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PG&E Letter DCL-04-046

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 Changes, Tests, and Experiments for the Period January 1, 2002, through December 31, 2003

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.59, "Changes, Tests, and Experiments," Pacific Gas and Electric Company (PG&E) is enclosing the 10 CFR 50.59 Report for Diablo Canyon Power Plant (DCPP), Units 1 and 2, for the period January 1, 2002, through December 31, 2003. The report provides a summary of all 10 CFR 50.59 evaluations performed during this period.

Evaluations performed in accordance with 10 CFR 50.59 are performed as part of PG&E's licensing basis impact evaluation (LBIE) process. Since the LBIE process is used to perform reviews for compliance with regulations in addition to 10 CFR 50.59, some LBIEs do not include a 10 CFR 50.59 evaluation and, therefore, are not included in this report.

The Plant Staff Review Committee has reviewed the referenced LBIEs and has concurred that the changes do not require prior NRC approval or require changes to the DCPP Technical Specifications.

If you have any questions or require additional information, please contact Stan Ketelsen at (805) 545-4720.

Sincerely,

David H. Oatley

IE47

jer/3664
Enclosure

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10 CFR 50.59 REPORT OF CHANGES, TESTS, AND EXPERIMENTS
for the Period
January 1, 2002, through December 31, 2003

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

Acronyms and Abbreviations

1R10	Unit 1 Refueling Outage No. 10
AC	alternating current
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CVCS	chemical and volume control system
DCM	Design Criteria Memorandum
DCP	design change package
DCN	design change notice
DCPP	Diablo Canyon Power Plant
DEH	digital electro-hydraulic
EARS	emergency assessment and response system
ECCS	emergency core cooling system
ECG	equipment control guidelines
EOF	Emergency Operations Facility
EPRI	Electric Power Research Institute
ERDS	Emergency Response Data System
ERFDS	Emergency Response Facility Data System
F	Fahrenheit
FHB	fuel handling building
FSARU	Final Safety Analysis Report Update
ISI	inservice inspection
ISLT	inservice leak test
IEEE	Institute of Electrical and Electronic Engineers
LA	license amendment
LAR	license amendment request
LBIE	licensing basis impact evaluation
LCO	limiting condition for operation
LOCA	loss-of-coolant accident
LTOP	low temperature overpressure protection
mg	milligram
MOV	motor-operated valve
MP	maintenance procedure
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OP	operating procedure
PAM	post-accident monitoring
PID	proportional integral derivative
PORV	power-operated relief valve
PPC	plant process computer

Acronyms and Abbreviations (continued)

PRA	probabilistic risk assessment
psig	pounds per square inch, gage
PSV	pressurizer safety valve
RCCA	rod cluster control assembly
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RSE	reload safety evaluation
RWST	refueling water storage tank
SE	safety evaluation
SER	safety evaluation report
SG	steam generator
SI	safety injection
SMM	subcooled margin monitor
SPDS	safety parameter display system
SSC	structures, systems, and components
SSER	supplemental safety evaluation report
TP	temporary procedure
TS	Technical Specification
TSC	Technical Support Center
U1	Unit 1
U2	Unit 2

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02-002 RHR Line 1-S6-509-8 Venting/SI-1-8818D Post-Stroke Leak Testing

Reference Document No.: TP TO-0110/TP TB-0106
Rev. No: 0
Reference Document Title: RHR Cold Leg Injection Line Venting, SI-1-8818D Leak Test

Activity Description:
Two temporary procedures are proposed to allow venting of the RHR system. Accumulator 1-3 out leakage is occurring through the RHR second-off check valve, SI-1-8818C. The nitrogen-laden water from the accumulator is degassing when transitioning from a high to a low-pressure system. A noncondensible gas void (primarily nitrogen) has accumulated in a 40-foot horizontal section of Line 1-S6-509-8, upstream of RHR second-off check valves, SI-1-8818C & D. The void has grown large enough that associated piping could experience a destructive water hammer at the onset of an SI event. Venting may be required to allow continued plant operation. There are no suitable vents on the upstream side of these check valves. This necessitates using a dual vent valve path, SI-1-104 and 105, on the downstream side of SI-1-8818D, because this is the high point for this specific pipe section.

TP TO-0110 will provide the operational guidