrical isolation devices. The ERFDS input cards are shown for convenience with the associated instruments. No changes will be made to the individual Instrument Schematics. The Instrument Classification for ERFDS/SPDS input cards in the Q-List and PIMS can be eliminated. These devices are QA Class Q , Electrical Class IE devices and have no Reg. Guide. 1.97 function. All input cards will be maintained as Class IE devices. DCM S-52 and the Q-list have been revised.

Conclusion:
This change eliminates the confusing Instrument Classifications for ERFDS/SPDS MUX'S 1 THRU 10 input cards but retains the IE Electrical Classification. PIMS and Q-LIST will only list the IE Electrical Classification. Instrument Classification in PIMS will be blank as the ERFDS/SPDS do not have a Reg. Guide. 1.97 function. This activity does not result in an unreviewed safety question.

99-085 ECCS Pump Suction Cross-Tie Header High Point Vent

Reference Document No.: DCP N-050486
Rev. No: 0
Reference Document Title: ECCS Pump Suction Cross-Tie Header High Point Vent

Safety Evaluation Description:
This modification will install a new ¾" vent (as close as possible to the high point of the header) with the vent line routed to the MEDT, via the closed drains system, with adequate valves and provision to visually verify adequate venting of the ECCS pump suction cross-tie piping on the charging pump side of MOV's 8807 A/B. The new vent line is designed to facilitate immediate venting and provisions to add gas void monitoring capability in the future, if required.

Safety Evaluation Summary:
DCP N-050486 allows the implementation of a new high point vent on the ECCS pump suction cross-tie header. This vent maybe used during all plant operating modes to verify that the ECCS pump suction piping does not contain any accumulated voids.

Breaching the Design Class I pressure boundary during the venting process in Operating Modes 1 through 4 has been evaluated and a licensing position has been provided in Attachment 5 of DCP N-050486.

This modification does not affect the operating license (including ITS and their bases), licensing bases or the design, function, or method of performing the function(s) as described in the SAR, including text, tables and figures. The vent valves are normally closed to maintain the pressure boundary integrity of the Design Class I portion of the system. This cross-tie header is only in operation during the recirculation phase after a LOCA, with the RHR pumps taking suction from the containment recirculation sump and feeding the CCPs and/or the SIPs.

The revision to the ISI boundary drawings and the Piping Schematics are part of the FSAR figures. However, this is acceptable since the changes relate to only the non-licensing basis feature of the SAR. The design, installation and operation of this vent, in accordance with the applicable design classification of the ECCS pump suction, meet the SAR description of the system and its function.

Conclusion:
The monthly venting of ECCS suction piping does not conflict with the current Technical Specifications (TS), Improved Technical Specifications (ITS) or result in a deviation from the DCPP licensing basis. The addition of a new

ECCS vent being implemented under DCP N-050486 does not result in an unreviewed safety question (USQ).

99-086 DCP J-50373, Change ERFDS/SPDS MUX 1-10 Input Cards Classification

Reference Document No.: J-50373
Rev. No: 0
Reference Document Title: Change ERFDS/SPDS MUX Input Cards Classification

Safety Evaluation Description:
The signal input cards in the ERFDS/SPDS MUX'S 1-10 have several different Classifications. A clarifying note will be added to the first page of Instrument Schematics 108031, 108032, 108033, 108034,108035, 108036 and 107931 stating that the ERFDS/SPDS input cards are electrical isolation devices and are Class IE. The ERFDS input cards are shown for convenience with the associated instruments. No changes will be made to the individual Instrument Schematics.

The Instrument Classification for ERFDS/SPDS input cards in the Q-List and PIMS can be eliminated. These devices are QA Class Q, Electrical Class IE devices and do not have a Reg. Guide. 1.97 function. The MUX 11 receives non safety-related inputs, but will be managed as Class IE equipment. All input cards were purchased Class IE and will be maintained as Class IE devices. DCM S-52 and the Q-list have been revised.

Safety Evaluation Summary:
The signal input cards in the ERFDS/SPDS MUX'S 1 THRU 10 have several different Instrument Classifications. The Q-List shows the input cards for MUX'S 1 THRU 10 Instrument Class IB and the Instrument Schematics show the MUX'S 1 THRU 10 input cards classified variously as Class IA, IB, and ID. In PIMS, the MUX'S 1 THRU 10 have various Instrument Classifications. MUX 11 receives non safety-related inputs but will be managed as Class I equipment. All MUX input cards were bought Class IE and will be maintained Class IE.

A clarifying note will be added to the first page of Instrument Schematics 108031, 108032, 108033, 108034,108035, 108036 and 107931stating that the ERFDS/SPDS input cards are Class IE Electrical isolation devices. The ERFDS input cards are shown for convenience with the associated instruments. No changes will be made to the individual Instrument Schematics. The Instrument Classification for ERFDS/SPDS input cards in the Q-List and PIMS can be eliminated. These devices are QA Class Q, Electrical Class IE devices and have no Reg. Guide 1.97 function. All input cards will be maintained as Class IE devices.

Conclusion:
This change eliminates the confusing Instrument Classifications for ERFDS/SPDS MUX'S 1 THRU 10 input cards but retains the IE Electrical Classification. PIMS and Q-LIST will only list the IE Electrical Classification. Instrument Classification in PIMS will be blank as the ERFDS/SPDS do not have a Reg. Guide 1.97 function. This activity does not result in an unreviewed safety question.

99-087 Boric Acid Heat Tracing Alarm Setpoint Change

Reference Document No.: M-049491
Rev. No: 0
Reference Document Title: Design Change Package (Boric Acid Heat Tracing System)

Safety Evaluation Description:
This DCP changes the Class II non-safety related Boric Acid System Heat Tracing Alarm Setpoints. The high alarm setpoint was changed from 125°F to 140°F, and the low alarm setpoint from 102°F to 85°F. The system operating points are 110°F for Train "A" and 100°F for Train "B" and they remain unchanged. Since the operating temperature for the system ranges from 95°F to 125°F, and because of the narrow range in the temperature alarms setpoint, numerous nuisance alarms are generated. The DCP intends to eliminate the nuisance alarms by setting new alarm setpoints that increase the alarm range beyond the system normal operating temperatures. The boric acid heat tracing system was previously downgraded from safety-related to nonsafety-related status via DCPs N-45376/46376.

Safety Evaluation Summary:
The change to the heat tracing alarm setpoint has no impact on plant operation or licensing commitments other than a change to the SAR to remove the alarm setpoint values. The heat tracing operating setpoints remain unchanged and therefore the system operation remains unchanged. The new alarm setpoint range is within the limits used to analyze the piping system and system instrumentation. The LBIE was required because the SAR is being changed. All the answers to the 10 CFR 50.59 Safety LBIE are negative. The boric acid system heat tracing was downgraded from safety related to Class II non-safety related via DCPs N-45376/46376.

The new low alarm setpoint allows both trains of heat tracing to fail before the setpoint is reached. This is acceptable because the heat tracing is a Class II non-safety related system and therefore redundancy is not required. The 85°F low alarm setpoint is still well above the 65°F limit for Tech Spec Surveillance and the 59°F 4% boric acid crystallization limit. Furthermore, even without the heat tracing, the BAST heaters are capable of maintaining the boric acid fluid temperature above the 65°F Tech. Spec. limit.

Conclusion:
The change in the boric acid heat tracing system alarm setpoint has no impact on the licensing basis of the plant and therefore, all the questions on the 10 CFR 50.59 evaluation have negative answers. The heat tracing system had been previously downgraded from safety related to non-safety related Class II when the boric acid concentration was changed from 12% by weight to 4% by weight. This was accomplished via DCPs N-45376/46376. The LBIE for those DCPs concluded that there is no licensing/design need for the boric acid heat tracing system since the 4% boric acid solution temperature can be maintained above its solubility temperature limit through the use of the BAST heaters. In addition, the NRC agreed that the solubility limit for the 4 weight percent boric acid solution is close to room temperature and therefore, the fluid temperature would approach its solubility limit slowly and allow time for operator intervention once the low temperature alarm was sounded. This activity does not result in an unreviewed safety question.

99-089 Revision to the QA Program Requirements for QA Records

Reference Document No.: N/A
Rev. No: 0
Reference Document Title: FSAR Update, Chapter 17, Section 17.17, Quality Assurance Records

Safety Evaluation Description:
The change to Chapter 17, Section 17.17 facilitates the inclusion of the process of electronic storage of QA records Into the QA Program. This represents a fundamental change in the process of archiving QA records previously described in the FSAR.

Safety Evaluation Summary:
PD AD10, "Records", Revision 2, approved on 4/29/99, incorporated provisions for the archiving of QA records utilizing electronic media. The use of electronic media was not a provision of the QA program previously approved by the NRC. A revision to FSARU Chapter 17, Section 17.17, "Quality Assurance Records," was therefore necessary.

Conclusion:
The revision of FSARU, Chapter 17, Section 17.17, does not change or have any impact on the operating license or design bases, the environmental plan, the emergency planning program or the security plan. In addition, this revision has no impact on the proposed improved (standard) technical specification or bases documents. This activity does not result in an unreviewed safety question.

99-090 Remove PASS Containment Air Sample Carts

Reference Document No.: DCP M-050465
Rev. No: 0
Reference Document Title: Remove PASS Containment Air Sample Carts
Safety Evaluation Description:
The Post Accident Sampling System (PASS) containment air sample carts were originally installed as a part of the Sentry system to automatically take three undiluted containment air samples in shielded carts with a fourth cart as backup. The carts are identified as SF-1, SF-2, SF-3, and SF-4. In 1983 a diluted grab sample loop was installed. The loop consists of removable filters for containment air particulates and radio iodines and a flask for noble gasses. The sample flask is identified as SF-5. Since a method could not be determined for analyzing the samples in the sample carts, sample flask SF-5 was identified as the preferred method of obtaining containment air samples. Since containment air sampling using SF-5 fully meets the NUREG-0737 requirements for post accident sampling, the carts will be removed to eliminate any confusion about the current post accident sampling capabilities.

This change removes the PASS containment air sample carts and abandons the solenoid valves and piping to the carts. The carts will be physically removed and discarded. The quick disconnect fittings on the Containment Air Sample Panel (CASP) will be removed and the lines will be capped. The electrical power to the solenoid valves that feed the sample carts will be de-energized and the valves and piping to the carts will be abandoned in place.

Safety Evaluation Summary:
The original design of the Sentry PASS CASP included the four sample carts with the capability to automatically take undiluted containment air samples in removable carts. Because of the inability to analyze the samples in the carts, a diluted containment air sample loop was designed and installed. This diluted containment air sample loop became the primary sample location for PASS sampling of containment air. Since the diluted containment air sample loop fully meets the containment air sampling requirements of NUREG-0737, the sample carts will be removed. This will eliminate the need to maintain the carts and develop the capability to analyze the samples.

<u>Compliance to NUREG-0737, Section II.B.3 (Only sections applicable to the containment air sampling requirements are included):</u>

(1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Compliance with this criteria has been successfully demonstrated through periodic implementation of STP G-14, "Operability Determination of Post Accident Sampling Program" using the diluted containment air sample loop.

(2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within 3-hour time frame established above, quantification of the following:

(a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gasses; iodines and cesiums, and nonvolatile isotopes);

Compliance with this criteria has been successfully demonstrated through periodic implementation of STP G-14, "Operability Determination of Post Accident Sampling Program" using the diluted containment air sample loop.

(b) hydrogen levels in the containment atmosphere;

Measurement of hydrogen concentration in containment using the PASS is no longer required (ref. Letter from the NRC dated February 5, 1998, Chron No. 234942).

(3) Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

The containment air sampling system is self contained system with no isolated auxiliary system ties (ref. DCM S-11, drawings 102011, sheet 6 and 108011, sheet 6).

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR 50) (i.e., 5 rem whole body, 75 rem extremities).

Compliance to this criteria has been demonstrated by dose calculation STA-097.

The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure.

Regulatory Guide 1.3 applies to BWR's, 1.7 addresses hydrogen monitoring which, as stated above is no longer required by the PASS. The levels corresponding to the source terms given in Regulatory Guide 1.4 were addressed in the design of the diluted sample loop (ref. IOM to Mr. G. A. Tidrick, dated October 9, 1981, rloc 02624-2895).

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

The ventilation system for the PASS keeps the sample panels at negative pressure relative to the PASS room to contain any potential leaks. The exhaust for this ventilation and the sample station are filtered through charcoal and HEPA filters.

In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:

1. Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

2. The diluted containment air sample loop was connected to the existing containment air sample line upstream of FCV-1102 which supplies sample cart SF-1. The sample system provided by Sentry Equipment

Corp. is used and has the same provisions for purging sample lines, minimizing sample loss and distortion, and returning the residues of sample collection back to the containment (ref. CAP P-3, "Sentry Post Accident Sampling System Containment Air Sampling and Analysis").

The sample collection point and configuration was designed to obtain a representative sample of the containment atmosphere following a transient or accident and to minimize potential debris clogging of the sample lines (ref. Drawings 102023, sheet 4 and 108023, sheet 4).

Provisions for minimizing sample line length and for reducing plateout in sample lines were considered in the design of the diluted sample loop (ref. DCN DC0-E-M-1104).

After analysis, the containment air samples are disposed of as low-level radioactive waste per procedure RCP RW-1, "Collection and Packaging of Low-Level Radioactive Waste."

- The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

The sampling system station contains both charcoal and HEPA filters (ref. Drawings 102023, sheet 7 and 108023, sheet 7).

Conclusion:
This change does not involve an unreviewed safety question. Removal of the sample carts has no potential to initiate or affect the initiation of an accident. This is a nonsafety-related system that has no accident mitigating function. The PASS containment air sampling is used as an aid to determine extent of core damage following a LOCA as is required by NUREG-0737. The technical specifications require sampling of containment air which is met by the diluted containment air sample loop. The removal of the carts have no affect on the technical specifications or their bases.

99-091 O-23, July 30, 1999

Reference Document No.: O-23
Rev. No: 0
Reference Document Title: Operating Instructions for Reliable Transmission Service for Diablo Canyon P. P.

Safety Evaluation Description:
O-23 is the PG&E Electric Transmission Planning and Operation instruction to control offsite power operability for the Diablo Canyon Power Plant (DCPP). O-23 is being incorporated as DCPP Plant Procedure OP J-2:VIII. O-23 is

used by: DCPP Operations, the Diablo 500 / 230 kV Switchyard Operators, PG&E's Transmission Operations Center (TOC), and the California Independent System Operator (Cal-ISO). O-23 is revised to account for the annual load growth projections in the PG&E System and the PG&E Los Padres Area. The PG&E System peak load used in the analysis is 24,083 MW. The Los Padres peak load is projected to increase by 3% over the 1998 number of 500 MW to a new peak of 515 MW. Previous studies show that the peak load occurs less than 1% of the time during the year.

The process and the detailed changes to O-23 are discussed below. In summary, the new O-23 is more restrictive than the previous issue, dated March 5, 1999. For the normal DCPP configuration (O-23 Table III), the load growth alone caused 3 new transmission line outage cases to become inoperable. With load growth in Los Padres, the 230kV System becomes more dependent on the capacitor banks installed at the Diablo 230kV Switchyard and the Mesa 115kV Switchyard. If one of the capacitor banks is unavailable (O-23 Tables IV and V) two to four additional line outage cases become inoperable. With both capacitor banks unavailable (O-23 Tables I and II) any transmission line outage causes the 230kV System to be inoperable.

The LBIE Screen concluded that an LBIE is required to change O-23 based on the discussion in the FSAR Update on administrative control of offsite power. That administrative control is O-23.

This annual update resulted in the following changes to the Operability Tables (Tables I, II, III, IV, and V) in O-23. The comparison is made with the latest approved version of O-23 dated March 5, 1999.

Table I - 230kV / 12kV Startup Transformers (SUT) 1-1 and 2-1 Load Tap Changers (LTCs) in Automatic and Capacitor Banks at Diablo and Mesa Unavailable

- Previously with all lines in service, the 230kV System was operable and all cases could be made operable, with compensatory measures.
- Now only all lines in service is operable and only with compensatory measures. DCPP is also operable if Mesa is disconnected from the 230kV System (both the Morro Bay-Mesa and the Diablo-Mesa lines are out of service).

Table II - SUT 1-1 and 2-1 LTCs in Manual and Capacitor Banks at Diablo and Mesa Unavailable

- The only change is in the limits of one voltage range for the startup transformers. (The only operable cases are all lines in

service and Mesa disconnected. Both cases require compensatory measures)

Table III - SUT 1-1 and 2-1 LTCs in Automatic and Capacitor Banks at Diablo and Mesa Available

- Previously all cases were operable, without compensatory measures.
- Now 6 of the 9 cases are operable, without compensatory measures.

Table IV - SUT 1-1 and 2-1 LTCs in Automatic and Capacitor Banks at Diablo Available and at Mesa Unavailable

- Previously all cases could be made operable, with compensatory measures.
- Now only two cases are operable - all lines in service and Mesa disconnected.

Table V - SUT 1-1 and 2-1 LTCs in Automatic and Capacitor Banks at Diablo Unavailable and at Mesa Available

- Previously 2 of 9 cases were inoperable.
- Now 4 of 9 cases are inoperable.

These changes in O-23 show the impact of no Morro Bay generation, coupled with the effects of load growth and capacitor bank unavailability.

Safety Evaluation Summary:
The annual update is performed in accordance with Electrical Engineering Group Procedure EE-13, Rev. 0 - "Procedure for Periodic Verification of the 230kV Grid Capability to Maintain Operability of the DCPP 230kV Offsite Power." The annual update complies with the FSAR Update Chapter 8.2.2 statement: "Compensatory measures and the voltages required to maintain operability are reviewed annually."

The update of O-23 is based on the results of Calculation 359-DC, Revision 2, "Determination of 230kV Grid Voltage Limits for DCPP System Operating Instructions O-23. Calc. 359-DC, Rev. 2 is a nuclear safety related computer analysis that models both the onsite and offsite electrical distribution system using the Electrical Transient Analyzer Program (ETAP) software. With the exception of Los Padres loading, the assumptions in the analysis are unchanged from previous analyses and conform to PSRC approved conditions, namely:

- 230 kV Operability is based on confirming that on a single unit trip or design basis event, the Emergency Diesel Generators (EDGs) may start, but do not load.

- To ensure the EDGs do not load, the voltage on the 4160 V vital buses must recover to greater than 93.75% within 16 seconds of actuation of the Second Level Undervoltage Relays (SLURs).
- The minimum pre-transfer 230 kV voltage at the DCPP Switchyard is based on maintaining a minimum post-transfer voltage of 207 kV at the Mesa 230kV Switchyard.
- The worst case design basis event is assumed to be a Reactor Trip - Delayed Safety Injection Signal scenario whereby the reactor trips and 30 seconds later the vital and non-vital loads transfer to the 230kV System. This scenario assumes the transfer of the standby condensate / condensate booster set.
- The Morro Bay Power Plant is assumed to be off-line, producing no generation.
- Los Padres peak load is 515MW.

Calc. 359-DC, Rev. 2 also included the impact of the design changes implemented during the 1R9 refueling outage.

Conclusion:
This annual update of O-23 results in fewer operable cases for offsite power system transmission line outage contingencies. Nevertheless, the DCPP licensing basis commitments are met and the offsite power system remains controlled and analyzed. The results of O-23 are based on a safety related calculation and the calculation conforms to the conditions of the DCPP design and licensing basis. This activity does not result in an unreviewed safety question.

99-092 Compensatory Measures for Open ASW Pump Room Doors

Reference Document No.: ECG 80.1
Rev. No: 3XPR
Reference Document Title: Doors Required for HELB, HVAC, or Flood Protection

Safety Evaluation Description:
ECG 80.1 Bases Section B.2 step 5 (ASW pump room doors) has been revised to state:
"5. Keep the affected doorway clear of obstructions that would prevent the door from being closed and dogged. This requirement is not intended to prevent use of equipment that requires cables, hoses, or other obstructions to be run through the doorway as long as those obstructions can be quickly* removed from the doorway, and personnel are designated to perform such removal in the event the door is required to be shut quickly*."

"*within 15 minutes"

The ECG has also been revised in several places to clarify that the compensatory measures for holding open an ASW pump room door can be applied to only one door.

Safety Evaluation Summary:
ECG 80.1 Bases Section B.2 lists the compensatory measures to be taken if an ASW pump room door is to be held open. Step 5 states the doorway is to be kept clear of obstructions, such as hoses, cables, etc. that would prevent the door from being closed and dogged.

This step has been revised to state that this requirement is not intended to prevent use of equipment that requires cables, hoses, or other obstructions to be run through the doorway as long as those obstructions can be quickly (i.e. within 15 minutes) removed from the doorway, and personnel are designated to perform such removal in the event the door is required to be shut quickly. This revision does not change the intent of the compensatory measures for holding open an ASW pump room door. The intent remains intact for a door watch to be stationed who will shut the door quickly when required to do so. Only cables, hoses or other obstructions that can be removed quickly from the doorway will be allowed to be used, and personnel will be designated to remove them quickly when needed.

The ECG has also been revised in several places to clarify that the compensatory measures for holding open an ASW pump room door can be applied to only one door.

Conclusion:
Since the intent of the compensatory measures remains the same and door closure could be accomplished before ASW would be threatened, this revision does not involve an unreviewed safety question.

99-093 Safety Evaluation for DCP M-049448

Reference Document No.: DCP M-049448
Rev. No: 0
Reference Document Title: CCW/SFP Cooling Systems evaluation for revised parameters

Safety Evaluation Description:
This DCP addresses the following analyses performed to validate that the CCW/SFP Cooling Systems' piping and equipment capabilities are within the newly-defined parameters:

- Containment dilation for CCW piping modes 10 and 11 (Split train operation)
- Piping/component analysis for cooling of the spent fuel pool during hot leg recirculation phase
- CCW piping analysis for a full core offload to the spent fuel pool

- CCW piping analysis for common equipment on header C
- Uprating the SFP cooling system to the pool boiling conditions.

Safety Evaluation Summary:
This DCP allows the implementation of the following change:
- Documents the changes outline above;
- Revises DCMs S-13, M-46 & M-71;
- Updates the FSAR Table 9.1-2 to show the higher temperature rating for the SFP cooling system components
- Revise the Q-List Notes to show the higher temperature rating of the SFP cooling system.

Conclusion:
Based on the LBIE, the changes listed above do not result in an unreviewed safety question (USQ). This conclusion is based on acceptable equipment and piping stress level with the system operating under the conditions outlined above.

99-095 Revision of MP M-50.7, "Reactor Head Lifting Device Handling"

Reference Document No.: MP M-50.7
Rev. No: 7
Reference Document Title: Installation and Removal of the Reactor Head Lifting Device

Safety Evaluation Description:
The Procedure is being revised to provide detailed instructions for the safe handling of the unloaded Reactor Head Lifting Device over the Reactor Vessel Exclusion Area's of Unit 1 and Unit 2 to facilitate reactor maintenance during unit outages.

Safety Evaluation Summary:
PG&E's NUREG-0612 submittal contains a commitment to provide detailed procedures for handling heavy loads over or in proximity to irradiated fuel or safe shutdown equipment. Further, PSRC review and approval of those procedures is required (commitment T03491).

Conclusion:
The proposed change does not result in any unreviewed safety question.

99-096 Change Rod Control Insertion Time Constants per Westinghouse Input

Reference Document No.: DCP J-049498
Rev. No: 0
Reference Document Title: Change Rod Control Insertion Time Constants per Westinghouse Input

Safety Evaluation Description:
This LBIE evaluates a design change to revise the Rod Control System dynamic compensation of the auctioneered Tavg signal from a lag/lead/lag value of 5/50/10 to 10/40/10 per DCP J-049498. This design change will reduce the sensitivity of the Rod Control system Tavg error signal to variations in Tavg due to hot leg streaming effects. This should eliminate the current undesirable situation in which intermittent rod steps have occurred during steady state operation with the Control Rods in automatic. The physical change consists of replacing an appropriate capacitor in Rod Control modules 1-TM-412P and 1-TM-421R which are located in the Hagan Rack 32.

Safety Evaluation Summary:
The LBIE determined that only changing the two capacitors in the automatic control circuitry does not change any failure or malfunction characteristics of the rod control system. The LBIE determined that the change in lead/lag compensation has not significantly changed the dynamic response of the automatic Tavg control during Condition I transients such as the 10% load reject. The LBIE determined that most DCPP safety analyses are not impacted at all since they do not credit the function of the automatic rod control circuitry for mitigation. In addition, detailed evaluations by Westinghouse, as documented in SECL-99-074, have determined there is no adverse impact on those events where function of the automatic Tavg control system is assumed to make the results more limiting. The LBIE determined that the lead/lag change does not impact any rod control system functions or margin of safety required per the Tech Specs.

Conclusion:
The revision of the automatic rod control lead lag compensation of the Tavg signal does not constitute an Unreviewed Safety Question.

99-099 Replace DCPP Unit 2 RVLIS and TMS

Reference Document No.: J-050434
Rev. No: 0
Reference Document Title: Replace DCPP Unit 2 RVLIS and TMS
Safety Evaluation Description:
This change removes the existing RVLIS and TMS processor chassis, displays, and related hardware from PAM-3 and PAM-4 cabinets and installs new processors and displays. This upgrade disconnects the PPC from the core exit thermocouples (CET) and from the TMS (reference junction RTD and hottest T/C data) and the ERFDS from the CETs. The PPC will be provided with the same data via a fiber-optic data link through the existing Validyne server. The ERFDS will be provided with the same data via analog output signals from the TMS. Amplifier modules in ERFDS MUX 9 and 10 are replaced with modules that are compatible with the new signal levels.

This change is being made because the existing RVLIS and TMS processors are obsolete. Most of the components typically replaced (i.e., printed circuit assemblies) are no longer commercially available. The original supplier of the system (Westinghouse) can no longer provide adequate support regarding diagnosis of system malfunctions and spare parts availability.

The RVLIS/TMS upgrade includes both hardware and software. The RVLIS hardware was purchased by PG&E. The DCP associated with this LBIE installs the hardware. The software was developed by PG&E in accordance with approved PG&E software quality assurance procedures. This LBIE includes both hardware and software, and assumes that all applicable verification and validation activities are completed satisfactorily before RVLIS and TMS are returned to service. Validation, in the form of PMT and engineering hold points specified in the DCP, assures the applications conform to their respective functional requirements.

Safety Evaluation Summary:
The Reactor Vessel Level Instrumentation System (RVLIS) and Thermocouple Monitoring System (TMS) are redundant microprocessor-based systems whose purpose is to supplement RCS pressure and temperature sensors and the subcooling margin monitor (SCMM) in the detection of inadequate core cooling. The RVLIS is a microprocessor-based system that measures RV level or relative void content of circulating reactor coolant. The inputs to the microprocessor include the level transmitter inputs, compensating inputs from the temperature measurements of the level transmitter impulse lines, compensating temperature and pressure measurements from the RCS, and status inputs from the RCPs. The system utilizes transmitters of differing ranges to cover different flow behaviors with and without RCP operation. The TMS receives in-core thermocouple signals, and provides a diverse means of assessing RCS hot leg temperature.

The RVLIS and TMS are located in the PAM-3 and PAM-4 main control room cabinets. Each system is comprised of two separate and redundant, seismically qualified microprocessor-based monitoring assemblies or trains. Each train monitors various analog (RVLIS and TMS) and digital (RVLIS only) inputs, performs calculations, and actuates alarm outputs when preset trip points are exceeded. Each system also provides an operator interface display of system parameters and status.

The new processor and display offer greater system reliability and enhanced features such as automatic self-test and diagnostics, greater flexibility, increased data availability, and ease of maintenance. The new components are seismically qualified to ensure that design basis earthquakes will not degrade system operation. The upgrade includes procurement of adequate

qualified spare parts to support the system for its anticipated life cycle. Should the spare parts supply be exhausted, the generic system design will allow use of commercially procured replacements following dedication.

This upgrade disconnects the PPC from the core exit thermocouples (CET) and from the TMS (reference junction RTD and hottest T/C data). The PPC will be provided with the same data via a fiber-optic data link through the existing Validyne server.

The fiber-optic data link provides inherent electrical and EMI/RFI isolation. Neither credible electrical failure in the Validyne server nor EMI/RFI sources near the fiber-optic cable will degrade operation of the safety-related TMS. The fiber-optic cable cannot generate EMI/RFI that would affect equipment or wiring in its vicinity.

The fiber-optic data link has two light pipes: transmit and receive. The receive light pipe is not connected to the optical modem at the TMS. Validyne software failures can not degrade TMS operation. In addition, the software is designed to not respond to any input from the data link receive circuit.

This upgrade also disconnects ERFDS from the CETs. The ERFDS will be provided with the same data via analog output signals from the TMS. Amplifier modules in ERFDS MUX 9 and 10 are replaced with modules that are compatible with the new signal levels. The TMS provides analog output signals to the ERFDS. The ERFDS amplifier modules are qualified electrical isolation devices that prevent credible failures in the non safety-related portion of the ERFDS from degrading operation of the TMS.

In the upgrade configuration, core exit thermocouple data that used to be provided to the PPC and ERFDS directly from the thermocouples now goes through TMS. Upon failure of a TMS train, PPC and ERFDS thermocouple data from the affected TMS train will be lost. The RVLIS and TMS integrated system designs were developed to minimize occurrence of common cause failures. In addition, the architecture of both systems includes features to mitigate against the unlikely event that any of the required applications halt or the system experiences a complete lockup. Following such a failure, a control room trouble alarm will alert the operators to the failure condition and the systems will automatically recover in less than five minutes (Reference respective V&V report) without operator intervention. The TMS auto-restart capabilities will support the PPC and ERFDS unavailability goal of .01 during all plant conditions (Reference DCM S-43Aand S-52).

Software changes in the PPC, Validyne server, and the ERFDS are performed in accordance their respective SQA plan and approved DCPP

procedures to ensure that the changes do not adversely affect operation of the affected systems.

Conclusion:
The RVLIS and TMS do not perform any active reactor protection or accident mitigation function but are consulted by the operators following an accident to obtain information allowing them to verify adequate reactor core cooling. Since these systems are only used for post-accident monitoring, their failure can not increase the probability of an accident or malfunction previously evaluated in the SAR, nor can such failure create the possibility of a new accident or equipment malfunction. No new failure modes are introduced because the existing and upgraded systems are both digital-based, with similar architecture and identical basic functions. Margin of safety, as discussed in any Technical Specification, is not affected because the RVLIS and TMS have no protection or actuation function.

However, RVLIS or TMS failures could delay access to information used by the operators to assess accident progress, and thus could secondarily affect accident recovery and consequences. The RVLIS and TMS architecture includes features to mitigate against the unlikely event that any of the required applications halt or the system experiences a complete lockup. Following such a failure, the system will automatically recover in less than five minutes (Reference respective V&V report) without operator intervention. A control room trouble alarm will alert the operators to the failure condition. As with the existing systems, the hardware comprising the upgraded systems is redundant and uses the same redundant ac power sources. Therefore, failure of the upgraded RVLIS or TMS cannot increase consequences of an accident or equipment malfunction, previously discussed in the DCPP Licensing Basis (SSER-31), as compared to the existing systems.

99-100 10 CFR 50.59 Safety Evaluation for DCP P-049392

Reference Document No.: A0487899
Rev. No: NA
Reference Document Title: Action Request
Safety Evaluation Description:
The referenced DCP replaces the existing back flow prevention check valves in the ASW Pump Room (Vault) drain lines with a different style valve and adds a strainer and sediment trap (standard piping Tee) upstream of the check valve. This DCP also modifies the pump room floor drains to reduce the potential for debris entering the pump room drain lines by applying a corrosion resistant coating to the cast iron sediment baskets. The new style check valve (single flapper swing check) has a configuration that is less susceptible to malfunction by debris than the existing check valve (duel flapper). The addition of a strainer upstream of the check valve reduces the

possibility of debris entering the check valve which could interfere with its capability to maintain a water tight boundary. The existing cast iron floor drain sediment baskets were susceptible to corrosion and contributed to debris entering the drain piping. Application of a corrosion resistant coating to the cast iron floor drain sediment baskets further reduces the potential for the sediment baskets to be a source of debris entering the floor drain system.

The affected documents are the ASW pump room drain system drawings shown on drawings 438086, 438087, 438088, 102017 Sht. 3 and 108017 Sht. 3. Drawings 102017 Sht. 3 (Unit 1) and 108017 Sht. 3 (Unit 2) are piping schematic drawings showing the drain line back flow prevention check valves. These drawings are FSAR Update (FSARU) Figures 03.02-17 Sht. 1 of 8 and 2 of 8 respectively.

The intended function of the check valves is to provide a water tight boundary to prevent backflow of water into the ASW pump rooms during and external flooding vent in the Intake Structure invert area. Currently, debris from the ASW pump vault floor drains cause the drain line check valves to malfunction or require excessive maintenance. The purpose of this DCP is to reduce the possibility that debris could enter the check valves by providing a strainer and sediment trap in the drain line upstream of the check valves and placing corrosion resistant coating on the pump room floor drain sediment baskets.

Safety Evaluation Summary:
The proposed activity will have no impact on plant operations or operator action. Replacement of the existing check valve with a check valve less susceptible to malfunction by debris, adding a strainer and sediment trap upstream of the check valve increases the assurance that the back flow prevention check valves maintain their capability to provide a water tight boundary in accordance with licensing commitments and requirements. The primary events that challenge the check valves are 1) Flooding of an adjacent ASW pump room as a result of a MELB inside the other ASW pump room, and 2) Flooding of either ASW pump room as the result of Intake Structure invert area flooding. None of the components added by this DCP can initiate an accident or seismic event. The effects of a MELB inside an ASW pump room is enveloped by current calculations. Piping calculations show that the drain piping, new check valve, sediment trap and clean-out flanges will remain intact following a seismic event. Therefore, the ASW pump room drain system will maintain a water tight boundary for flooding events associated with Intake Structure invert area flooding. The new components (strainer, sediment trap and clean-out flanges) are maintained under existing plant procedures to assure that the drain system remains water tight following maintenance activi6ties. Therefore, the addition of new components fot the ASW pump room drain line will not create the potential for a malfunction of a

different type.

Conclusion:
This activity does not increase the probability or consequences of accidents or malfunctions as described in the FSAR Update. This activity does not create any malfunctions of a different type than described in the FSAR Update. The cavity will not adversely impact any safety margin. Therefore, this activity will not result in an unreviewed safety question.

99-101 Containment Spray Interposing Relays (K645AX & K645BX) STP Frequency Relaxation

Reference Document No.: STP M-16HA1 (STP M-16HB1)
Rev. No: 1 (1)
Reference Document Title: Slave Relay Test for Operation of Interposing Relay for Cont. Spray Pump 2, K645AX (Pump 1, K645BX)

Safety Evaluation Description:
Background: LAR 94-11, LA-115 (Unit 1) and LA-113 (Unit 2) document changes to the slave relay surveillance test program. Changes to Technical Specifications section 3/4.3.2 "Engineering Safety Features Actuation System Instrumentation Surveillance Requirements" were made to relax the test frequency of slave relay testing from Quarterly (Q) to Refueling (R). Justification is documented in WCAP-13878, WCAP-14117, and WCAP-13900.

The frequency of some Slave Relay tests were not relaxed because they included an interposing or "Auxiliary" relay which did not meet the criteria presented in the WCAP's. These interposing relays did not meet the criteria because they were not Westinghouse AR or Potter & Brumfield MDR relays. Quarterly (Q) testing of the associated slave relays will be administratively controlled until an alternate method of testing the interposing relays is developed. Procedure commitment T35555 was created for administratively controlling the quarterly testing of the interposing relays.

Current Status: AT MM AR A0440861 (Unit 1) and A0440898 (Unit 2) replaced Containment Spray Interposing Relays (K645AX and K645BX) with Potter and Brumfield MDR relays. These relays now meet the criteria required by the WCAP's. This LBIE and associated 10 CFR 50.59 Safety Evaluation is being performed because LA-113 and LA-115 does not provide for automatic test frequency relaxation upon relay replacement. LAR 94-11 Attachment A page 9 specifically states: *"Changes to the testing frequency will be made in accordance with 10 CFR 50.59."*

Action: Perform a 10 CFR 50.59 safety evaluation for relaxing the test

frequency of interposing relays K645AX and K645BX. If no unreviewed safety questions are found, quarterly tests STP M-16HA1 and STP M-16HB1 may be rescinded. K645AX and K645BX will continue to be tested with the associated slave relays during the performance of STP M-16HA and M-16HB on a refueling interval.

Safety Evaluation Summary:
LA-115 (Unit 1) and LA-113 (Unit 2) as requested by LAR 94-11 relaxed slave relay test frequency from Quarterly (Q) to Refueling Interval (RI) but did not include the interposing relays because they did not meet the criteria evaluated in the referenced WCAP's. STP M-16HA and M-16HB test the associated slave relays on a Refueling Interval (RI) frequency. STP M-16HA1 and M-16HB1 were created to provide an alternate quarterly test for the interposing relays K645AX and K645BX in accordance with procedure commitment T35555.

AT MM AR A0450861 (Unit 1) and A0440898 (Unit 2) replaced K645AX and K645BX Agastat relays with Potter and Brumfield MDR relays. LA-115 (Unit 1) and LA-113 (Unit 2) limited frequency relaxation to Potter & Brumfield MDR or Westinghouse AR relays as evaluated by WCAP-13877, WCAP 13878 and WCAP-14117. The Agastat relays were not included as being eligible for test frequency relaxation. Replacing the Agastat relays with Potter & Brumfield MDR relays will allow test frequency relaxation of the interposing relays once a 10 CFR 50.59 safety evaluation is performed.

WCAP-13878 (page 65) discusses aging of relays. This discussion includes a description of how the relays are protected from the "damaging effects of debris and contamination, and is protected from extremes of high ambient temperature and high relative humidity by HVAC provided to the rooms in which SSPS is installed." Because the interposing relays are NOT installed in the SSPS room, the environmental conditions to which the interposing relays are exposed must be evaluated.

<u>Environmental Considerations</u>: The Containment Spray Interposing Relays K645AX and K645BX are installed in 4kV cubicles G7 and H9. The environmental conditions in the 4kV switchgear rooms are controlled by the HVAC system in the Turbine building as described in DCN S-23E. In addition, ECG 23.1 specifies temperature requirements for the 4kV switchgear rooms of ≤ 103°F. The SSPS room temperature limit is discussed in DCM S-38A and is stated as -5°C to 50°C (23-122°F). Therefore, the 4kV switchgear room temperature limits are milder than the SSPS room. The containment spray breakers are normally de-energized and there are no significant heat sources inside the 4kV breaker cubicles. It may therefore be assumed that there is no significant increase from ambient room temperature inside the

breaker cubicles. The SSPS cabinets are vented and are subject to dust and debris from within the SSPS room. However, the SSPS room is small and supplied with filtered air, thereby minimizing dust and debris. Interposing relays K645AX and K645BX are mounted inside the 4kV breaker cubicles. The breaker cubicles are unvented and are not subject to dust and debris from the 4kV switchgear rooms. The interposing relays in the 4kV cubicles are subject to no more dust and debris than that of the slave relays in the SSPS cabinets. It can be concluded that the environmental conditions to which the interposing relays are exposed is as mild as that of the slave relays mounted in SSPS.

Application: (Seismic, I&C Design Considerations, and Electrical Design Considerations) Design considerations including contact loading were evaluated and documented in AT MM AR A0450861 (Unit 1) and A0440898 (Unit 2) for the relay replacement and determined to be acceptable. There are no application issues that would cause the interposing relays to be less reliable than the slave relays.

Conclusion:
This evaluation concludes that the surveillance test frequency for interposing relays K645AX and K645BX may be relaxed from Quarterly (Q) to Refueling Interval (RI) without affecting relay or system reliability.

In addition, as stated Attachment A of LAR 94-11, *"relaxing the surveillance interval reduces the number of times safety systems must be removed from service and disabled from performing their required safety function. This results in an increase in safety system availability and a reduction in risk."*

Further, the Technical Specifications Improvement Program (TSIP) examined surveillance requirements as recommended in NUREG-1024. The results of the TSIP effort are presented in NUREG-1366 and concluded that, *"while some testing at power is essential, safety can be improved, equipment degradation decreased, and unnecessary personnel burden prevented by reducing the amount of testing at power."*

99-102 Rescind Commitment for Class 1E Flow Switches in Battery Rooms Ventilation

Reference Document No.: DCP J-49414; FHARE 132
Rev. No: 0
Reference Document Title: Battery Rooms Ventilation Alarm System
Safety Evaluation Description:
This change rescinds a commitment documented in SSER 8 regarding the classification of a flow alarm system in the ventilation system for the vital battery rooms system. The commitment is to install a Class 1E alarm system

to annunciate upon loss of battery room ventilation flow (ref. SSER 8). The purpose of the flow alarm system is to alert plant operators to a potential for hydrogen buildup to an explosive level in the battery rooms. PG&E met this commitment by purchasing and installing redundant Class I flow switches in the supply duct to the battery rooms. The switches are connected to the Class II main annunciator system. HVAC Calculation 83-46 demonstrates that natural ventilation alone is sufficient to provide the necessary dilution to preclude a buildup of hydrogen to explosive or combustible limits. The battery room ductwork is seismically qualified to Class I criteria to assure the natural ventilation flow paths are maintained during a seismic event. Following the rescission, the flow switches will be classified as non-class I. This classification is consistent with current industry and regulatory standards.

Safety Evaluation Summary:
As documented in SSER 8 and a series of submittals to the NRC, DCPP has a commitment to provide a Class 1E alarm system in the Class II ventilation system that serves all three vital battery rooms. This alarm system is intended to provide an alarm in the main control room upon loss of flow so that plant operators could take the necessary actions to prevent a buildup of hydrogen in the battery rooms before it could reach combustible/explosive limits. PG&E attempted to meet this commitment with redundant flow switches (and associated cables) that were installed as Class I devices except that they were connected to the Class II main annunciator system (reference PG&E Letter dated 7/20/79). Two major events occurred subsequent to the time the commitment was made. In 1983, an analysis (HVAC Calculation 83-46) was performed as part of the Independent Design Verification Project to demonstrate that natural ventilation alone is sufficient to eliminate the potential for hydrogen buildup to combustible or explosive limits in the battery rooms upon loss of operation of the normal Class II ventilation system. This information, along with the Class I designation of the flow switches is included in the FSAR. The second major event occurred in the 1984 - 1986 time frame when the classification of the switches was revised from Class I to Class II. It is not clear whether this was a deliberate or inadvertent change. Regardless of the intent, the switches have been maintained as Class II devices throughout most, if not all, of the operating life of the plant._While no longer meeting the original commitment, the current plant configuration meets all current regulatory and industry standards for a ventilation system to vital battery rooms. Therefore, rather than upgrading the flow switches to meet the original commitment (or something close to it since the installation cannot be made completely Class 1E without adding redundant Class I annunciators), the commitment will be rescinded and the FSAR and all associated drawings, specifications, DCMs and any other identified documentation will be revised to consistently indicate that the flow switches are non-class I.

To support this decision, FHARE 132 was generated to evaluate the adequacy of the battery rooms' ventilation system to maintain hydrogen below the explosive limit, even in the case of loss of forced ventilation, and to document the impact on the DCPP Fire Protection Program.

This LBIE is valid for FHARE 132 which evaluates the change, FSAR revisions resulting from this change, the DCP that will be generated to revise drawings to consistently show the non-class I designation of the switches, and revisions to DCMs to reflect the resulting configuration and licensing/design bases.

Conclusion:
This change does not result in a USQ. The current plant configuration meets current industry and regulatory standards. There is no past operability concern because the referenced analysis demonstrating that hydrogen buildup is not a concern in the vital battery room even upon loss of operation of the associated ventilation system was in place prior to commercial operation of the plant. There is no credible reason for loss of the ventilation system ductwork (e.g., loss of non-vital power, seismic event including Hosgri, etc.) that would adversely affect the availability of natural circulation.

99-103 New Fuel Elevator Modification

Reference Document No.: DCP N-50494
Rev. No: 0
Reference Document Title: New Fuel Elevator Modification
Safety Evaluation Description:
This design change modifies the new fuel elevator to temporarily accept spent fuel from the spent fuel bridge crane and contain it during reconstitution (repair). The purpose is to remove damaged fuel rods from the fuel assembly such that the repaired assembly can be returned to the reactor for another fuel cycle's use.

Safety Evaluation Summary:
The change provided by the modification to the new fuel elevator is to limit the movement of the spent fuel assembly within the elevator. This is accomplished by resetting the upper limit switch to stop the elevator before the assembly is raised to within 9 feet of the surface of the spent fuel pool. A mechanical stop is also designed to backup the electrical limits and prevent a spent fuel assembly from coming within 8-1/2 feet of the spent fuel pool surface.

The design features are similar to that required for the spent fuel pool bridge hoist which are specified in the design bases discussion in the FSAR.

1) The spent fuel pool bridge has provisions to avoid dropping or jamming

fuel. This is provided by an electrical and mechanical brake which the new fuel elevator is equipped with as well.

2) The fuel lifting devices are capable of supporting maximum loads under seismic conditions. This design includes the upgrade of the new fuel elevator to withstand the actual Hosgri loads as stipulated for the fuel handling system in the FSAR.

3) The hoists used to lift spent fuel assemblies have a limited maximum lift height on the spent fuel bridge to maintain minimum water shielding. The particular requirements are not specified. The spent fuel pool bridge hoist has a geared limit switch, which is backed up by a arm actuated limit switch. The failure of these items would result in the hook drum solid against the hoist drum and being mechanically bound. The new fuel elevator would engage a mechanical stop upon failure of the single geared electrical limit which is adjusted in this design to account for the minimum shielding required above a spent fuel assembly. Though there is not an active backup to the first impediment to raising an assembly too high, the passive nature of the stop provides the same assurance as that on the spent fuel pool bridge that there is a structural obstacle to such an occurrence. The added risk is damage to the hoist motor which cannot overcome the force exerted by the stop. This is an economic concern as the hoist brakes will stabilize the elevator as would the spent fuel hoist bridge.

4) There are other protective failures associated with the mechanical stop such as thermal overloads on the hoist motors which are also provided on both spent fuel pool bridge hoist and new fuel elevator hoist.

The physical safeguards are in addition to the primary method of ensuring the safe condition of the spent fuel assemblies which is the administrative control provided by the fuel handling, operating and other procedures.

Conclusion:
The fuel repair equipment and modifications to the new fuel elevator impose the same extensive design requirements for fuel assembly travel and structural integrity as is imposed on the spent fuel handling equipment. The effect is to bound any failure of this equipment in handling a spent fuel assembly to that associated with the same fuel handling accident described in Chapter 15 of the FSAR. The fuel repair work is to be performed on one assembly at a time and in conjunction with no other fuel movement within the spent fuel pool nor the containment. The shielding provided by the spent fuel pool is always maintained. The loss of elevator function will leave the assembly in a stable position. There is no generation of missiles or seismic

interaction sources which would impact any other fuel assemblies stored in the fuel handling building. The result of the design considerations and the assumption that the same rigorous limits on fuel handling procedures are imposed lead to the conclusion that the probability of the fuel handling accident remains as "very small" as described in Section 15.4.5.1 of the FSAR. Additionally, the exposure to the public from the fuel handling in the fuel handling area is not increased above what is calculated as "well within" the 10 CFR 100 limits.

As a result, no unreviewed safety question is created by the use of the modified new fuel elevator to handle a spent fuel assembly for fuel repair.

99-104 Unrated Penetrations Through Unit 1 Control Room Floor (Barrier 458)

Reference Document No.: FHARE 137
Rev. No: 0
Reference Document Title: Unrated Penetrations Through Unit 1 Barrier 458

Safety Evaluation Description:
FHARE 137 was written to evaluate the adequacy of the unrated penetration seals in the barrier separating Fire Areas/Zones CR-1/8A from 7A, CR-1/8C from 7A, and CR-1/8E from 7A to prevent the propagation of a design basis fire. FHARE 137 also evaluates the impact of the unrated penetration seals on the ability to achieve and maintain safe shutdown of the plant, and the affect on safety related equipment.

Safety Evaluation Summary:
The unrated penetrations which exists in the barriers will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configuration will not adversely affect redundant safe shutdown components or circuits and will not adversely affect the ability to achieve and maintain safe shutdown. The existence of these penetrations will also not adversely affect the operation of any safety related components or systems.

Conclusion:
Based on the lack of combustibles in the Fire Areas/ Zones, the existence of fire detection and suppression systems, and the configuration of the penetrations, the unrated penetrations will not adversely affect the ability to achieve and maintain safe shutdown conditions. Furthermore, alternate shutdown capability from the hot shutdown panel and manual actions are credited for shutting down the plant in the event of a fire in the Control Room or the Cable Spreading Room.

99-105 Rated Penetrations Through Unit 2 Control Room Floor (Barrier 458)

Reference Document No.: FHARE 140
Rev. No: 0
Reference Document Title: Rated Penetrations Through Unit 2 Control Room Floor (Barrier 458)

Safety Evaluation Description:
FHARE 140 was written to evaluate the adequacy of the unrated penetration seals in the barrier separating Fire Areas/Zones CR-1/8C from 7B, and CR-1/8D from 7B, to prevent the propagation of a design basis fire. FHARE 140 also evaluates the impact of the unrated penetration seals on the ability to achieve and maintain safe shutdown of the plant, and the affect on safety related equipment.

Safety Evaluation Summary:
The unrated penetrations which exists in the barriers will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configuration will not adversely affect redundant safe shutdown components or circuits and will not adversely affect the ability to achieve and maintain safe shutdown. The existence of these penetrations will also not adversely affect the operation of any safety related components or systems.

Conclusion:
Based on the lack of combustibles in the Fire Areas/ Zones, the existence of fire detection and suppression systems, and the configuration of the penetrations, the unrated penetrations will not adversely affect the ability to achieve and maintain safe shutdown conditions. Furthermore, alternate shutdown capability from the hot shutdown panel and manual actions are credited for shutting down the plant in the event of a fire in the Control Room or the Cable Spreading Room.

99-106 FHARE 127, Revision 0

Reference Document No.: FHARE 127
Rev. No: 0
Reference Document Title: Evaluation of Combustible Loading Values in Selected Appendix R Fire Areas

Safety Evaluation Description:
During a review of DCPP fire protection NRC supplemental safety evaluation reports (SSERs) 23 & 31 it was identified that these SSERs specified specific insitu combustible loading values in BTU/ft2 for fire areas that received an approved deviation from the requirements of 10 CFR 50, Appendix R, section III.G. This section of Appendix R requires that redundant divisions of hot shutdown equipment be separated by 20 feet with no intervening combustible and with area wide detection and suppression, or have a 3 hour fire rated

barrier, or have a 1-hour rated barrier with area wide detection and suppression between redundant divisions. NRC SSERs 23 & 31 identify fire areas where deviations from 10 CFR 50, Appendix R, section III.G requirements are acceptable. Approximately 24 cases have been identified were the combustible loading and fire duration values have increased above the values stated in SSERs 23 & 31. All 24 identified fire areas\zones contain a LOW fire duration categorization (less than a 45 minute fire duration). See FHARE 127, Attachment 1 for specific fire areas and fire loading numbers and a comparison between the current insitu configuration and the approved SSER fire loading numbers.

FHARE 127 provides the basis that the increase in combustible loadings does not affect the technical basis of the approved 10 CFR 50, Appendix R, section III.G deviations and demonstrates that there is no adverse impact on Diablo Canyon Power Plant's (DCPP's) Fire Protection Program, plant safety, or the ability to safely shut down the Plant.

Safety Evaluation Summary:
FHARE 127 re-evaluates the 10 CFR 50, Appendix R, section III.G deviations that were submitted to the NRC via the Unit 1 and Unit 2 Appendix R report on Appendix R compliance and subsequently approved by the NRC via SSERs 23 & 31. This FHARE concludes that the increase in combustible loadings does not affect the technical basis of the approved 10 CFR 50, Appendix R, section III.G deviations and demonstrates that there is no adverse impact on Diablo Canyon Power Plant's (DCPP's) Fire Protection Program, plant safety, or the ability to safely shut down the Plant.

Conclusion:
All fire areas\zones discussed in this evaluation remain in the low fire hazard category, which equates to fire duration of less than 45 minutes. The highest fire duration is in the Unit 1 laboratory area (fire zone 4-A-1) and equates to a fire duration of 39.4 minutes. Many of the fire zones\areas discussed in this evaluation contain less than a 20 minute fire duration. Based on the fact that the combustible loading in each of the subject fire zones has remained in the low fire loading category and the fire protection systems have not changed from those described in SSERs 23 & 31, the increased combustible loading does not affect the conclusions reached by the NRC in SSER 23 & 31 and the bases for granting the Appendix R deviations have not been affected.

In addition, on May 18, 1999, PG&E conducted an teleconference call with the NRC fire protection staff to discuss this evaluation. The NRC stated that combustible loading changes and increases are expected to occur at DCPP. The NRC stated that it is acceptable to change the combustible loading in the areas discussed in SSERs 23 & 31. The combustible loading changes shall be documented in an a 50.59 evaluation.

Therefore, DCPP's Fire Protection Program, plant safety and the ability to safely shutdown the plant is not adversely affected by the combustible loading increases in the subject fire zones\areas. This evaluation does not result in an unreviewed safety question.

99-107 Valve CVCS-2-HCV-142 Trim Replacement (increased from Cv of 13 to Cv of 26)

Reference Document No.: DCP N-50496
Rev. No: 0
Reference Document Title: Trim replacement for valve CVCS-2-HCV-142
Safety Evaluation Description:
Replace the trim set for Unit 2 valve CVCS-2-HCV-142 (Primary System and RCP Seal, Charging Flow Control Valve) from 3/4" (Cv of 13) to 1-1/4" (Cv of 26) trimset. This will reduce the system resistance and will add flexibility for operators in controlling pressurizer level at 120 GPM letdown flow while operating one centrifugal charging pump, as requested by AR A0423190. This design change will restore HCV-142 to its previous Cv configuration prior to 1991, as intended by Westinghouse; the previous configuration had functioned acceptably from initial plant startup to 1991.

Safety Evaluation Summary:
The change to the CVCS system valve HCV-142 trim increases the normal charging flow, therefore it enhances the system ability to perform its function. The increased charging flow affects the LTOP analyses, which considers mass injection challenges to the LTOP. The recently completed LTOP analyses (STA-063 Rev. 2) which support LA 131/133 (P/T curves through 16 EFPY) include consideration of the larger HCV-142 Cv of 26. As a result of the revised LTOP analyses, changes to administrative equipment restrictions regarding RCP operation at low temperatures, and injection flowpath limitations, have also been developed. OPS procedures OP L-1, OP L-5 and OP A-6:II will be revised to reflect these LTOP limits calculated by STA-063 Rev. 2. These needed procedure changes are being tracked under AR A0480951, which tracks actions to implement Licensing Amendment 133/131. Implementation of these procedural changes is required before the modified HCV-142 is returned to service. The impact on small break LOCA spectrum is positive, while there is no impact on uncontrolled boron dilution accident, the Fire Hazards Analysis, and Hosgri safe shutdown credits.

Conclusion:
The change in the CVCS system valve HCV-142 trim has no adverse impact on the licensing basis of the plant and therefore, all the questions on the 10 CFR 50.59 evaluation have negative answers. This not an Unreviewed Safety Question.

99-108 Replacement of Main Bank Transformers by Bragg Rigging and Crane

Reference Document No.: TP TB-9903
Rev. No: 0
Reference Document Title: Transport Procedure for Removing Main Bank and Spare Transformers and Installing Elin Main Bank Transformers

Safety Evaluation Description:
This safety evaluation is for heavy load lifting and moving of the Main Bank Unit 2 Transformers. The scope of the change includes the removal of the McGraw Edison 500/25kV transformers and installation of the Elin 500/25kV transformers by an outside contractor. The spare Unit 2 main bank transformer will be removed prior to the refueling outage and the inservice transformers will be replaced during the 2R9 refueling outage.

There are two aspects of this evaluation that require review. The first aspect for consideration is the removal of the spare transformer adjacent to "A" phase transformer. This transformer will be removed preoutage with Unit 2 at power or in Modes 1 through 4. The second is the removal of the inservice "A", "B", and "C" phase transformers during the refueling outage.

The previously approved Design Change Package E-50331 includes the LBIE for the replacement of the transformers and is not part of this evaluation.

Safety Evaluation Summary:
The replacement of these transformers by the contractor involves jacking the transformer off (onto) the pad, moving the transformer onto (off) a multiwheeled transporter and delivering them to a staging area in parking lot 1 approximately 1/2 mile away. This method has been reviewed by DCPP civil engineering. The movement of the transformers will be on plant designated roads and areas that have been designed to accommodate such loads.

The evolution of installing and removing transformers has been performed previously. The methods and contractor used have been evaluated and accepted. A "lessons Learned" has been applied to capture past experiences and improve the process. The installation of these transformers is part of DCP E-50331. The transformers will be handled and moved by expert handlers with experience in this type of load handling. They have already performed work on site during 2R8 and are familiar with DCPP controls, procedures, transformers and equipment.

NES Civil has evaluated for the temporary reinforcement of underground structures which could otherwise potentially be disturbed by the forces imposed during the movement of the transformer. Multiple axle trailers for moving the transformer will be used to spread the weight and reduce

concentrated loads over plant underground equipment. Any potentially vulnerable structures/components will be temporarily protected to preclude crushing due to the load weight. The route available to move the transformers has been evaluated and approved by Civil Engineering. There are no SISI targets along the haul route or near the transformers. Reference Load path AR A0340091 eval 19, A0367538 eval 28 and Civil calculation EQP-718 rev3.

A Qualified Electrical Worker will assure adequate clearance is maintained from all energized overhead lines to eliminate any potential for a loss of offsite power as a direct result of this move. The use of forklifts to position equipment and material will be monitored by a Qualified Electrical Worker. This action is already contained in existing plant procedures OM6.ID7 "Activities Near High Voltage Equipment".

Compensatory measures will be taken to assure adequate fire protection is maintained during changes to the transformer deluge system. The Industrial Fire Officers (IFO) has developed a "Fire Department Action Plan while moving Main and Aux. Transformers". This plan is in place to fight a transformer oil fire in the event that an unplanned release took place and the oil found an ignition source, or contain an unplanned release of oil. This plan will include compensatory measures for deluge systems not in operation, normal hydrants located along the route inside the P.A. which may have access blocked and the need for application of class-B foam in the event of a fire.

Transformer Tipping

Transformer tipping was evaluated. Tipping of the transformer or the transformer sliding off the transporter will not occur. There are multiple and redundant safety features built into the equipment and around the personnel to prevent this from occurring. The failure of either one zone of control, velocity fuse, or operator error would not cause the transformer to tip over while on the transporter.

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There will be additional controls and safety requirements placed on Bragg to ensure that the safety of the Plant and personnel are maintained.

- The procedural requirements of OM6.ID7 for work in the vicinity of energized overhead lines will be used to maintain safety during the movement of the spare transformer at power. Specifically an Electrical observer will be present while the Bragg is in the process of moving the spare transformer.
- A premove walkdown of the load path with the Bragg operator, Safety, Operations, and Civil Engineering to review the route and the procedure.

Civil Engineering has evaluated the load path. The transporter and tractor loading on the path is slightly greater than H-20 specification (This is the maximum load a normal highway is designed for). Underground utilities along the load path will be covered or plated as required to assist in distributing the weight of the load. These structures were evaluated in AR A0367538 eval 28. Previous transformer moves have been successfully performed with transformers approximately the same size and larger.

Chapter 15 15.2.7 "Loss of external load and / or Turbine Trip" was determined to be the most limiting fault as described in the SAR. This fault is considered "Condition II - Fault of Moderate Frequency". Faults in this condition at worst result in reactor shutdown with the plant being capable of returning to operation. Faults in this category are expected to happen approximately once per year. The movement of the spare transformer from its pad is in close proximity to the inservice "A" phase main bank transformer. A postulated toppling failure of the spare transformer could cause a loss of load (500kV) as described in section 15.2.7. This condition will not lead to an accident other than existing analyzed cases. In addition there are sufficient controls on the contractor and safety features incorporated in equipment and personnel to preclude the transformer from initiating a scenario as described in 15.2.7.

Conclusion:
The replacement of the existing Main Bank Unit 2 transformers does not constitute a Unreviewed Safety Question and is within the existing licensing basis documents to perform. The replacement of transformers by the drayage contractor has been thoroughly evaluated by DCPP personnel for effects to the plant, safety, and environmental impacts. The contractor and DCPP have incorporated necessary controls to perform this activity safely and within existing construction standards and codes.

99-109 Change Rod Control Insertion Time Constants per Westinghouse Input

Reference Document No.: DCP J-050498
Rev. No: 0
Reference Document Title: Change Rod Control Insertion Time Constants per Westinghouse Input

Safety Evaluation Description:
This LBIE evaluates a design change to revise the Rod Control System dynamic compensation of the auctioneered Tavg signal from a lag/lead/lag value of 5/50/10 to 10/40/10 per DCP J-050498. This design change will reduce the sensitivity of the Rod Control system Tavg error signal to variations in Tavg due to hot leg streaming effects. This should eliminate the current undesirable situation in which intermittent rod steps have occurred during steady state operation with the Control Rods in automatic. The

physical change consists of replacing an appropriate capacitor in Rod Control modules 1-TS-412P and 1-TM-412R which are located in the Hagan Rack 32.

Safety Evaluation Summary:
The LBIE determined that only changing the two capacitors in the automatic control circuitry does not change any failure or malfunction characteristics of the Rod Control system. The LBIE determined that the change in lead/lag compensation has not significantly changed the dynamic response of the automatic Tavg control during Condition I transients such as the 10% load reject. The LBIE determined that most DCPP safety analyses are not impacted at all since they do not credit the function of the automatic rod control circuitry for mitigation. In addition, detailed evaluations by Westinghouse, as documented in SECL-99-074 and the Unit 2 Cycle 10 preliminary Reload Safety Evaluation, have determined there is no adverse impact on those events where function of the automatic Tavg control system is assumed to make the results more limiting. The LBIE determined that the lead/lag change does not impact any rod control system functions or margin of safety required per the Technical Specifications.

Conclusion:
The revision of the automatic rod control lead/lag compensation of the Tavg signal does not constitute an unreviewed safety question.

99-110 FCV-95 Interposing Relays (K632AX & K632BX) STP Frequency Relaxation

Reference Document No.: STP M-16N1
Rev. No: 1
Reference Document Title: Slave Relay Test for Operation of Interposing Relay for FCV-95 (K632AX and K632BX)

Safety Evaluation Description:
Background: LAR 94-11, LA-115 (Unit 1) and LA-113 (Unit 2) document changes to the slave relay surveillance test program. Changes to Technical Specifications section 3/4.3.2 "Engineering Safety Features Actuation System Instrumentation Surveillance Requirements" were made to relax the test frequency of slave relay testing from Quarterly (Q) to Refueling (R). Justification is documented in WCAP-13878, WCAP-14117, and WCAP-13900.

The frequency of some Slave Relay tests were not relaxed because they included an interposing or "Auxiliary" relay which did not meet the criteria presented in the WCAP's. These interposing relays did not meet the criteria because they were not Westinghouse AR or Potter & Brumfield MDR relays. Quarterly (Q) testing of the associated slave relays will be administratively controlled until an alternate method of testing the interposing relays is developed.

Current Status: AT MM AR A0459042 (Unit 1) and A0440902 (Unit 2) replaced FCV-95 Interposing Relays (K632AX and K632BX) with Potter and Brumfield MDR relays. These relays now meet the criteria required by the WCAP's. This LBIE and associated 10 CFR 50.59 Safety Evaluation is being performed because LA-113 and LA-115 does not provide for automatic test frequency relaxation upon relay replacement. LAR 94-11 Attachment A page 9 specifically states: *"Changes to the testing frequency will be made in accordance with 10 CFR 50.59."*

Action: Perform a 10 CFR 50.59 safety evaluation for relaxing the test frequency of interposing relays K632AX and K632BX. If no unreviewed safety questions are found, quarterly test STP M-16N1 may be rescinded. K632AX and K632BX will continue to be tested with the associated slave relays during the performance of STP M-16N on a refueling interval.

Safety Evaluation Summary:
LA-115 (Unit 1) and LA-113 (Unit 2) as requested by LAR 94-11 relaxed slave relay test frequency from Quarterly (Q) to Refueling Interval (RI) but did not include the interposing relays because they did not meet the criteria evaluated in the referenced WCAP's. STP M-16N tests the associated slave relays on a Refueling Interval (RI) frequency. STP M-16N1 was created to provide an alternate quarterly test for the interposing relays K632AX and K632BX in accordance with procedure commitment T35555.

AT MM AR A0459042 (Unit 1) and A0440902 (Unit 2) replaced K632AX and K632BX Cutler Hammer relays with Potter and Brumfield MDR relays. LA-115 (Unit 1) and LA-113 (Unit 2) limited frequency relaxation to Potter & Brumfield MDR or Westinghouse AR relays as evaluated by WCAP-13877, WCAP 13878 and WCAP-14117. The Cutler Hammer relays were not included as being eligible for test frequency relaxation. Replacing the Cutler Hammer relays with Potter & Brumfield MDR relays will allow test frequency relaxation of the interposing relays once a 10 CFR 50.59 safety evaluation is performed.

WCAP-13878 (page 65) discusses aging of relays. This discussion includes a description of how the relays are protected from the *"damaging effects of debris and contamination, and is protected from extremes of high ambient temperature and high relative humidity by HVAC provided to the rooms in which SSPS is installed."* Because the interposing relays are NOT installed in the SSPS room, the environmental conditions to which the interposing relays are exposed must be evaluated.

Environmental Considerations: FCV-95 Interposing Relays K632AX and K632BX are installed in MCC 1G (2G) Aux. Relay Panel SPG. The

environmental conditions of the Aux. Relay Panel are controlled by the Auxiliary Building HVAC system as described in DCN S-23B. In addition, ECG 23.1 specifies the temperature limit for the 480 V Bus Area's to be ≤ 103°F. This temperature limit may not be exceeded for > 8 hours or for > 30° in order for the equipment to be operable. The SSPS room temperature limit is discussed in DCM S-38A and is stated as -5°C to 50°C (23-122°F). The 480 V Motor Control Center temperature "limits" are milder than the SSPS room. K632AX and K632BX are normally de-energized and there are no significant heat sources inside the Aux. Relay Panel. It may be assumed that there is no significant increase in temperature inside the Aux. Relay Panel from ambient room temperature in the Motor Control Center. Therefore, the FCV-95 Interposing Relays are not exposed to any greater temperatures than that of the Slave Relays.

The SSPS room is a closed room supplied with filtered air from the Control Room HVAC system. The SSPS room is designated as Housekeeping Zone 4 as defined in Housekeeping Procedure AD4.DC2. The Motor Control Center is supplied with filtered air from the Aux. Building HVAC system. The Control room HVAC and Aux. Building HVAC is similar in that the air filters and "roughing" filters. The Motor Control Center is also designated as Housekeeping Zone 4 as defined in Housekeeping Procedure AD4.DC2. The Interposing Relays and the Slave Relays are exposed to very similar environments regarding dust and debris, it is reasonable to assume that there is no negative impact to the reliability of the Interposing Relays due to dust and debris as compared to the Slave Relays.

Application: (Seismic, I&C Design Considerations, and Electrical Design Considerations) Design considerations including contact loading was evaluated and determined to be acceptable as documented in AT MM AR A0459042 (Unit 1) and A0440902 (Unit 2) for FCV-95 Interposing Relay replacement. There are no application issues that would cause the interposing relays to be less reliable than the slave relays.

Conclusion:
This evaluation concludes that the surveillance test frequency for interposing relays K632AX and K632BX may be relaxed from Quarterly (Q) to Refueling Interval (RI) without affecting relay or system reliability. This activity does not result in an unreviewed safety question.

99-111 Temporary Jumper for EDG 22 Lube Oil Heater.

Reference Document No.: A0484596
Rev. No: 0
Reference Document Title: Temporary Jumper for EDG 22 Lube Oil Heater

Safety Evaluation Description:
With plant in Modes 5/6, temporary jumper will be installed from panel MPH-23 to panel MPG-62 to energize lube oil heater of EDG 22 during outage of Vital 4kV/480V Bus G. Presently power supply to panel MPH-23 is from Vital 480V Bus H, and to panel MPG-62 (lube oil heater) is from Vital 480V Bus G. Lube oil temperature must be maintained above 90 degrees F. to meet EDG operability requirements.

Safety Evaluation Summary:
Loads powered by the auxiliary panels are considered Non-Class 1. The breakers were evaluated and determined to be of sufficient size. Jumper wires were determined to be 3 conductor # 6. The jumper will be installed only for a short duration to avoid unnecessary diesel starts to warm-up the oil. This jumper will not effect the EDG ability to start automatically and to maintain speed and voltage.

Conclusion:
All seven questions of the safety evaluation were answered "NO". This jumper will prevent unnecessary EDG starts to warm-up the lube oil during Bus G outage. In this jumper configuration the power supply to the heater will be from its EDG associated 4kV/480V bus. Thus, not dependent on availability of the other train (G). Air receivers will be fully charged thus the need for the air compressors will be minimized. All three air compressors will be powered from bus H which is associated with EDG 22. This activity does not result in an unreviewed safety question.

99-112 Install a Copy Machine in the Control Room Area, STA's Office

Reference Document No.: DCP E-049502
Rev. No: 0
Reference Document Title: Design Change Package
Safety Evaluation Description:
Install a copy machine in the Control Room area, STA's office.

Safety Evaluation Summary:
A copy machine is needed in the Control Room area, STA's office. The combustible loading tally in the Living FSAR did not account for plastics and must be changed. Fire Protection Engineering has reviewed and concurs with this change.

The Control Room Heat Load Tabulation is also updated. This has been coordinated with HVAC, Mechanical Engineering.

Conclusion:
It is acceptable to install a copy machine in the Control Room area, STA's office. This activity does not result in an unreviewed safety question.

99-113 FHARE 6, Rev. 4

Reference Document No.: FHARE 6
Rev. No: 4
Reference Document Title: Seismic Gap at Concrete Block Walls
Safety Evaluation Description:
FHARE 6 was prepared to evaluate the adequacy of nonrated structural gap seals in the 480V Switchgear (SWGR) Rooms, elevation 100 ft, in the Units 1 and 2 Auxiliary Building at DCPP. These gaps are located in fire areas 5A1, 5A2, 5A3, 5A4, 5B1, 5B2, 5B3, and 5B4. FHARE 6 is being revised to include nonrated vertical gap seals in the evaluation.

Safety Evaluation Summary:
The non-load bearing block walls that separate the subject fire areas were designed with structural gaps; both horizontal and vertical. Horizontal gaps run between the top of the wall and the ceiling slab. Vertical gaps run between two walls. The walls under consideration in this evaluation separate redundant safe shutdown components and are identified in the FSAR as being 3-hour fire rated barriers. The gaps in these walls were sealed with non-rated assemblies. The horizontal gap seal uses an angle iron covering with a fiberglass fill material. Vertical gap seals use a "plaster ground" L-Shaped covering. Both the horizontal and vertical assemblies have some small areas of incomplete coverage where, typically, an object penetrates the seal. At these points, fiberglass insulation or foam may be exposed. In a few cases, a small gap may exist. DCPP had committed to provide 3-hour fire rated assemblies for these walls with the exception of duct and damper assemblies. This evaluation will determine the adequacy of the as-installed structural gap seals, per the requirements of GL 86-10, Section 4 of Enclosure 1, "Fire Area Boundaries", to perform their design function; which is to contain the fire hazard associated with these areas.

The combustible loading in the fire areas evaluated herein have an associated fire severity of less than 45 minutes. In most cases the fire severity is under 30 minutes, with two areas at 32.5 minutes. A fire of sufficient magnitude to challenge the fire resistance capability of the barriers is not expected to occur. Therefore, the structural gap seals would not adversely affect the ability of the barriers to perform their design function.

If a fire starts in one of these areas, the automatic smoke detection system would alert the Control Room during the incipient stage of the fire. The plant fire brigade would then respond quickly and extinguish the fire before it became fully developed. Cable insulation is the predominate combustible material found in these areas. Cable insulation fires develop slowly.

For a fire in these areas, redundant safe shutdown systems are available or manual actions can be taken to operate equipment required to safely shut the plant down. Based on the above evaluation, the available fire protection features are sufficient to contain the fire hazard associated with these areas, therefore, redundant safe shutdown equipment outside these areas will remain available.

Conclusion:
Even though the gaps seals are not fire rated and do contain penetrants, the integrity of the boundary will not be adversely challenged due to the low combustible loading in the fire areas. A similar (horizontal) gap seal configuration was identified that uses steel angles and ceramic fill material that provides a 3 or 4-hour fire rating. The difference in fill material will not significantly impact the fire resistance characteristics of the seal based on the insulating properties of fiberglass and the protection from direct flame impingement provided by the steel angles. The fact that the as-installed horizontal gap seals are similar to a fire rated assembly provides additional assurance that they will withstand the effects of the hazards associated with these areas. This evaluation also demonstrates that the vertical gap seals provide ample protection to contain the hazard associated with these areas. The following factors support the conclusion that the plaster ground seals are adequate to perform their design function: 1) the plaster is 7/8" thick with sheet metal backing; 2) the seals overlap the gaps by more than 1"; 3) both sides of the gap are protected; and, 4) the combustible loading is not in the immediate proximity of the gaps such that direct flame impingement is not likely to occur. This activity does not result in an unreviewed safety question.

99-114 Test Frequency Relaxation for MOV OC Bypass Relays (CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G and CIAX-H)

Reference Document No.: STP M-16K
Rev. No: 18
Reference Document Title: Slave Relay Test K613A, K13B, and K614A (Phase A Isolation)

Safety Evaluation Description:
Background: LAR 94-11, LA-115 (Unit 1) and LA-113 (Unit 2) document changes to the slave relay surveillance test program. Changes to Technical Specifications section ¾.3.2 "Engineering Safety Features Actuation System Instrumentation Surveillance Requirements" were made to relax the test frequency of slave relay testing from Quarterly (Q) to Refueling ®. Justification is documented in WCAP-13878, WCAP-14117, and WCAP-13900.

The frequency of some Slave Relay tests were not relaxed because they included Interposing or "Auxiliary" relays which did not meet the criteria

presented in the WCAP's. These Relays did not meet the criteria because they were not Westinghouse AR or Potter and Brumfield MDR relays. Quarterly (Q) testing of the associated slave relays will be administratively controlled until an alternate method of testing the Interposing (MOV OC Bypass Relays in this case) is developed.

Current Status: AT MM AR A0463245 (Unit 1) replaced and A0472467 (Unit 2) is scheduled to replace MOV OC BYPASS Relays (CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G, and CIAX-H) with Potter and Brumfield MDR relays. These Potter and Brumfield MDR relays meet the criteria required by the WCAP's. This LBIE and associated 10 CFR 50.59 Safety Evaluation is being performed because LA-113 and LA-115 does not provide for automatic test frequency relaxation upon relay replacement. LAR 94-11 Attachment A page 9 specifically states: *"Changes to the testing frequency will be made in accordance with 10 CFR 50.59."*

Action: Perform a 10 CFR 50.59 safety evaluation for relaxing the test frequency of MOV OC Bypass Relays CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G, and CIAX-H. If no unreviewed safety questions are found, STP M-16K may be revised to extend the required frequency from Quarterly (Q) to a Refueling Interval (RI), once the relays have been replaced.

Safety Evaluation Summary:
LA-115 (Unit 1) and LA-113 (Unit 2) as requested by LAR 94-11 relaxed slave relay test frequency from Quarterly (Q) to Refueling Interval (RI) but did not include Interposing Relays because they did not meet the criteria evaluated in the referenced WCAP's. STP M-16K tests the Slave Relays (K613A, K613B, and K614A) and MOV OC Bypass Relays (CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G, and CIAX-H) on a Quarterly (Q) frequency. The Quarterly Requirement is due to Procedure Commitment T35555 for the Interposing Relays.

AT MM AR A0463245 (Unit 1) replaced and A0472467 (Unit 2) is scheduled to replace CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G, and CIAX-H Cutler Hammer relays with Potter and Brumfield MDR relays. LA-115 (Unit 1) and LA-113 (Unit 2) limited frequency relaxation to Potter & Brumfield MDR or Westinghouse AR relays as evaluated by WCAP-13877, WCAP 13878 and WCAP-14117. The Cutler Hammer relays were not included as being eligible for test frequency relaxation. Replacing the Cutler Hammer relays with Potter & Brumfield MDR relays will allow test frequency relaxation of the Interposing Relays once a 10 CFR 50.59 safety evaluation is performed.

WCAP-13878 (page 65) discusses aging of relays. This discussion includes a description of how the relays are protected from the *"damaging effects of debris and contamination, and is protected from extremes of high ambient*

temperature and high relative humidity by HVAC provided to the rooms in which SSPS is installed." Because the Interposing Relays are NOT installed in the SSPS room, the environmental conditions to which the Interposing Relays are exposed must be evaluated.

Environmental Considerations: MOV OC Bypass Interposing Relays CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G, and CIAX-H are installed in the Vital 480V MCC Aux. Relay Panels associated with each Bus. The environmental conditions of the Aux. Relay Panel are controlled by the Auxiliary Building HVAC system as described in DCN S-23B. In addition, ECG 23.1 specifies the temperature limit for the 480V Bus Area's to be $\leq$ 103°F. This temperature limit may not be exceeded for >8 hours or for >30° in order for the equipment to be operable. The SSPS room temperature limit is discussed in DCM S-38A and is stated as -5°C to 50°C (23-122°F). The 480 V Motor Control Center temperature "limits" are milder than the SSPS room. CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G, and CIAX-H are normally de-energized and there are no significant heat sources inside the Aux. Relay Panel. It may be assumed that there is no significant increase in temperature inside the Aux. Relay Panel from ambient room temperature in the Motor Control Center. Therefore, the MOV OC Bypass Relays are not exposed to any greater temperatures than that of the Slave Relays in the SSPS rooms.
The SSPS rooms are closed rooms that are supplied with filtered air from the Control Room HVAC system. The SSPS rooms are designated as Housekeeping Zone 4 as defined in Housekeeping Procedure AD4.DC2. The Motor Control Center is supplied with filtered air from the Aux. Building HVAC system. The Control room HVAC and Aux. Building HVAC is similar in that the air filters and "roughing" filters. The Motor Control Center is also designated as Housekeeping Zone 4 as defined in Housekeeping Procedure AD4.DC2. The MOV OC Bypass Relays and the Slave Relays are exposed to very similar environments regarding dust and debris, it is reasonable to assume that there is no negative impact to the reliability of the MOV OC Bypass Relays due to dust and debris as compared to the Slave Relays.

Application: (Seismic, I&C Design Considerations, and Electrical Design Considerations) Design considerations including contact loading has been evaluated and determined to be acceptable as documented in AT MM AR A0463245 (Unit 1) and A0472467 (Unit 2) for MOV OC Bypass Relay replacement. There are no application issues that would cause the MOV OC Bypass Relays to be less reliable than the slave relays.

Conclusion:
This evaluation concludes that the surveillance test frequency for MOV OC Bypass Relays CIAX-F, CIAX2-F, CIAX1-G, CIAX2-G, and CIAX-H may be relaxed from Quarterly (Q) to Refueling Interval (RI) without impacting relay or

system reliability. This activity does not result in an unreviewed safety question.

99-115 DCPP Units 1 & 2 Generic Zinc Addition Program Evaluation

Reference Document No.: Westinghouse SECL 99-044
Rev. No: 0
Reference Document Title: Diablo Canyon Power Plant Units 1&2 Zinc Addition Program

Safety Evaluation Description:
This LBIE allows for the implementation of a full cycle zinc injection program in conjunction with cycle specific Core Reload Safety Analyses.

Safety Evaluation Summary:
This LBIE is being performed by Turbine/Steam Generator Engineering to present SECL 99-044 to the PSRC to support full cycle zinc injection on DCPP Units 1 and 2. Zinc injection is planned to typically begin one month following unit return to power after a refueling outage and continue until one month prior to the next scheduled unit refueling shutdown. The maximum nominal concentration of zinc in the primary coolant for injection at DCPP will be 40 ppb. However, the planned RCS nominal zinc concentration for both units will be reduced until effects of full cycle zinc injection on the fuel are more fully understood. The injection of zinc into the RCS has previously been done on DCPP Unit 1 under LBIE No. 98-038 and DCPP Unit 2 under LBIE No. 98-117. Previously approved zinc injection program LBIEs had been for injection durations of nine months and twelve months respectively. The injection rate and maximum nominal RCS zinc concentration as described in this LBIE remains consistent with the prior zinc injection programs. Zinc injection is not described in the FSAR, therefore, per the LBIE Screen and TS3.ID2, "a test, experiment, condition or configuration that might affect safe operation of the plant but was not anticipated, described or evaluated in the SAR" requires a 10 CFR 50.59 evaluation. This LBIE addresses impacts of full cycle zinc injection on the RCS and associated safety-related support systems and components (i.e., Reactor Heat Removal, Chemical and Volume Control, and Emergency Core Cooling systems and components). Impact on the fuel due to zinc injection will be addressed on an ongoing bases via the cycle specific core reload safety analysis.

Conclusion:
Based on the results of the analyses and evaluations described within, it is concluded that the Zinc Addition Program may proceed at Diablo Canyon Power Plant Units 1 and 2 without violating any of the design bases or licensing bases described in the FSAR. Operation of DCPP with the Zinc Addition Program will not adversely affect any component or system in the RCS, or the RCS pressure boundary, and, therefore, does not represent an

unreviewed safety question in accordance with the requirements and definitions presented in 10 CFR 50.59(a)(2). However, in view of the fact that fuel design and performance may vary from cycle to cycle, the fuel rod design criteria will have to be shown to be met for subsequent cycles via the Core Reload Safety Analysis.

99-116 Allow Alternate LTOP XMTRs PT-403A/405A for Use in Mode 4

Reference Document No.: DCP J-050480
Rev. No: 0
Reference Document Title: Design Change Package
Safety Evaluation Description:
This evaluation is prepared to allow use of Alternate LTOP Transmitters PT-403A, -405A during Mode 4.

Safety Evaluation Summary:
The SAR in Section 5.2.2.3 describes the Low Temperature Over Pressure (LTOP) protection feature of the Plant. This feature is not changed. The alternate transmitters, PT-403A/405A, used to provide the necessary inputs to the LTOP circuitry are left in service during Mode 4. The LTOP feature described in the SAR does not specify the input pressure transmitters by number. It is acceptable to leave the alternate LTOP transmitters in service in Mode 4 because they meet all engineering functional requirements for this service. The transmitters are redundant and do not share a common tap. The taps are from the same RHR suction line from Loop 4. Sharing different taps from the same line is acceptable for this service. The alternate transmitters, PT-403A/405A, are fed from the same redundant power supplies as the normal transmitters, PT-403/405. The difference in location between the alternate and normal transmitters is that PT-403A/405A are located in Containment while PT-403/405 are located in the Auxiliary Building.

Conclusion:
Although PT-403A/405A are located below the flood elevation, there are no Mode 4 accidents that could result in a harsh environment that would result in the loss of these transmitters while the LTOP function is required. This activity does not result in an unreviewed safety question.

99-117 Install Vent Valve Upstream Of SI-2-8804B (RHR HX 2-2 To SIP 2-2)

Reference Document No.: DCP P-50501
Rev. No: 0
Reference Document Title: Install Vent Valve Upstream Of SI-2-8804B
Safety Evaluation Description:
Line SI-735-8 from Residual Heat Exchanger 2-2 to Safety Injection Pump 2-2, just upstream of MOV SI-2-8804B is subject to the accumulation of gas

voids during system vent/fill and plant start-up operations. If a gas void accumulates in this section of piping, it could migrate to the Safety Injection Pumps and potentially create a condition which could make the ECCS inoperable. The existing vent piping configuration at this location requires clearing SI pump 2-2 per STP M-89B to allow valve SI-2-8804B to be opened to vent the upstream piping.

DCP-P-50501 will install a new ¾" vent valve on the piping elbow upstream of valve SI-2-8804B next to the valve/elbow weld joint. This new vent location will improve the venting capabilities of the ECCS upstream of valve SI-2-8804B. The vent valve configuration consisting of a half coupling, pipe nipple, valve and terminal nipple with threaded cap or quick disconnect which will be welded to the outside of the elbow after a ¾" hole has been drilled in the elbow with a hole saw at the prescribed location.

Safety Evaluation Summary:
DCP P-50501 adds a vent valve assembly to the 8" pipe elbow immediately upstream of SI-2-8804B in the RHR Heat Exchanger line to SIP 2-2 suction. The vent assembly will be installed during 2R9 when this portion of the ECCS is not required to be operable. This vent valve may be used during all plant operating modes to verify that the ECCS pump suction piping does not contain any accumulated voids. Current TS 4.5.2.b.1 stipulates that operability of the ECCS be demonstrated at least once in 31 days by venting the ECCS pump casings and accessible discharge piping high points to verify that the piping is full of water. This is accomplished via STP M-89, ECCS System Venting and STP M-89B, Venting RHR to SI Pumps in Modes 1-4. By requiring this venting, the TS allows opening the discharge vent valves during Modes 1 through 4. The current TS does not limit how often the system must be vented. Improved TS (ITS) SR 3.5.2.3 requires verifying every 31 days that the ECCS piping is full of water. It does not specify how to perform the verification. The addition of the new vent valve does not conflict with the CTS or the ITS.

DCN 2-SM-50501 stipulates detailed precautionary measures to preclude introduction of the debris into the ECCS during the hole saw drilling process used to penetrate the ECCS during the installation of this vent valve assembly. However, there is a potential that this process could introduce a small quantity of debris particles into the ECCS. Debris of the type produced from a porta-band saw cut was previously evaluated by Westinghouse for the RTD Bypass Elimination Project (RTDBE) in Westinghouse Safety Evaluation Check List (SECL) 93-095, Revision 1, which was submitted and approved by the PSRC as part of the LBIE for DCP M-44425, Rev 2. The SECL (Attachment 2) evaluated potential debris impacts on the Fuel, Reactor Coolant Pumps, Primary System Components (i.e. RCS/Reactor), the

Chemical and Volume Control System, Residual Heat Removal System, Emergency Core Cooling System, Centrifugal Charging, Safety Injection and Containment Spray Pumps, Containment Spray System, and the Reactor Vessel Head Vent System. The SECL concludes that modification activities associated with the RTD Bypass Elimination Project and the debris introduced into systems during that modification would not adversely impact plant safety nor the following Westinghouse safety analyses:

- Radiological dose evaluations
- Mechanical equipment performance criteria
- LOCA and LOCA-related accidents including large and small break LOCA, LOCA hydraulic forces, post-LOCA long term cooling, and hot leg switchover time to prevent boron precipitation
- Steam Generator Tube Rupture
- Non-LOCA accident analyses
- Containment Response

As demonstrated by mockup testing (AR A0487099, AE06), the size and quantity of the debris particles introduced into the ECCS during this modification using a 6 tooth per inch hole saw are enveloped by the Westinghouse SECL.

Since the implementation of the RTDBE project, three other design changes were made to the ECCS that could be relevant to this modification. They are: Addition of pressure reducing orifices; Modification of the RHR sump screen to 1/8" hole size; and Replacement of the throttle valves. Since the size of the debris particle as determined by the mock up test is less than 1/8", the debris particle will not be lodged in these components. In addition, these changes evaluated the acceptability of particles of 1/8" in size to pass through the ECCS without impacting the system operability. Therefore, the conclusion of the Westinghouse SECL is not adversely affected by these three recent changes to the ECCS.

Conclusion:
The monthly venting of the ECCS suction piping at this location does not conflict with current Technical Specifications (TS), Improved Technical Specifications (ITS) or result in a deviation from DCPP licensing basis. The potential of introducing debris into the ECCS will not adversely affect the ability of the ECCS to perform its normal and accident mitigation function. The addition of a new ECCS vent being implemented under DCP-P-50501 does not result in an Unreviewed Safety Question.

99-118 Team Generator Tubesheet Cleaning Procedure

Reference Document No.: SSS-2.2.2-PGE-3
Rev. No: 10
Reference Document Title: Steam Generator Tubesheet Cleaning Procedure

Safety Evaluation Description:
This LBIE addresses 2R9 steam generator (SG) pressure pulse cleaning (PPC). Although the 2R9 scope includes PPC in Steam Generators 2-1 and 2-2 only, this Safety Evaluation is applicable to all Unit 2 SGs. PPC is a maintenance process performed as part of the refueling outage steam generator tube bundle cleaning activities. The process is designed to assist in the removal of scale and deposits from the SG tubes and the tube support plates. The cleaning is achieved via the controlled periodic discharge of high-pressure nitrogen gas into the tube bundle secondary side via quick-acting valves. This pressure pulse creates sonic waves, water movement and resultant tube motion which tends to break up and remove tube bundle deposits and scale.

This process takes place when the reactor is shut down and with the SG secondary side filled to various levels. PPC is followed by another cleaning process (e.g., Upper Bundle Hydraulic Cleaning and/or Sludge Lancing).

Safety Evaluation Summary:
The Pressure Pulse Cleaning system introduces a volume of high pressure nitrogen gas into the water-filled secondary side of the SG via quick-acting valves installed in the SG handholes. The resulting pressure pulse in the water generates sonic waves, water movement and resultant tube motion intended to help break up and otherwise prepare the scale and deposits residing on the SG tubes and on the tube support plates (TSP) for removal by other processes. The PPC process is implemented in the refilled SG following soaking of the tubes and TSP in water containing scale conditioning agents. PPC is followed by the Upper Bundle Hydraulic Cleaning and/or Sludge Lancing process(es) which remove the scale and deposits.

The PPC equipment is composed of a gas supply and delivery system, a fluid fill, recirculation and drain system, and various controls, including an operator situated in the immediate area of the SG. The gas discharge apparatus consists of two "pulsers" which are mounted in the handholes at the end of the SG tube lanes. Also included in the handhole-mounted apparatus is the suction header for the fluid recirculation system. Neither the pulsers nor the suction header require permanent modification of the handholes; no part of either component comes in contact with the SG tubes. The pulsers are designed to release the pressurized gas almost instantaneously and at a controlled volume and rate as specified in the Procedure. The SG secondary

water level is varied from approximately 6 inches above the first TSP to approximately 12 inches above the highest U-bend during pulsing in order to impart a range of agitation forces to the tubes. Pulser pressure is controlled as a function of water level, so that full pulser pressure of 800 psi is applied only when the level is at the fifth TSP and above. The effects of the PPC energy release into the SG secondary side have been evaluated by Westinghouse in detail as described in SECL-99-111. The evaluation considers ASME Code stress and fatigue usage, the potential for propagation of existing tube cracks, and thinned tubes effects. The analytic evaluation, which has been qualified against model testing, demonstrates the acceptability of structural stress and fatigue considerations under the Code. Fatigue usage was calculated for pulsing at a 30-second interval for 22 hours as specified in the Procedure. With each application of PPC adding a very small amount to the predicted lifetime usage factor, repeated applications of PPC at past and future outages are judged acceptable.

Although the analysis and testing support the conclusion that crack initiation and propagation will not occur, the probability of growing a pre-existing crack is not identically zero, and there are two (plugged) cracked tubes within the pulser region in SG 2-2. For this reason, the DCPP process will include eddy current examination of SG tubes following PPC, including 100% Rotating Pancake Coil (RPC) examination of tubes within the pulser region (as defined in the SECL). Tube indications will be evaluated in accordance with Tech Spec (TS) requirements, including approved alternate repair criteria (ARC), and plugged if so indicated.

The stress and fatigue evaluations include the effect of the locked SG tube-TSP interfaces that occur at DCPP. The deposits within the tube-TSP crevices which cause this tube-locking are in no way degraded by the PPC process.

Acceptability of Tube Support Plate response to PPC has been shown via comparison against steam line break loadings; the latter being 18 to 100 times the 0.26 psi load generated by PPC. DCPP TSP indications, by their localized nature, do not have significant influence on plate response. However, locations of previously-identified plate indications will be reinspected following PPC in order to assure there has been no propagation of these indications caused by PPC.

Westinghouse has also provided a review of the extensive industry experience with the PPC process twenty-eight times in sixteen different units since 1987. Included in this review are several reports of tube damage possibly attributed to PPC. Of the several reported incidents, one conclusively attributes PPC as the direct tube damage mechanism; but, this was in Model 54F tubes at D. C. Cook and is considered unique to that model. Another of the cited incidents,

that of Farley Unit 1 (Model 51), reports "...the possibility of PPC induced corrosion fatigue" as a contributing cause of accelerated propagation in an existing crack. At Sequoyah, the propagation of a pre-existing crack caused secondary-to-primary leakage during PPC, not an unexpected result. The DCPP post-PPC RPC examination described above has been prescribed for just such an occurrence.

The water chemistry requirements for the water used for PPC are the same as for steam generator sludge lancing. Only residual chemicals from the scale conditioning application applied prior to Pressure Pulse Cleaning may be present in the steam generator water at the time of PPC. When a scale conditioning agent is used, a separate safety evaluation on water chemistry is provided.

Use of the Pressure Pulse Cleaning system under procedure SSS-2.2.2-PGE-3, Rev 10 has been evaluated to contain sufficient requirements and precautions, including a local operator for quick response to off-normal operation, to provide for safe and effective operation of the cleaning system. Previous application of PPC during the 2R2, 1R3, 1R5 and 2R5 refueling outages has produced no undesirable results at DCPP.

Conclusion:
The use of the Pressure Pulse Cleaning system in the Diablo Canyon Unit 2 Steam generators has been evaluated using the criteria of 10 CFR 50.59(a)(2). Based on the evaluation it has been concluded that the plant components and safety systems will not be adversely affected during normal operation and accident conditions by the use of the cleaning system. As such, this situation does not represent an unreviewed safety question.

99-119 FHARE 103, Rev. 3

Reference Document No.: FHARE 103
Rev. No: 3
Reference Document Title: Fire Barrier Configurations in the Emergency Diesel Generator Rooms

Safety Evaluation Description:
FHARE 103 was revised to determine if the non-rated vertical structural gap seal separating diesel generator room (fire zone 22-C-1) and the diesel generator room corridor (fire area 22-C) will perform its safety function; which is to prevent a fire from propagating to the other area.

Safety Evaluation Summary:
A nonrated vertical gap seal (Component ID No. 2-22C-22C1-2011-54) is located between fire barriers No. 2011 and No. 2012, on EL. 85 ft, in a 3-hour rated fire barrier. This gap communicates between the Unit 2 Diesel Generator

Room corridor (fire area 22-C) and the Unit 2-3, Diesel Generator Room (fire zone 22-C-1). On the fire area 22-C side of the gap, the seal consists of a "plaster ground". The bottom 86 inches is a flat piece of pyrocrete 1.25 inches thick and 5.75 inches wide. From 86 inches up to the ceiling, the seal consists of an L-shaped plaster ground. Some degradation exists on the flat piece of pyrocrete. On the other side of the gap (in fire area 22-C-1), the gap is filled with Styrofoam and fiberglass insulation with a surface layer of grout. A 12 inch section of this configuration has the surface grout removed. FHARE 103 will determine the adequacy of: the as-installed gap seal; the fire barrier system; and, the other available fire protection features, to perform their design function; which is to contain the fire hazard associated with these areas.

NRC Generic Letter 86-10, Section 4 of Enclosure 1, "Fire Area Boundaries", states that, "in order to meet the regulation, fire area boundaries need not be completely sealed floor-to-ceiling, wall-to-wall boundaries. However, all unsealed openings should be identified and considered [in] evaluating the effectiveness of the overall fire barrier. Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating of the boundaries, licensees must perform an evaluation to assess the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area."

Fire Area 22-C contains circuits for diesel fuel oil transfer pumps 01 and 02 and circuits for day tank level control valves LCV-85, 86, 87, 88, 89, and 90. SSER-31 (Unit 2) documents the deviations for these circuits which have been accepted by the NRC. Circuits for all three Unit 2 diesels are located in this area. However, offsite power will remain available for safe shutdown. This fire area also has circuits associated with the CO^2 Suppression System manual actuation switches which may be affected by a fire in the areas. Spurious actuation of the manual switches may render the diesel generators in operable. As stated earlier, offsite power remains available for safe shutdown.

Fire zone 22-C-1, the 2-3 Diesel Generator Room, contains circuits for the following plant systems: AFW, CVCS, CCW, DFOS, EP, MS, MUS, RCS, SIS, SWS, and VACS. Redundant diesels and systems remain available outside this fire area. Also, offsite power remains available for a fire in this area.

If a fire starts in the DG corridor (22-C), which has a low fire load and a 9-minute fire duration, the automatic wet pipe sprinkler would suppress the fire and annunciate in the control room. This would provide early indication to the Operators who would alert the fire brigade to initiate manual fire fighting activities. If a fire starts in diesel generator room (22-C-1), which has a moderate fire load and a 1.5-hour fire duration, the detection system would annunciate in the control room and actuate the CO^2 suppression system. The fire brigade would also be alerted to ensure the fire is extinguished.

Conclusion:
Based on the minimal combustible loading in the Diesel Generator corridor, Fire Area 22-C, which has a 9 minute fire severity, and the available fire suppression system in the area, a design basis fire is not expected to propagate to the Diesel Generator Room 2-3. For a fire in 22-C, redundant systems remain available to safely shut the plant down. If a fire should occur in 22-C-1, which has a moderate fire load, fire severity of 1.5-hours, the fire detection and suppression system would adequately contain the hazard associated with this area until the fire brigade arrives to ensure the fire has been extinguished. Therefore, redundant diesel systems will remain available for a fire in these areas and the ability to safely shutdown the plant as analyzed in the FSAR will not be compromised. In addition, offsite power remains available for these fire scenarios, therefore, the Diesel's are not required to be available to achieve safe shutdown.

99-120 Westinghouse Fuel Repair Procedure FP-PEG-FR2 and FSAR Update

Reference Document No.: FP-PEG-FR2
Rev. No: 0
Reference Document Title: Fuel Repair for Diablo Canyon Unit # 2 (2R9)
Safety Evaluation Description:
This is a new Westinghouse procedure governing repair of damaged spent fuel assemblies found during 2R9 inspections. Specifically, this procedure addresses 1) damaged top nozzle removal and reinstallation, and 2) damaged fuel rod removal and storage, and subsequent installation of stainless steel pins in-place of the removed fuel rods. Repairs will be performed in the SFP with the damaged fuel assembly placed in the new fuel elevator.

Safety Evaluation Summary:
Top nozzle fuel repair consists of placing a spent fuel assembly in a suitable structure, in this case the new fuel elevator (NFE), to allow removal of a damaged top nozzle, pulling of the damaged fuel rod(s) and replacement of the rod(s) with a solid stainless steel pin. The pins removed are stored in a container in the spent fuel pool. DCP N-050494 was issued to allow modification of the new fuel elevator to support fuel repair. The modification consists of installing hard stops of solid aluminum bars in the wheel tracks for the NFE to physically prevent the spent fuel assembly from being raised above its allowed depth in the pool, resetting the upper limit switch to stop the NFE before it contacts the hard stops, installing an adapter plate to act as the work platform at the top of the NFE basket assembly, and inserting a spacer in the bottom of the NFE basket to raise the top nozzle and the first grid strap above the adapter plate so the fuel can be repaired. In addition the NFE was analyzed to support its seismic qualification to survive the loads imposed by a Hosgri earthquake. Three levels of barriers are used to assure that minimum required water shielding is maintained when working with the spent fuel

assembly. These are the same three barriers discussed in the SAR for fuel handling operations. The first barrier is administrative controls, including this procedure. The second barrier is the length of the tools in combination with slings and hoists such that the fuel rod or assembly is physically restrained from violating the minimum water level. The third barrier is a limit switch on the hoist to stop the hoist from locking up at the maximum upward travel location of the hoist.

Conclusion:
This procedure contains an acceptable method to perform top nozzle fuel repair on a spent fuel assembly. There is adequate industry experience and no design or licensing basis reason to not perform this procedure. The conclusion of this LBIE is that there is no unreviewed safety question involved with the performance of spent fuel repair in accordance with this procedure.

99-123 Fuel Rod Clipping/Removal Procedures

Reference Document No.: PEP R-9A and PEP R-9B
Rev. No: 0
Reference Document Title: "Installation of Fuel Rod Clips on Fuel Assemblies - General Procedure" and "Fuel Rod Clip Installation and Removal - Specific Instructions"

Safety Evaluation Description:
The clipping installation/removal procedure involves the use of equipment positioned on top of the SFP racks and it involves the application of a lateral force onto fuel rods during clip installation/ removal. After clipping, the clipped fuel assemblies will be stored temporarily in the SFP racks until they are loaded into the core. A separate DCP will address issues related to core operation and long-term storage in the SFP.

Safety Evaluation Summary:
The fuel clip installation/removal procedure limits the force applied to the fuel rods and the potential for grid strap or fuel rod damage is minimized by the use of a backup plate behind the fuel rods during clip installation. The maximum compressive and hoop stresses caused by the clip fingers on the fuel rods occur during installation when the finger displacement is at a maximum. Siemens (the clip vendor) has shown that these stresses are still below the yield stress of the clad.

The use of the equipment (approximate weight of 560 lbs) on top of the SFP racks is bounded by Calculation File No. SQE-43 for seismic events. The procedure requires the verification that no spent fuel is stored under or near the location to be used for the clipping equipment. This will prevent the blockage of coolant flow and will limit the potential for dropping objects onto spent fuel.

Also, since the clips will add stiffness to a fuel assembly, their presence will conservatively decrease the maximum assembly deflection under seismic conditions. Furthermore, the clips are designed to slightly enhance the mechanical strength of the fuel assemblies during fuel handling accidents without increasing the probability of mechanical interactions with fuel handling tools or rack cells during handling in the SFP.

Finally, the Inconel 718 alloy that the clips are made of is chemically and physically compatible with the existing components and systems in the SFP. The clip installation and removal equipment has a hydraulic system which contains less than five gallons of unborated water. Should a hydraulic hose break and the entire system volume empty into the spent fuel pool, dilution of the pool water will be insignificant.

Conclusion:
Based on consideration of the potential for fuel assembly damage during installation/removal, the seismic impact of the clipping equipment positioned on top of the SFP racks, the seismic impact of clips on the clipped fuel assemblies, the impact on fuel handling, and the chemical/physical compatibility of Inconel 718 clip material and installation/removal equipment with SFP components and systems, it is concluded that fuel rod clips can be safely installed on fuel assemblies in the spent fuel pool utilizing PEP R-9A & PEP R-9B and the clipped fuel can be stored and handled in the SFP prior to core reload in 2R9.

99-124 Unit 2 Cycle 10 Reactor Core Fuel Load and COLR

Reference Document No.: DCP N-50443
Rev. No: 0
Reference Document Title: DCPP Unit 2 Cycle 10 Reactor Core Fuel Load and Core Operating Limits Rpt.

Safety Evaluation Description:
This DCP incorporates the new fuel loading pattern for Cycle 10 into the plant design. In addition to changes to the fuel assemblies, a new pair of secondary sources will be installed in this core to replace the current secondary sources, which are the original set. To resolve observed baffle jetting damage, anti-baffle jet clips will be installed on 8 fuel assemblies. The impact of the clips on the fuel is evaluated herein.

Safety Evaluation Summary:
The RSE identified limits on Cycle 10 Operation in concluding that there were no unreviewed safety questions resulting from the Unit 2, Cycle 10 core design. These bases are:

1. Cycle 9 Shutdown is at a burnup between 21,300 MWD/MTU and 22,750 MWD/MTU.

2. Cycle 10 Burnup is limited to a maximum burnup of 1,000 MWD/MTU beyond end-of-full-power capability.

In addition, during coastdown:

1. The Cycle 10 extended operation is achieved by reducing the 100% power Tavg to a temperature no lower than 565°F and then proceeding to a power reduction coastdown.
2. The Lo-Lo Tavg set point and the Tavg program for rod control and steam dump systems are not changed.

The power reduction coastdown will follow a reduced Tavg program and is parallel to the original Tavg program with a minimum Tavg of 547°F. The RCCA's will be placed in the manual mode during the coastdown and the NIS power will be adjusted daily based on calorimetric power.

Conclusion:
The evaluation performed in the Westinghouse RSE for Cycle 10 and the additional evaluations performed as a part of this design change ensure that the Nuclear Fuel for Cycle 10 is designed in accordance with the proper licensing and design bases and that no impact on nuclear safety results from its implementation. The changes introduced by the addition of the two secondary sources and the presence of the anti-baffle jet clips on the fuel are accounted for in the RSE. This activity does not result in an unreviewed safety question.

99-125 Nuclear Fuel Anti-Baffle Jetting Clips

Reference Document No.: DCP N-46066
Rev. No: 1
Reference Document Title: Installation of Anti-Baffle Jet Fuel Clips
Safety Evaluation Description:
This change to install anti-baffle jet fuel clips on specific fuel elements to reduce the potential for fuel rod failure due to baffle jetting. Baffle jets are jets of water that enter the fuel rod area from the down flow area outside the baffle through the baffle plate joints. In some cases, these jets impinge on the fuel rods adjacent to the baffle plates, causing significant rod vibration. This vibration may result in high cycle fatigue failures or rod fretting wear due to the rods impacting either each other or the baffle plates. Each clip is attached to 6 rods positioned midway between the gridstraps or IFM's along the length of the fuel assembly, to reduce the rod vibration amplitude. The decision to install clips, and which locations to install them, will be made by Reactor Engineering after core off load inspection. The physical installation of the clips is performed by PEP R-9A and PEP R-9B.

Safety Evaluation Summary:
This design installs anti-baffle jetting clips, one per span, at the midspan between consecutive grids and/or IFM's, along the length of the fuel assemblies exposed to baffle jetting. The purpose of the fuel clips is to locally stiffen the fuel rods, thereby increasing the natural frequency of each individual rod under the influence of baffle jetting. By providing additional support points and forcing several rods to vibrate together, damping of the system is increased considerably. The threshold or minimum momentum of the water jet required to induce unstable, large amplitude vibrations thus increased beyond the range of conditions that can exist in the reactor.
This design change provides the bases for incorporating clips on up to 28 faces of 24 assemblies subject to jetting. The exact number of modified assemblies will be determined by the presence of jetting symptoms upon fuel inspection. The licensing bases impact of the resultant configuration will be evaluated within each fuel cycle's core design as part of the Reload Safety Evaluation process.

The clips are installed on the outside edges of relatively low power peripheral assemblies where perturbations in neutron flux are minimal. The low leakage core further reduces the power in these assemblies. The material of the clips, Inconel 718 was chosen for its structural strength not on its neutronic efficiency. Thus, along with the minimal core volume taken up by the clips, the effect will be to decrease neutron leakage very slightly.

Conclusion:
The design of the clips has been evaluated to be compatible with the fuel for a cycle of operation. The increase in reliability of the fuel assemblies enhances core operation and refueling activities. The maintenance of cladding integrity reduces potential for loose parts and spread of contamination in the reactor coolant and connected systems. The clips are incorporated into the core design in accordance with approved methodologies and do not cause the nuclear fuel to operate outside any safety limits. The design of the clip installation platform does not create any safety impacts on spent fuel pool. Thus, no new accidents or safety related equipment malfunctions arise from this design nor are any evaluated accidents related to fuel integrity or fuel handling more likely to occur or have greater consequences. This activity does not result in an unreviewed safety question.

99-126 Sensitivity Testing of Inaccessible Detectors

Reference Document No.: FHARE 116 and AD13.DC6
Rev. No: 1
Reference Document Title: Testing and Maintenance of Ionization Smoke Detectors, and Fire Protection System Performance Based Surveillance Program

Safety Evaluation Description:
FHARE 116 and AD13.DC6 are being revised to provide additional clarification for the sensitivity testing frequency of inaccessible detectors, such as detectors in containment and over main steam lines, that are inaccessible for ALARA or safety reasons. Both the FHARE and the procedure do not explicitly state that areas inaccessible during plant operations will only be tested during outages. The original Fire Protection TS Surveillance Requirement 4.3.3.8.1, which was replaced by ECG 18.3, "Fire Detection Instrumentation", states that "detectors which are not accessible during plant operation shall be demonstrated operable... during each cold shutdown...". This revision to FHARE 116 and AD13.DC6 implements this exception for the sensitivity testing of smoke detectors.

Safety Evaluation Summary:
FHARE 116 and AD13.DC6 are being revised to provide additional clarification for the sensitivity testing frequency of inaccessible detectors, such as detectors in containment and over main steam lines, that are inaccessible for ALARA or safety reasons. The NFPA 72 code requires that replaced detectors be tested within one year, then in alternate years thereafter for a minimum of three total tests. Both the FHARE and the procedure do not explicitly state that areas inaccessible during plant operations will only be tested during outages. The original Fire Protection TS Surveillance Requirement 4.3.3.8.1, which was replaced by ECG 18.3, "Fire Detection Instrumentation", states that "detectors which are not accessible during plant operation shall be demonstrated operable... during each cold shutdown...". This revision to FHARE 116 and AD13.DC6 implements this exception for the sensitivity testing of smoke detectors. This revision is consistent with the intent of the original TS wording for functional testing of smoke detectors contained in 4.3.3.8.1. License Amendments 75 and 74 moved the fire protection detector testing requirements from TS to ECG 18.3. TS 4.3.3.8.1 provides an exception for testing detectors, inaccessible during plant operation, to allow testing at the next cold shutdown of 24 hours duration. This revision is consistent with that provision.

Conclusion:
This change to the Fire Protection Program does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. This change to the Program just explicitly states the licensing position originally accepted by the NRC and DCPP in Technical Specifications (TS) for detector testing, also includes sensitivity testing, even though sensitivity testing was not mentioned as required testing in the original program. This constitutes an enhancement to the Fire Protection Program. This activity does not result in an unreviewed safety question.

99-127 Revision of LTOP Administrative RCS Temperature Restrictions

Reference Document No.: OP L-1, L-5,L-6
Rev. No: 52, 47 ,32
Reference Document Title: Operating Procedures
Safety Evaluation Description:
The existing LTOP administrative RCS temperature restrictions as delineated in the subject Operating Procedures are being revised to implement the new reactor vessel P/T heatup/cooldown curves for the burnup interval to 16 EFPY. The new curves and the ASME Code Case 514 have been approved in the NRC License Amendment 133/131. The revised LTOP temperature restrictions have been documented in design calculation STA-063 Rev. 2.

Safety Evaluation Summary:
This LBIE evaluates the revision to the Operating procedures to incorporate the new LTOP administrative temperature controls associated with the License Amendment 133/131 which extends the Appendix G reactor vessel heatup/cooldown requirements to 16 EFPY. Similar previous procedure revisions have been completed without requiring a LBIE. However, certain text within the latest NRC SER /LAR could be construed to impose more restrictive requirements than are currently administratively implemented. This includes a 107°F value for establishing the RCS vent, and the ability to administratively control the PDP in the LTOP range. Therefore, this evaluation focuses on establishing that the applicable SER/LAR text represents characteristic descriptions of the administrative limits, and does not constitute part of the technical basis for establishing NRC acceptance of the LTOP license amendment. This is accomplished by defining the LTOP margin to safety as established in the Tech Specs and associated SER/LARs and demonstrating that revising the plant administrative temperature controls does not impact this margin of safety.

1. LTOP Margin of Safety

The LTOP margin of safety is established by the NRC specified acceptance criteria for mitigating an LTOP event, the Tech Spec LCO and associated setpoints which are specified for the LTOP system, and additional requirements or exemptions which have been established in the applicable NRC SER/LARs for LTOP.

1.A LTOP Acceptance Criteria

The acceptance criterion for LTOP event mitigation is stated in the licensing basis for the LTOP system design per the most recent License Amendment 133/131 to the Technical Specifications (TS) 3/4.9.3 "Overpressure Protection Systems" which now state "

The OPERABILITY of both Class I PORVs and an RCS vent opening of at least 2.07 square inches ensures that the RCS will be protected from pressure transients that could exceed 110% of the limits of Appendix G to 10 CFR Part 50 when operating at low temperatures."

This is consistent with the wording of ITS B 3.4.12 "LTOP System". - which states, *"The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1)."*

This acceptance criteria is delineated within the Tech Specs based on the requirement that the peak pressure for any cold over pressure event must remain below the maximum allowable pressure limit corresponding to the applicable RCS temperature as shown in the 0 °F /hr cooldown curve in Figure 3.4-3.

1.B LTOP Tech Spec setpoints

The LTOP margin of safety also includes the following Tech Spec setpoints and requirements for the LTOP system:

1) A pressurizer PORV opening setpoint of 435 psig (LCO ¾.9.3.a)
2) An LTOP enable temperature of 270 °F (LCO ¾.9.3. APPLICABILITY)
3) A minimum RCS vent size of 2.07 square inches (LCO ¾.9.3.b).

It should be noted that the ITS relocates these requirements from the Tech Specs to the licensee controlled Pressure Temperature Limit Report (PTLR). However, since the PTLR is not fully implemented, it is conservative to still consider these current TS ¾ values as still part of the LTOP margin of safety.

Conclusion:

Since all of the revised LTOP administrative temperature limits still ensure that the latest LTOP analysis remains bounding, there is no impact on any NRC basis for acceptance in the SER and there is no reduction in the margin of safety as defined in the Tech Specs. This activity does not result in an unreviewed safety question.

99-128 Unit 2 Cycle 10 Fuel Assembly Loose Parts

Reference Document No.: Westinghouse SECL 99-127
Rev. No: 0
Reference Document Title: Fuel Assembly Loose Parts
Safety Evaluation Description:

This Westinghouse SECL 99-127 provides the analysis and justification for continued operation in Unit 2 Cycle 10 with pieces of damaged fuel

assemblies not retrieved from the RCS and Reactor Vessel during 2R9 outage.

Safety Evaluation Summary:
The loose parts evaluation in the Westinghouse SECL 99-127 looks at the missing end plugs, fuel cladding and grid strap pieces and evaluates the presence of these objects in the RCS and Reactor Vessel against material compatibility, RCS components, Fuel, Reactor Vessel and internals, CRDMs, SGs, RCPs and RCS piping. The loose parts are not unlike loose debris that has been lost in the Unit 2 RCS in past cycles, and reference is made to SECLs previously generated, in particular, for missing fuel assembly grid strap pieces and a hex head bolt with associated nut and washer.

Conclusion:
The evaluation performed by Westinghouse in SECL 99-127 concluded that operation in Unit 2 Cycle 10 with the missing loose debris form damaged fuel in Cycle 9 will not impact plant safety and does not represent an unreviewed safety question in accordance with 10 CFR 50.59. There exists the possibility, however, that some debris may get trapped in fuel assemblies and subsequent vibration may result in fuel rod fretting failures during Cycle 10. Such failures would be detected by routine primary coolant activity monitoring.

99-129 Replace Feedwater Pump Speed Control System, Unit 2

Reference Document No.: J-050419
Rev. No: 1
Reference Document Title: Replace Feedwater Pump Speed Control System, Unit 2

Safety Evaluation Description:
This change replaces the existing hybrid analog/digital main feedwater control systems with a high-integrity, triple-redundant system that is almost exclusively digital. In the existing system, control algorithms are executed in analog modules, while monitoring for off-normal conditions is performed in digital subsystems. The upgrade modifies the servos to add new servo pilot positioners and feedback linkages, and installs a separate control oil supply skid for each pump to supply clean oil to the new servo pilot positioner. The upgrade includes linear voltage differential transformers (LVDT) to indicate governor valve position, a third speed probe, and an interface to the plant process computer. New alarms are added to the plant main annunciator system to indicate various abnormal system conditions.

Safety Evaluation Summary:
Critical electronic components in the MFPCS are obsolete and no longer supported by the manufacturer. If the system is not replaced, eventually a failure will occur within the feedwater pump controls that cannot be repaired.

Such a failure would most likely cause lost generation or stationing a reactor operator at he pump full time for local manual control. The new system is supplied by a large U.S. national control system supplier who is expected to support the system for the reasonable future.

The control hydraulics have been a continuing source of problems and unit trips due to entrained moisture and particulates. Failures and upsets within the feedwater pump control system have resulted in significant unit transients or trips. In the existing MFPCS, control oil is derived from the MFP lubrication oil system. Ongoing hydraulic system problems result form water and particulates in the control oil system. This change installs a separated, redundant control oil supply skid for each pump to supply clean oil to the new servo pilot positiioner. The dedicated control oil system eliminates components that are susceptible to control oil contamination. However, MFP lubricating oil will still provide the motive force for the main servo pistons.

The MFPCS upgrade is fault-tolerant, mitigating the condition of the existing system, where failure of a single device can trip the unit. The new system is a proven commercial design that employs triple-redundant electronic processors. If a signal process fails, the redundant processors will allow the system to continue automatic operation. The process I/O is also triple-redundant. Failure of a single I/) module, channel, or processor will not cause loss of control.

The new system does not entirely eliminate single points of failure in the feedwater pump control system. The new servo pilot positioners and the existing servos and governor valves are not redundant. Based on field experience, the servo pilot positioners will not contribute to a failure rate greater than the existing cup valves that are replaced by the upgrade.

The upgrade includes linear voltage differential transformers (LVDT) to indicate governor valve position, a third speed probe, and an interface to the plant process computer. New alarms will be added to the plant main annunciator system to indicate various abnormal system conditions. The added instrumentation will enhance reliability and improve the operator's ability to perform feedwater system diagnostics.

Conclusion:
The DCPP main feedwater pump control system (MFPCS) is a nonsafety-related control systems that is not required to mitigate any accidents or events evaluated in the FSARU. However, malfunctions in the main feedwater pump control system can contribute to the probability of events involving supply (or lack) of main feedwater. Although the MFPCS is not safety-related, components of the MFPCS that comprise the Feedwater Isolation(FWI)

function are required to mitigate certain FSARU accidents or events. The MFPCS upgrade makes a minor modification to the trip oil system; i.e., the check valve between the trip oil and control oil systems is removed. Several new trip functions are added, but the existing mechanical trips are not modified or affected. The changes will not adversely affect any Engineered Safeguards (ESF) systems or components associated with detection or mitigation of events associated with the FWI function. The upgrade specifically addresses aspects of the existing system that contribute to lack of reliability and lost generation. The upgrade will not increase probability, frequency, or consequences of evaluated events or equipment malfunctions.

99-130 Removal of Ventilation Dampers From SSPS Room Detection Operability Requirements

Reference Document No.: ECG 18.3
Rev. No: 5
Reference Document Title: Fire Detection Instrumentation
Safety Evaluation Description:
ECG 18.3, "Fire Detection Instrumentation", surveillance requirement 18.3.4 is being revised to delete the requirement for surveillance testing of the ventilation dampers located in the Unit 1 and 2 SSPS rooms (Fire Areas 8-G & 8-H). Also, ECG Table 18.0-3, "Fire Detection Instruments Panel D", note 4, is being revised to delete the reference to associated dampers.

Safety Evaluation Summary:
ECG 18.3, surveillance requirement 18.3.4 currently requires that for SSPS room detectors connected to panel POFC, verify the system (detectors, control room alarm, panel and ventilation dampers) actuate automatically upon receipt of a simulated test signal every 18 months. This proposed change will remove ventilation dampers (FD-26, SD-26, SD-27, and FD-27) from the system description.

DCP M-49295 for Unit 1 and M-50295 for Unit 2 removed the Halon suppression system from the SSPS rooms. The ventilation damper described above were designed to electrically close upon an actuation signal from the Halon system. The dampers were required to close during a Halon system discharge in order to maintain Halon concentration inside the SSPS rooms. In addition, the ventilation dampers also function as a 3 hour rated fire dampers. This change does not affect the fire rating of the dampers.

The fire dampers will continue to close in the event of a fire in the SSPS room because the thermal links will remain in place. The fire dampers will continue to be maintained as fire rated dampers and will be continued to be tested on periodic frequency in accordance with STP M-70B, "Inspection and Testing of Fire Dampers."

Conclusion:
This change eliminates the requirements to surveillance test the ventilation dampers with detection located in the Unit 1 and 2 SSPS room. This surveillance testing was conducted to establish the operability of each SSPS room fire detection instrumentation. A simulated test signal was injected into the ventilation damper control circuit in order to determine if the damper would automatically actuate. The ability of the dampers to close from an electrical signal generated during a Halon system discharge is no longer a required function due to the removal of the Halon system.

The impact of the removal of the Halon system was evaluated in FHARE 112. The removal of the Halon system did impact the fire protection defense-in-depth design philosophy for the SSPS rooms. Nevertheless, the Halon suppression system did not protect equipment or circuits required to achieve and maintain safe shutdown for a fire in fire area 8-G (U1 SSPS room) or 8-H (U2 SSPS room). The FHARE concluded that the removal of the Halon suppression system will not affect the ability to safely shut down the plant. Since the Halon suppression system is no longer required, it can be concluded that the ventilation dampers (FD-26, SD-26, SD-27, and FD-27) are also no longer required to actuate upon a Halon system discharge signal. Therefore, the removal of the ventilation dampers from ECG 18.3 does not adversely impact the ability to safely shutdown or the DCPP Fire Protection program.

99-131 Staircases S-6 and S-7 – Fire Area Boundary

Reference Document No.: FHARE 122
Rev. No: 1
Reference Document Title: Staircases S-6 and S-7 – Fire Area Boundary
Safety Evaluation Description:
FHARE 122 Revision 1 was generated to evaluate acceptability of the steel angles covering gaps along the barriers separating Staircase S-7 from Fire Area 22-C and Staircase S-6 from Fire Zone 25.

Safety Evaluation Summary:
The extremely low fire hazards (less than 15 minute fire duration) and the available fire protection features (partial coverage wet pipe sprinkler system in fire zone 25), including hose stations and portable fire extinguishers, provide assurance that the steel angles protecting the gaps will not compromise the effectiveness of the fire barrier. The 2-hour rating of the barriers between fire zones S-6 and 25, and the 3-hour rating of the barrier separating fire zone S-7 and fire area 22-C will not be compromised by the steel angles covering the gaps. This configuration will not adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading and available fire protection features in fire zones S-6/25 and S-7/22-C, the steel angles covering gaps would not affect the ability of the fire barriers to protect against the hazards. The ability to achieve and maintain safe shutdown would not be adversely affected by the non-rated fire barrier configurations.

99-132 FHARE 133, Revision 1

Reference Document No.: FHARE 133
Rev. No: 1
Reference Document Title: FHARE 133, Revision 1
Safety Evaluation Description:
FHARE 133 Revision 1 was generated to evaluate the acceptability of all the gaps, instead of just a few specific ones, in the barriers separating the 12kV Switchgear Rooms (Fire Areas 10&20) from the Diesel Generator Corridors (Fire Areas 11D&22C) and in the barriers separating the 12kV Switchgear Rooms (Fire Areas 10&20) from the Main Transformer Yards (Fire Areas 28&29) in both Units 1 and 2.

Safety Evaluation Summary:
The gaps, which exist in the barriers of the 12kV Switchgear Rooms, will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configurations will not adversely affect redundant safe shutdown components, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in the Diesel Generator Corridors and the 12kV Switchgear Rooms, the fire detection and suppression systems, the location of redundant components, the torturous path between redundant components, the availability of offsite power, and the spatial separation in the Main Transformer Yards, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown.

99-134 Abandon the Boron Concentration Monitoring System

Reference Document No.: FSAR 7.7.1.10
Rev. No: 12
Reference Document Title: Boron Concentration Measurement System
Safety Evaluation Description:
PG&E is planning to abandon the boron concentration monitoring system (BCMS) installed on DCPP Units 1 and 2. This action will revise a PG&E commitment regarding guidance provided by NUREG-0737 and Regulatory Guide (RG) 1.97, Revision 3 for monitoring reactor coolant system (RCS) soluble boron concentration.

In a letter from PG&E to the NRC dated September 9, 1983, PG&E committed to complying with RG 1.97 including installing instrumentation to provide continuous indication of RCS soluble boron concentration. This instrumentation is the BCMS.

The BCMS has proved to be unreliable. Replacement parts are no longer available, and modifications and maintenance necessary to provide a reliable system would not be cost effective. PG&E plans to use the post accident sampling system (PASS) instead of the BCMS for measuring RCS soluble boron concentration following an accident. PASS is capable of providing a boron concentration sample promptly (within one hour) as required by NUREG-0737, Section II.B.3, "Post Accident Sampling Capability." Although the BCMS is capable of providing continuous boron concentration indication, continuous indication is not required by NUREG-0737, Section II.B.3. Also, any accident that results in isolating the letdown system will render the BCMS useless, as it is installed in the letdown system.

Removal of the BCMS from service will require a revision to the Final Safety Analysis Report (FSAR) Update. Specifically, FSAR Update Section 7.7.1.0, Table 7.7-2 and Figures 7.7-10 through 7.7-13 will be deleted.

The BCMS utilizes an AmBe neutron source. This source will be dispositioned in accordance with applicable regulations and procedures.

Safety Evaluation Summary:
PG&E is planning to remove the BCMS from service, and use the PASS to monitor soluble boron concentration during and following an accident. This is being done due to unavailability of replacement parts, excessive down time, high maintenance costs and frequent (monthly) functional checks and adjustments that are needed on the BCMS.

The PASS provides facilities to perform prompt remote sampling from several RCS sample points that are not isolated by the letdown isolation valves during an accident. The PASS facilities are designed and located such that plant personnel are able to obtain the necessary samples and analyses under accident conditions while limiting personnel radiation exposure. The PASS meets all the criteria of Item II.B.3 in NUREG-0737, including the capability to sample RCS boron concentration within one hour under accident conditions, and has been found acceptable by the NRC for post accident monitoring as discussed in SSER 14.

Conclusion:
Removal of the BCMS from service does not constitute an unreviewed safety question, and DCPP will remain within its licensing basis.

99-135 O-23, November 29, 1999

Reference Document No.: O-23
Rev. No: 0
Reference Document Title: Operating Instructions for Reliable Transmission Service for Diablo Canyon P. P.

Safety Evaluation Description:
O-23 is the PG&E Electric Transmission Planning and Operation instruction to control offsite power operability for the Diablo Canyon Power Plant (DCPP). O-23 is incorporated as DCPP Plant Procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP." O-23 is used by: DCPP Operations, the Diablo Canyon Switching Center, the Midway and Gates Switching Centers, PG&E's Transmission Operations Center (TOC), and the California Independent System Operator (Cal-ISO). O-23 is revised to account for a transmission configuration change due to the addition of the Templeton Substation. The Templeton substation will divide the existing 230 kV Transmission line #1 between Morro Bay and Gates. Morro Bay-Gates #1 will be re-named Morro Bay-Templeton, and Templeton-Gates. The Morro Bay-Gates #2 line will continue to be named the #2 line.

Both of the existing 230kV transmission lines between Morro Bay and Gates have a tap at Templeton feeding a 230/70kV transformer. In the new configuration, the tap will be removed from the Morro Bay-Gates #2 line, and the entire Templeton load will be fed from the Morro Bay-Gates #1 line. The purpose of these changes is to accommodate load growth in the Paso Robles area. The load growth for the entire Los Padres Region has been factored in the annual update of O-23, completed in July 30, 1999.

Safety Evaluation Summary:
This change to O-23 was performed in accordance with Electrical Engineering Group Procedure EE-13, Rev. 0 - "Procedure for Periodic Verification of the 230kV Grid Capability to Maintain Operability of the DCPP 230kV Offsite Power." We can also credit this change to comply with the FSAR Update Chapter 8.2.2 statement: "Compensatory measures and the voltages required to maintain operability are reviewed annually."

The change to O-23 is based on the results of Calculation 359-DC, Revision 3, "Determination of 230kV Grid Voltage Limits for DCPP System Operating Instructions O-23. Calc. 359-DC, Rev. 3 is a nuclear safety related computer analysis that models both the onsite and offsite electrical distribution system using the Electrical Transient Analyzer Program (ETAP) software. With the exception of, the Templeton voltage and impedance changes, the assumptions in the analysis are unchanged from previous analyses and conform to PSRC approved conditions, namely:

- 230 kV Operability is based on confirming that on a single unit trip or

design basis event, the Emergency Diesel Generators (EDGs) may start, but do not load.

- To ensure the EDGs do not load, the voltage on the 4160 V vital buses must recover to greater than 93.75% within 16 seconds of actuation of the Second Level Undervoltage Relays (SLURs).
- The minimum pre-transfer 230 kV voltage at the DCPP Switchyard is based on maintaining a minimum post-transfer voltage of 207 kV at the Mesa 230kV Switchyard.
- The worst case design basis event is assumed to be a Reactor Trip - Delayed Safety Injection Signal scenario whereby the reactor trips and 30 seconds later the vital and non-vital loads transfer to the 230kV System. This scenario assumes the transfer of the standby condensate / condensate booster set.
- The Morro Bay Power Plant is assumed to be off-line, producing no generation.
- Los Padres peak load is 530MW.

Conclusion:
This revision to O-23 results in more operable cases for offsite power system transmission line outage contingencies. The DCPP licensing basis commitments are met and the offsite power system remains controlled and analyzed. The results of O-23 are based on a safety related calculation and the calculation conforms to the conditions of the DCPP design and licensing basis. This activity does not result in an unreviewed safety question.

99-136 Relocation of TS ¾.11.1.4 to ECG 19.1
Reference Document No.: ECG 19.1
Rev. No: 0
Reference Document Title: Liquid Radwaste - Temporary Outdoor Tanks
Safety Evaluation Description:
This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.11.1.4 to a new ECG (ECG 19.1). Relocation of TS 3/4.11.1.4 to the ECG program was authorized by Reference 1. The TS 3/4.11.1.4 bases are also being included in the ECG. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
TS 3/4.11.1.4, "Liquid Radwaste - Temporary Outdoor Tanks," requires the quantity of radioactive material contained in any temporary outdoor tanks to be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.11.1.4 to a new ECG (ECG 19.1). Relocation of TS 3/4.11.1.4 to the ECG program was authorized by LA 135. The ECG and the ECG Bases have been written in the ITS format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Conclusion:
Relocation of TS 3/4.11.1.4 to the ECG program was authorized by LA 135. Use of ITS formatting, and the addition of licensing information for completeness, are administrative in nature and do not change the technical requirements of the relocated TS. Therefore, this change does not involve an unreviewed safety question.

99-137 FSAR Update of Emergency Diesel Generator Loading Tables 8.3-3 and 8.3-5

Reference Document No.: FSAR Update
Rev. No: 12
Reference Document Title: FSAR Update to Tables 8.3-3 and 8.3-5
Safety Evaluation Description:
Update FSAR Tables 8.3-3 and 8.3-5 to incorporate changes in assumptions in design calculations 15-DC and 125-DC for calculating the maximum electrical load on the emergency diesel generators (EDG). Secondly, perform an editorial change in Table 8.3-3, changing the Containment Fan Cooler's Rating and Maximum BHP from 100 hp to 300 hp. Finally, the fire water pumps are being removed from the tabulations as they have been a source of confusion in understanding the maximum steady state loading of the Bus 1F and 1H EDGs.

The electrical load change, increases the electrical losses in transformers and cables from 4 kW to 6 kW on each vital bus. The increase resulted from using the break horsepower full load amps in lieu of the rated horsepower full load amps. Another load change resulted from using the same break horsepower rating for all four (4) Centrifugal Charging Pumps, which increased the total loading on Bus 1G by 24 kW.

Safety Evaluation Summary:
Two new assumptions used when calculating the maximum electrical loading of the emergency diesel generators, affect Tables 8.3-3 and 8.3-5. The first assumption change was to use the full load current during maximum break horsepower (BHP) to determine the losses in the cabling between the switchgear and the load. These losses are due to the I^2R heating losses in

the cables and transformers. The previous assumption used the full load current at rated horsepower. Because the calculation was to determine worst case loading, this assumption is worst case. The second assumption change was to increase the maximum BHP for Centrifugal Charging Pump 1-2 (620 hp) to the BHP of the other three pumps (650 hp). This is a conservative assumption change, resulting in a higher kW loading on the EDG. The results of either assumption do not exceed the 2750 kW rating of the diesel generator.

Removing the fire water pumps from these tables is to clarify the true intent of the tables. These tables show the maximum steady state loading of each EDG is within its capabilities. These values represent the load conditions that occur in the early stages of an accident, with the loss of offsite power. The fire water pumps are an intermittent load and are not required to operate in conjunction with any design basis accident, therefore, it is appropriate to remove this information.

Another change to FSAR Table 8.3-3 is an editorial change in the horsepower rating for the CFCUs. The electrical load for the CFCUs in this table was based on the motors running at high speed. At high speed, the CFCU motors have a horsepower rating of 300 hp. The existing FSAR Table was found to show the low speed horsepower rating, but the high speed electrical loading. This change did not affect the generator loading, and is a simple editorial change to have the horsepower rating match the load value.

Conclusion:
The changes in design calculation 15-DC and 125-DC did not change how the EDGs function or operate. Their calculated loading during a design basis accident is based on the automatically sequenced loads. The total load remained within the capability of the EDGs. The calculations and tables addressed by this FSAR change are not associated with anticipated EDG loading during the continuous operational phase of EDG operation. The fire water pumps are intermittent loads and have been removed from the steady state load tabulations for the emergency diesel generators.

Various emergency operating procedures direct the Control Room Operators to control the EDG's loading in accordance with EOP ECA-0.3. The loading of the EDGs is within their capabilities as defined in RG 1.9. No adverse consequences result from this FSAR change and the EDGs operate within their design and licensing basis. This activity does not result in an unreviewed safety question.

99-138 Nonrated Structural Gap Seals in Fire Barriers

Reference Document No.: FHARE 134
Rev. No: 0
Reference Document Title: Nonrated Structural Gap Seals in Fire Barriers
Safety Evaluation Description:
Non-rated structural gap seals exist in walls at Diablo Canyon Power Plant (DCPP). These walls are designated as Appendix R and Appendix A fire barriers, and typically have a 3-hour fire endurance rating. The Appendix R barriers separate redundant safe shutdown (SSD) systems. DCPP is committed to: seal openings in fire barriers with rated assemblies equivalent to the fire endurance rating of the wall; or, prepare a Fire Hazard/Appendix R Evaluation (FHARE) to demonstrate that the seals will prevent a fire from spreading to the adjacent area. FHARE 134 evaluates these non-rated structural gap seals to determine if they are capable of performing this function.

Safety Evaluation Summary:
Diablo Canyon Power Plant has prepared a FHARE to demonstrate that the seals will perform their safety function. This evaluation concludes that the Fire Protection Program is not adversely affected by the presence of these non-rated structural gap seals in fire barriers. As a result of this evaluation, no new operator actions are necessary to ensure safe shutdown is achieved and maintained in the event of a fire. In all cases evaluated, the fire area/zone combustible loading is "low". This fact is an important attribute in determining the adequacy of these seal configurations. FHARE 134 concludes, that in the event of a fire, these gap seals will prevent a fire from spreading to the adjacent area.

Structural gap seals are a "passive" fire protection feature.

At DCPP, the fire protection program is based on the concept of "defense-in-depth". The goal is to prevent, contain, detect and suppress fires. This is accomplished by:

- Maintaining combustible loading levels, "as-low-as-reasonably-achievable".
- Implementing "Equipment Control Guidelines".
- Implementing procedures to control ignition sources.
- Using "passive" fire barrier systems.
- Using "active" fire detection and suppression systems.
- Providing manual fire fighting equipment throughout the plant.
- Staffing and training a "Full-time" fire brigade.
- Providing Operations with shutdown procedures in the event of a fire.

Passive fire barriers are an integral part of this defense-in-depth strategy. Passive barriers are the simplest and most reliable type of fire protection

system. Since "containing a fire" is a program goal, establishing adequate barriers is an important first-line-of-defense. To establish an adequate fire barrier, it does not need to be completely sealed floor-to-ceiling, wall-to-wall. However, all unsealed openings should be identified and considered by evaluating the effectiveness of the overall barrier. This evaluation uses the later approach, and determines if the "as-installed" gap seals will prevent a fire from spreading to the adjacent area.

This evaluation covers cases where structural gaps are sealed using either "steel angles" or "plaster enclosures", or a combination of both.

Fire tests of steel angle and plaster seal configurations are not available, therefore, a fire rating can not be assigned. However, fire tests of seals made of similar materials and configurations provide evidence that the as-installed seals will provide some level of fire protection. The materials (steel, grout, fiberglass, plaster and pyrocrete) used to construct the seals have excellent fire resistive properties. Plaster, pyrocrete, and fiberglass have low coefficients of thermal conductivity. Typically, steel, plaster or pyrocrete covers the gap. The gap is either filled with fiberglass or has "no" fill material (a dead-air-space). Both fiberglass and air are excellent insulators. The steel angle and plaster covers protect the gaps from direct flame impingement. Both fiberglass and air provide for low thermal conductivity across the gap.

The DCPP Operating License, Amendment 51, Question No. 27, states: "In Unit 1, 4.16 kV cable spreading rooms separate each division with 3-hour fire rated barriers." DCPP's response states: "The walls that separate and enclose the three 4.16kV cable spreading rooms will be upgraded to two hour rated barriers. This will be accomplished by fireproofing the exposed steel beams in the walls and ceiling of the 4.16 kV cable spreading room and fireproofing the one-inch gaps between the concrete block walls and the poured concrete walls." The bases for compliance is, the walls separating these rooms are concrete block and poured concrete which have (at least) a 2-hour rating. The exposed steel beams and gaps have fireproofing as described above. This evaluation concludes that the structural gap seals (fireproofing), which covers beams, columns, and gaps, will prevent a fire from spreading to the adjacent area.

Conclusion:
Based on: a review of fire tests, the low fire loadings, and the detection and suppression systems in the areas being evaluated, these seals will withstand the hazard associated with the area. This conclusion takes into consideration the materials used, how they react when exposed to a fire, the seal configurations, and construction techniques. These seals are located in 3-hour fire barriers. In all cases, the fire duration in the area is less than 45 minutes,

and in most cases less than 15 minutes. Therefore, these "passive" fire barrier systems are capable of containing the hazard that is present. In addition, FHARE 134 also credits available "active" fire protection features and/or the manual fire fighting equipment. These fire protection features further increase the margin of safety. Based on the results of this evaluation, the as-installed gap seals will prevent a fire from spreading to the adjacent area/zone. In the event of a fire in these areas/zones, safe shutdown conditions will be achieved and maintained.

99-139 Gaps in Appendix-A Fire Rated Boundaries

Reference Document No.: FHARE 135
Rev. No: 0
Reference Document Title: Gaps in Appendix-A Fire Rated Boundaries
Safety Evaluation Description:
FHARE 135, Revision 0 evaluated the acceptability of the unrated gap assemblies in fire barriers separating Fire Area 13-D from Fire Zone 12-E, Fire Area 14-B from Fire Zone 14-A, Fire Area 15 from Fire Zones 14-A and 14-C, and Fire Area 18 from Fire Zones 19-A and 19-B.

Safety Evaluation Summary:
The available fire protection features, including hose stations and portable fire extinguishers, provide assurance that the unrated gap assemblies will not compromise the effectiveness of the fire barrier in preventing the propagation of a fire. Also, these as-built configurations will not adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the available fire protection features in the affected areas, the unrated gap assemblies would not affect the ability of the fire barriers to protect against the hazards. The ability to achieve and maintain safe shutdown would not be adversely affected by the non-rated fire barrier configurations.

99-140 Unrated Gaps in Appendix R Barriers in the Turbine Building

Reference Document No.: FHARE 30
Rev. No: 5
Reference Document Title: FHARE 30, Revision 5
Safety Evaluation Description:
FHARE 30, Revision 5 was generated to address all the gaps in the interior barriers and the exterior barriers of the Diesel Generator Fire Areas in Units 1and 2 (TB1, TB2, TB3, TB8, TB9, and TB17). This includes gaps that are bounded on one side by the Turbine Building Skin. This revision of FHARE 30 also address unique barrier configurations located within these Fire Areas.

Safety Evaluation Summary:
The non-rated openings and penetrants which exist in the barriers separating redundant emergency diesel generators and adjacent fire areas will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configurations will not adversely affect redundant emergency diesel generators, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in the diesel intake and exhaust rooms (silencer rooms) and the rooms that bound the Diesel Generator Areas, available fire detection and CO^2 systems in the Diesel Generator Rooms, the location of redundant components, and the availability of offsite power, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown. The non-rated features will not adversely affect the ability of the fire barriers to compartmentalize a fire within the affected area.

99-141 Fire Barrier Configurations in the Emergency Diesel Generator

Reference Document No.: FHARE 103
Rev. No: 4
Reference Document Title: FHARE 103, Revision 4
Safety Evaluation Description:
FHARE 103 Revision 4 was generated to remove the fire barrier configurations in the barriers communicating from the Emergency Diesel Generator rooms and their corresponding Diesel Generator Corridors from this FHARE. These barriers are now addressed in FHARE 30.

Safety Evaluation Summary:
The non-rated openings and penetrants which exist in the barriers separating redundant emergency diesel generators will not affect the integrity of the fire barrier to compartmentalize a design basis fire. The configurations will not adversely affect redundant emergency diesel generators, and the configurations will not involve a fire that would adversely affect the ability to achieve and maintain safe shutdown.

Conclusion:
Based on the low combustible loading in the diesel intake and exhaust rooms (silencer rooms), the fire detection and suppression systems, the location of redundant components, and the availability of offsite power, a design basis fire is not expected to adversely affect the ability to achieve safe shutdown. Removal of the evaluations related to the fire area boundaries, and adding them into FHARE 30 does not affect the Fire Protection Program as approved.

99-142 Relocation of TS 3/4.1.2.1 to ECG 8.5.

Reference Document No.: ECG 8.5
Rev. No: 0
Reference Document Title: Boration Systems - Flow Path - Shutdown
Safety Evaluation Description:
This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.1.2.1 to a new ECG (ECG 8.5). The TS 3/4.1.2.1 bases are also being included in the ECG. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format.

Safety Evaluation Summary:
This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.1.2.1 to a new ECG (ECG 8.5). Relocation of TS 3/4.1.2.1 to the ECG program was authorized by LA 135. The ECG and the ECG Bases have been written in the ITS format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Conclusion:
Relocation of TS 3/4.1.2.1 to the ECG program was authorized by LA 135 Therefore, this change does not involve an unreviewed safety question.

99-143 Relocation of TS 3/4.1.2.5 to ECG 8.8.

Reference Document No.: ECG 8.8
Rev. No: 0
Reference Document Title: Borated Water Source - Shutdown
Safety Evaluation Description:
This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.1.2.5 to a new ECG (ECG 8.8). Relocation of TS 3/4.1.2.5 to the ECG program was authorized by LA 135. The TS 3/4.1.2.5 bases are also being included in the ECG. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.1.2.5 to a new ECG (ECG 8.8). Relocation of TS 3/4.1.2.5 to the ECG program was authorized by LA 135. The ECG and the ECG Bases have been written in the ITS format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the

Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Conclusion:
Relocation of TS 3/4.1.2.5 to the ECG program was authorized by LA 135. Use of ITS formatting, and the addition of licensing information for completeness, are administrative in nature and do not change the technical requirements of the relocated TS. Therefore this change does not involve an unreviewed safety question.

99-145 FSAR Update, Rev. 12

Reference Document No.: FSAR Update
Rev. No: 12
Reference Document Title: FSAR Update, Rev. 12
Safety Evaluation Description:
Westinghouse (West.) was contracted to review Chapters 6 and 15 and ensure that the technical information presented was consistent with the latest safety analyses of record. West. issued letter PGE-99-501 dated January 6, 1999, which identified numerous potential FSAR changes to Chapters 6 and 15. After reviewing the West. summary, it has been determined that 71 FSAR changes will be made as summarized in Attachment 1. These FSAR changes have been categorized into 14 typographical errors/grammatical corrections, 23 clarifications, 18 Reference updates or corrections, and 16 technical revisions. While all of these corrections help ensure the accuracy and the quality of the DCPP FSAR, only the technical revisions affect the "design basis description" of the FSAR and required any detailed evaluation.

Several technical revisions have been identified as requiring additional review and/or require a potentially significant text revision. Therefore, technical revisions 12, 15, 31, 32, and 57 will be processed and tracked separately from this revision package by AR A00490338. The technical revisions 21, 22, 50A, 52, 56, 68, 70A will be implemented as part of the Unit 1 Uprate LAR, 25 is addressed in the Spurious SI LAR, and 43 is being as part of an update to the Accident Analysis Profile Database.

The remaining technical revisions address inaccurate discussions of the input assumptions and /or analysis methodology, and provide additional detail based on subsequent evaluations. Only two technical revisions are considered significant. As documented in AR A0476550, there were two Offsite dose values for the Steam Generator Tube Rupture analysis listed in Table 15.5-71 which had not been updated.

1) UFSAR Table 15.5-71; SGTR Accident initiated iodine spike low population zone thyroid dose of 0.3 should be 1.3 REM.

2) UFSAR Table 15.5-71; SGTR Accident initiated iodine spike 2 hour thyroid dose of 28.8 REM should be 29.5 REM.

It should be noted that no new SGTR analysis has actually been performed and these slightly larger dose values were established in the SGTR evaluation which was documented in the Westinghouse Safety Assessment of increased Pressurizer Pressure Control System Uncertainty in PGE-93-659 (Ref. 2). These dose values are still within the 10CFR100 limits which the NRC established as the basis for acceptance in the SER (See LBIE response 2) and were previously reviewed and approved by PG&E. Although an FSAR revision was made in 1994 to reflect the increased pressure control uncertainty, it was determined that the existing dose values in Table 15.5-71 did not require revision at that time.

Safety Evaluation Summary:
The evaluation focuses on the fact that these FSAR revisions are essentially administrative changes being made to ensure consistency and accuracy with the documented safety analyses of record. There has been no actual change in any plant equipment, procedures, or analysis methodology.

Conclusion:
These FSAR corrections do not change any analysis related assumptions, methodology, or acceptance criteria and actually ensure the FSAR more accurately reflects the design basis analysis of record as performed by Westinghouse. Therefore these FSAR revisions/corrections do not constitute an unreviewed safety question.

99-146 Relocation of TS 3/4.11.2.6 to ECG 24.3

Reference Document No.: ECG 24.3
Rev. No: 3
Reference Document Title: Gaseous Radwaste - Explosive Gas Mixture
Safety Evaluation Description:
This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.11.2.6 to a new ECG (ECG 24.3). Relocation of TS 3/4.11.2.6 to the ECG program was authorized by LA 135. The TS 3/4.11.2.6 bases are also being included in the ECG. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
TS 3/4.11.2.6, "Gaseous Radwaste - Gas Storage Tanks," requires the quantity of radioactivity contained in each gas decay tank to be limited to less than or equal to 10^5 curies noble gases (considered as Xe-133 equivalent).

This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.11.2.6 to a new ECG (ECG 24.3). Relocation of TS 3/4.11.2.6 to the ECG program was authorized by LA 135. The ECG and the ECG Bases have been written in the ITS format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Conclusion:
Relocation of TS 3/4.11.2.6 to the ECG program was authorized by LA 135. Use of ITS formatting, and the addition of licensing information for completeness, are administrative in nature and do not change the technical requirements of the relocated TS. Therefore, this change does not involve an unreviewed safety question.

99-147 Relocation of TS 3/4 10.4 to ECG 41.2

Reference Document No.: ECG 41.2
Rev. No: 0
Reference Document Title: Special Test Exceptions - Position Indication
Safety Evaluation Description:
This change relocates the requirements of TS 3/4 .10.4 "Special Test Exceptions - Position Indication System - Shutdown," to a new ECG (ECG 41.2). Relocation of TS 3/4 .10.4 to the ECG program was authorized by License Amendments (LAs) 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999. The TS 3/4.10.4 Bases are also being relocated to the ECG. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional information has been added to the Bases to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
This permits the Position Indication Systems to be inoperable during rod drop time measurement. The data necessary to determine the rid drop time is derived from the position indicator coils and is small compared to the normal voltage and cannot be observed if the system remains operable. This change relocates the requirements of TS 3/4.10.4 "Special Test Exceptions - Position Indication System - Shutdown," to a new ECG (ECG 41.2). Relocation of

TS 3/4.10.4 to the ECG program is an administrative change, authorized by License Amendments (LAs) 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999, that does not impact the requirements for operation and control of the plant.

Conclusion:
Since the relocation of TS 3/4.10.4 to ECG 41.2 has been approved by the NRC, based on a safety evaluation (Reference LAs 135/135), the technical requirements remain unchanged. In addition the reformatting and additional information added to the Bases is administrative. Therefore, proposed ECG 41.2 does not result in an unreviewed safety question.

99-148 Relocation of TS 3/4.9.3 to ECG 42.1

Reference Document No.: ECG 42.1
Rev. No: 0
Reference Document Title: Refueling Operations - Decay Time
Safety Evaluation Description:
This change relocates the requirements of TS 3/4.9.3, "Refueling Operations - Decay Time," to a new ECG (ECG 42.1). Relocation of TS 3/4 .9.3 to the ECG program was authorized by License Amendments 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional information has been added to the Bases to improve usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
The requirements in this ECG provide for short lived fission products in the reactor core decaying. This results in reduced fuel handling personnel exposure and reduces the decay heat demands on the spent fuel pool cooling system. This change relocates the requirements of TS 3/4.9.3, "Refueling Operations - Decay Time," to a new ECG (ECG 42.1). Relocation of TS 3/4.9.3 to the ECG program is an administrative change, authorized by License Amendments 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999, that does not impact the requirements for operation and control of the plant.

Conclusion:
Since the relocation of TS 3/4.9.3 to ECG 42.1 has been approved by the NRC, based on a safety evaluation (Reference LA 135/135), and the technical requirements remain unchanged. In addition, the reformatting and additional information added to the Bases is administrative. Therefore, the proposed change does not involve an unreviewed safety question.

99-149 Relocation of TS 3/4.9.3 to ECG 42.2

Reference Document No.: ECG 42.2
Rev. No: 0
Reference Document Title: Refueling Operations - Communications
Safety Evaluation Description:
This change relocates the requirements of TS 3/4.9.5, "Refueling Operations - Communications," to a new ECG (ECG 42.2). Relocation of TS 3/4 .9.5 to the ECG program was authorized by License Amendments 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional information has been added to the Bases to improve usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
The requirements in this ECG provide for short lived fission products in the reactor core decaying. This results in reduced fuel handling personnel exposure and reduces the decay heat demands on the spent fuel pool cooling system. This change relocates the requirements of TS 3/4.9.5, "Refueling Operations - Communications," to a new ECG (ECG 42.2). Relocation of TS 3/4.9.5 to the ECG program is an administrative change, authorized by License Amendments 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999, that does not impact the requirements for operation and control of the plant.

Conclusion:
Since the relocation of TS 3/4.9.5 to ECG 42.2 has been approved by the NRC, based on a safety evaluation (Reference LA 135/135), and the technical requirements remain unchanged. In addition, the reformatting and additional information added to the Bases is administrative. Therefore, the proposed change does not involve an unreviewed safety question.

99-150 Relocate TS 3/4.9.7 to ECG 42.4

Reference Document No.: 0
Rev. No: ECG 42.4
Reference Document Title: Refueling Operations - Crane Travel - Fuel Handling Building
Safety Evaluation Description:
This change relocates the requirements of TS 3/4.9.7, "Refueling Operations - Crane Travel - Fuel Handling Building," to a new ECG (ECG 42.4). Relocation of TS 3/4 .9.7 to the ECG program was authorized by License Amendments 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional information has been added to the Bases to improve usefulness of the Bases.

The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
The restriction on movement of loads in excess of the nominal weight of fuel and control assembly and associated handling tool, except the moveable building walls, over other fuel assemblies in the spent fuel pool ensures that in the event a load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of the fuel in the storage racks will not result in a critical array. These assumptions are consistent with the activity release assumed in the safety analyses. This change relocates the requirements of TS 3/4.9.7, "Refueling Operations - Crane Travel - Fuel Handling Building," to a new EC (ECG 42.4). Relocation of TS 3/4.9.7 to the ECG program is an administrative change, that does not impact the requirements for operation and control of the plant.

Conclusion:
The relocation of TS 3/4.9.7 to ECG 42.4 has been approved by the NRC, based on a safety evaluation (Reference LAs 135/135). The technical requirements remain unchanged. In addition, the reformatting and additional information added to the Bases is administrative. Therefore, proposed ECG 42.4 does not result in an unreviewed safety question.

99-151 Secure the Plant

Reference Document No.: STP
Rev. No: 0
Reference Document Title: Secure the Plant
Safety Evaluation Description:
PG&E is implementing a series of design changes that make unauthorized access to the power block or intake structure more difficult. These changes target intruders after they penetrate the protected area barrier. The changes reduce the number of entry points into the power block; harden weaker potential entry points; add three hardened guard booths at strategic locations; and add fencing between the protected area barrier and the power block to delay access or force intruders past guard booths.

As part of this project, PG&E has conducted evaluations to ensure that the design changes satisfy design and licensing bases. Evaluations were performed to address: effects on operation of the ventilation systems; natural phenomena including seismic events and tornado effects; internally initiated events including high energy line breaks and fire; maintenance and construction activities; environmental protection; and impact on operator response, the security plan and emergency plan.

Safety Evaluation Summary:
The following is a summary safety evaluation which describes each important design, operation and maintenance feature, and safety consideration involved in evaluating the project. PG&E has performed Licensing Basis Impact Evaluations and/or screenings for the project design changes.

This enveloping safety evaluation was performed independent of the design change process. Licensing documents were reviewed for potential impact by the changes. EDMS searches were used as an aid to focus on specific sections of documents which may be impacted by the changes. The documents searched therefore included all those in EDMS LBIE searches: the License, Technical Specifications, Improved Technical Specifications, Equipment Control Guidelines, FSAR Update, and Design Criteria Memorandums. In addition, the FSAR was searched manually, in particular, drawings.

Increased Time to Enter the Plant

Installation of the project may result in a slight delay in access to the plant. However, this slight delay does not affect the ability of personnel to perform required safety functions. The most applicable requirement is to staff the TSC within an hour. The normal entrance to the TSC from outside the plant is unaffected by this project. The following summarizes evaluations of the changes which could impact entering the plant:

- New security door on the bridge (door 517A-2): The delay associated with an additional key card door is insignificant.
- Elimination of the entrance to the diesel hallway (door 102-2): This entrance has been strongly discouraged for some time as an access to the plant. The loss of this entrance causes a minor inconvenience for a few who use this entrance to access the diesel rooms from outside, but this is not a significant impact. This change does not result in a licensing basis impact.
- Magnetic switch gate (door 129A) is battery backed so that it won't fail closed on a loss of power. Personnel are still able to use doors 129A and 129 as an exit in the event of a loss of both normal power and the backup battery. Any delay caused by the door is considered insignificant.
- In general, Operations personnel are not affected by the changes because they do not respond from outside the plant.

Also, prior to implementation, the changes were posted for two weeks to model their affect on plant activities. An Operations crew reviewed the changes for potential impact to required response times. The changes were found not to impact the required response times.

Operator Response Time

The changes affect the exterior doors of the power block. Since required operator actions following an event are within the power block, by operators stationed in the power block, transiting between locations within the power block were concluded to not be significantly affected. However, some actions require operators to transit between:

- Control Room to TSC
- Control Room to Intake

Based on assessments by operations fire protection and emergency planning, adequate access to, and egress from, the Turbine Bldg., Aux. Building and TSC are maintained for operators, fire protection brigade response, and emergency exit of personnel. The impacted doors were posted for an extended period to collect concerns from personnel. All concerns were resolved. The number of doors to the TSC is being reduced which was determined to not impact operators response time during an emergency. The addition of the card reader on the 5th floor bridge was determined to result in no significant delay to support personnel in the administration building. Personnel are aware that bypassing a card reader, during an emergency exit, is acceptable.

Operator training cycles 99-1 and 99-2 included briefings to all crews on the Secure The Plant changes, with no substantial issues identified.

Procedure OP1.DC11 "Conduct of Operations-Abnormal Plant Conditions" Attachment 7.3 "Operator Action Times Assumed in the FSAR" was reviewed and verified to not be impacted by the changes.

Ventilation System

Doors 389-1 and 389-2 are the only ventilation boundary doors affected by the Secure The Plant changes. Since these doors are sealed shut, the potential for a breach of this ventilation boundary has been reduced. Calc. HVAC-90-32 R20 reflects these changes. Design pressures have been included/enveloped with the structural evaluation. STP M-5 "Routine Surveillance Test of the Fuel Handling Building Ventilation System" makes reference to doors 389 & 389-2 and is revised. DCM S-23C is unaffected by this change. ECG 80.1 (from an HVAC concern) is unaffected by this change. TS 3/4.9.12 (ITS 3.7.13) "Fuel Handling Building (FHB) Ventilation System" and its bases are unaffected by this change; no change to the SAR is required. Surveillance 4.9.12.d.3) (ITS 3.7.13.4) requires the FHB achieve a negative pressure, which may be easier to achieve with the elimination of these doors.

The new security grating was evaluated for potential impact to the emergency diesel generators' hot weather plan. Engineering confirmed that the grating would cause a very small increase in resistance to air flow. The change was concluded to not affect the overall air flow of the diesel generator ventilation system.

High Energy Line Break

The high energy line break (HELB) analysis for the turbine building was revised to address the planned modifications. The analyses confirmed there is no adverse impact on the HELB analysis because the net effective vent area is maintained or increased.

Doors 389 and 389-2 are HELB Doors. These doors are part of the Area L Compartment 1 HELB boundary in plant P/T analyses. Changing these doors to concrete block walls does not adversely impact the Area L P/T analyses since these doors are modeled as closed, and were capable of withstanding the associated HELB differential pressure conditions. The replacement block walls meet the same requirements as the doors had previously; refer to DCM T-12 Section 4.3.3.1.2 (Area L) and Figures 4-42 through 4-45 for Area L P/T analyses results for the design input. To support this change, engineering issued calcs N-011 Rev. 2, M-297 Rev. 6, M-458 Rev. 2, M-355, Rev. 4 and M-478 Rev. 2, all to reflect the new plant configuration (for HELB issues). No change to the facility design, function, or method of performing the function as described in the SAR, including text, tables, and figures are created by this proposed modification.

Internal Missile Analysis

In the event of a high energy line break in the Turbine Building, several doors are assumed to fail or "blowout" to relieve the steam pressure. These blowout doors are a potential missile hazard. The number of potential internal missiles are being reduced because several of these blowout doors are being eliminated and replaced with permanent grating, or block walls, which do not blowout and become missiles.

Flood Protection

The changes do not affect doors credited with flood protection.

Fire Protection

The fire brigades timely access to fires in certain areas could be hampered by the changes. This is particularly a concern in the transformer area. The fences are provided with gates to allow access for fire fighting, and access

to hydrants. Fire brigade will be able to unlock the gates in a timely manner. Therefore, the new fences do not reduce the level of fire protection.

Fire rated doors being eliminated are replaced with an equivalent (or better) fire rated concrete block wall which results in no adverse impact to the fire zone (199-1, 389-1, 389-2). As stated above, only changes to several FSAR figures are required. Since the passive fire rating of the compartment is maintained and the wall's function, design & commitment is maintained, the screen question may be answered "No". The fire rating "A" function of doors 101-1, 101-2. 102-1, and 102-2 is not affected by the hardening of these doors. There is no change to the fire rating of the new card reader door 517A-2.

Seismically Induced Systems Interaction Issues

The project does not have an adverse impact on SISIP. The changes are not in areas containing SISIP targets. Furthermore, the project replaces several items such as louvers and doors with items of greater strength.

Tornado Protection

While the components being changed are not specifically credited with tornado protection, the changed components have equal or greater strength. For example, the hardened grating on exterior openings are much less likely to be pulled off and become a missile in the event of a tornado.

Loss of Power

Power for the access control system equipment changes (key card doors) remains operable from independent power sources in the event of loss of normal power. Switchover to standby power is automatic.

The new gate at door 129 has a battery to the magnetic control panel to provide a backup power source for the switch. This will prevent inadvertent release of the magnetic switch (and subsequent closure of the gate) in the event of primary power loss. Egress through door 129A remains available in a loss of normal and battery power.

FSAR

No change to the facility design, function, or method of performing the function as described in the SAR, including text, tables, and figures are

created by this proposed modification except as described here. The only portions of the FSAR that are relevant to the changes being proposed by this DCP is as follows:

1) FSAR Appendix 9.5A (Fire Hazard Analysis), Fire Area TB-7, Fire Zone 19-A, Sec 1.3.2 (Elev 104 Feet, West) identifies the doors BU201-2, BU211-2 & BU216-2 as "3 nonrated doors communicate to the buttress area." While the doors remain in place, door 201-2 will not be able to "communicate with the buttress area" due being welded shut and having a block wall installed behind it. Therefore, the statement in the FSAR Fire Hazards analysis is being revised (A0486995 eval 5), and LBIE Screen section 4.2 was checked "yes".
2) Several of the fire protection barrier drawings are affected by the removal of several doors; these drawing are presented in FSAR appendix 9.5F (Fig's 9.5F-7. 9-5F-9, & 9.5F-17). However, since these are controlled drawings (515568, 515570 & 515578), and are being revised as part of this DCP, the Licensing department prepares a change request as part of their standard protocol within 6 months after completing each Unit 2 refueling to cover all of the controlled drawings that changed. This sole change to the aforementioned figures does not impact the contents of the licensing basis documents related to the design, function, methods of performing and regulatory commitments.

Conclusion:
The above safety evaluation describes each important design, operation and maintenance feature, and safety consideration involved in evaluating the project. PG&E has performed Licensing Basis Impact Evaluations and/or screenings for the project design changes. In accordance with 10 CFR 50.59, 10 CFR 50.54p, and 10 CFR 51.22, respectively, PG&E determined that these changes do not constitute an unreviewed safety or environmental question or require a change to the TS and do not reduce the effectiveness of the security plan.

99-152 Revise Raw Water Reservoir Capacity

Reference Document No.: DCP M-049497
Rev. No: 0
Reference Document Title: Revise Raw Water Reservoir Capacity
Safety Evaluation Description:
The capacity of the raw water reservoir is determined to be 5.0 million gallons based on more accurate measurement method. Plant Manual Volume 9B data has been updated to reflect this correct capacity. Also, the reference bench for measuring the reservoir capacity is changed to the centerline of the drain pipe. Plant design and Licensing documents need to be updated as a result through the design change process. There is no physical change required in the plant.

Safety Evaluation Summary:
The change to the raw water reservoir capacity requires update of various sections of the UFSAR. The reference bench mark of the reservoir level is also changed to the centerline of the gravity drain pipe instead of the top of the pipe. This results in changes to surveillance test procedures which are described in the UFSAR and committed in ECG 18.1. Changes to these Licensing bases documents requires a Licensing Bases Impact Evaluation per 10 CFR 50.59.

Conclusion:
The increase in the raw water reservoir capacity will not cause any accident or malfunction of equipment. It will not increase the probability of occurrence of any accident or equipment malfunction. There is no change to the current Technical Specification or the Improved Technical Specification or its bases. The change to these documents will not raise an Unreviewed Safety Question.

99-153 Relocation of TS 3/4.11.2.5 to ECG 24.2

Reference Document No.: ECG 24.2
Rev. No: 0
Reference Document Title: Gaseous Radwaste - Explosive Gas Mixture
Safety Evaluation Description:
This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.11.2.5 to a new ECG (ECG 24.2). Relocation of TS 3/4.11.2.5 to the ECG program was authorized by LA 135. The TS 3/4.11.2.5 bases are also being included in the ECG. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
TS 3/4.11.2.5, "Gaseous Radwaste - Explosive Gas Mixture," requires the concentration of oxygen in the Gaseous Radwaste System to be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

This LBIE evaluates the acceptability of relocating the requirements of TS 3/4.11.2.5 to a new ECG (ECG 24.2). Relocation of TS 3/4.11.2.5 to the ECG program was authorized by LA 135. The ECG and the ECG Bases have been written in the ITS format. Additional licensing information has been added to the Bases as necessary for completeness to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Conclusion:
Relocation of TS 3/4.11.2.5 to the ECG program was authorized by LA 135. Use of ITS formatting, and the addition of licensing information for completeness, are administrative in nature and do not change the technical requirements of the relocated TS. Therefore, this change does not involve an unreviewed safety question.

99-154 Relocate Technical Specification 3/4.9.6 to ECG 42.3

Reference Document No.: ECG 42.3
Rev. No: 0
Reference Document Title: Refueling Operations - Manipulator Crane
Safety Evaluation Description:
This change relocates the requirements of TS 3/4.9.6, "Refueling Operations - Manipulator Crane," to a new ECG (ECG 42.3). Relocation of TS 3/4.9.6 to the ECG program was authorized by License Amendments (LA) 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999. Both the ECG and the ECG Bases have been written in the improved TS (ITS) format. Additional information has been added to the Bases to improve the usefulness of the Bases. The additions to the Bases, and use of ITS formatting, are administrative in nature and do not change the technical requirements of the relocated TS.

Safety Evaluation Summary:
The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of control rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. This change relocates the requirements of TS 3/4.9.6, "Refueling Operations - Manipulator Crane," to a new ECG (ECG 42). Relocation of TS 3/4.9.6 to the ECG program is an administrative change authorized by License Amendments (LA) 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999, that does not impact the requirements for operation or control of the plant.

Conclusion:
Since the relocation of TS 3/4.9.6 to ECG 42.3 has been approved by the NRC, based on a safety evaluation (Reference LA 135/135), and the technical requirements remain unchanged. In addition, the reformatting and additional information added to the bases is administrative. Therefore, the proposed change does not involve and unreviewed safety question.

99-155 Revision 5 to ECG 0.0

Reference Document No.: ECG 0.0
Rev. No: 5
Reference Document Title: Equipment Control Guidelines - Applicability
Safety Evaluation Description:
As noted in AR A0492262 and associated A/E #1, ECG 0.3 provides no guidance for ECG violations that occur in Modes 5, 6, or Defueled. ECG 0.3 is currently only applicable in Modes 1 through 4. If a violation occurs, or no action is provided for a condition for an ECG that is applicable outside of Modes 1 through 4, ECG 0.3 provides no guidance for resolving the issue. Other guidance applies, e.g., OM7.ID1, "Problem Identification and Resolution - Action Requests," provides guidance for problem identification and resolution, but that guidance is not explicitly stated in ECG 0.3. To provide the necessary guidance, the proposed change will remove the Mode 1 through 4 limitation from ECG 0.3.

This change also replaces the reference to OP1.DC16, Attachment 7.2 (which no longer exists) with a reference to FSAR Table 16.1-1.

Safety Evaluation Summary:
In response to Quality Evaluation Q0011796, ECG 0.0, Revision 2 was issued in 1997 to make the wording of ECG 0.4 consistent with TS 3.0.4. As part of this change, ECG 0.3 was also modified to make it applicable only in Modes 1 though 4, consistent with TS 3.0.3. AR A0492262 and its associated A/E #1 discuss the fact that ECG 0.3 provides no guidance for ECG violations that occur in Modes 5, 6, or Defueled. (Guidance does exist elsewhere in OM7.ID1, "Problem Identification and Resolution - Action Requests," for problem identification and resolution.) Regulatory Services has rereviewed the intent of ECG 0.3, and finds that it is not intended to match TS 3.0.3. The intent of TS 3.0.3 is to establish the shutdown actions that must be met when a TS LCO is not met or the condition is not specifically addressed by the action statements. ECG 0.3 is intended to provide guidance when an ECG action is not met, for all conditions for which the ECG applies. Accordingly, ECG 0.3 should not be restricted to Modes 1 through 4. Removing the Mode 1 through 4 limitation from ECG 0.3 will make it applicable for all conditions for which any given ECG applies, and will make the requirement more conservative than currently. The other requirements of ECG 0.3 remain unchanged.

The change to replace the reference to OP1.DC16, Attachment 7.2 (which no longer exists) with a reference to FSAR Table 16.1-1 is administrative only, and has no affect on ECG requirements.

Conclusion:
The change to remove the Mode 1 through 4 limitation from ECG 0.3 will result in providing the necessary guidance on resolving ECG violations that occur in any conditions for which an ECG applies. ECG 0.3 will be more conservative in that it will apply for all conditions that apply to an ECG. The other requirements of ECG 0.3 remain unchanged.

The change to replace the reference to OP1.DC16, Attachment 7.2 with a reference to FSAR Table 16.1-1 is administrative only, and has no affect on ECG requirements. Therefore this change does not result in an unreviewed safety question.

99-156 Lesser Rated Penetration Seals in Selected Barriers

Reference Document No.: FHARE 109
Rev. No: 0
Reference Document Title: Acceptance Criteria for Penetration Seals in Selected Barriers

Safety Evaluation Description:
Licensee Event Report (LER) 1-94-001-01, "Inadequate Fire Barrier Penetration Seals Resulting From A Lack Of Damming", dated August 11, 1994, identified that the fire endurance ratings of a certain population of silicone foam penetration seals at Diablo Canyon Power Plant (DCPP) were "indeterminate" due to the missing damming material. A review of the fire tests for this population of foam seals, with missing damming material, revealed that insufficient data is available to assign a fire rating. This review indicated that these seals would not satisfy the non-fire side (cold side) temperature criteria established by fire penetration test standard ASTM E-119. Due to the type and size of penetrating commodities these penetration seal configurations do not lend themselves to replacement with existing 3 hour rated standard penetration seal designs. FHARE 109 demonstrates the acceptability of lesser rated penetration seals. This evaluation also documents the basis for implementing criteria for acceptance of lesser rated penetration seals installed in selected fire barriers at DCPP.

Safety Evaluation Summary:
NRC guidance requires that DCPP satisfy one of the following options to qualify penetration seals: 1) Seals must be installed with the same fire endurance rating as the barrier (BTP APSCB 9.5-1 and 10CFR50, Appendix R); or, 2) Perform an evaluation to demonstrate that an "equivalent level of protection" is provided or perform an evaluation to demonstrate that the seal(s) will prevent a fire from spreading to the adjacent area (GL 86-10). For a penetration seal to be qualified using the first option, the following three fire test criteria must be satisfied:

- No passage of flame, smoke or hot gasses for the duration of the test;

- Withstand the effects of a hose stream following the fire without projection of water; and,
- Maintain the temperature on the unexposed (i.e., non-fire) side as follows:
  - 250 °F above the initial (ambient) surface temperature of the barrier
  - 325 °F above the initial (ambient) surface temperature of the seal
  - 700 °F absolute temperature for electrical cable penetrations

As discussed above, the seals evaluated by FHARE 109 may not pass the non-fire side temperature criteria. NRC guidance also permits penetration seals to be qualified using different test standards. In this evaluation, ASTM E-814 is used in conjunction with option 2, above. The ASTM E-119 method assigns each penetration seal a "F" and "T" rating based on fire test performance. The "F" rating indicates a seals fire endurance capability, in hours. The "T" rating indicates the non-fire side temperature after a specified time period, in hours or minutes. In summary, this evaluation will determine if these seals will prevent a fire from spreading to an adjacent area by comparing each seals "F" and "T" rating with the known fire duration for each area/zone.

FHARE 109 establishes the basis for assessing lesser rated penetration seals. Acceptability of these seals will be based on evaluating their ability to perform, i.e., prevent propagation of flame, passage of smoke and hot gas, and satisfy non-fire side temperature criteria, against the expected fire exposure. This is accomplished by having the fire rating of the seal at least equal to the hazard present on either side of the barrier or equal to the fire rating of the barrier. FHARE 109 demonstrates that for specific penetration seal located in low fire duration areas satisfying the non-fire side temperature criteria based on the hazard present provides an adequate level of fire protection.

Conclusion:
The combustible loading in the areas/zones where the penetration seals evaluated by FHARE 109 are located is low. In addition, general housekeeping practices and strict administrative controls (i.e., controls of ignition sources and transient combustibles) consistently maintain DCPP in a safe, controlled condition consistent with the fire protection Defense-In-Depth philosophy. A review of the basis for qualification of penetration seals was performed, and it is concluded that the acceptance criteria as defined by the regulatory guidance to fully qualify penetration seals in fire barriers is not warranted for all fire areas.

Based on a review of the existing field conditions at DCPP, it is determined that the penetration seals installed in fire barriers which serve to separate Low Combustible Fire Areas (i.e., less than 45 minutes of fire duration) will prevent a fire from spreading to an adjacent area. The fire loads in the area\zones

where these seals are located warrant use of a relaxed non-fire side temperature criteria for penetration seal designs. It is DCPP's position that the standard non-fire side temperature criteria is not critical to qualify the penetration seals evaluated by FHARE 109. Therefore, the lesser rated penetrations seals discussed in FHARE 109 do not adversely impact the DCPP fire protection program. This activity does not result in an unreviewed safety question.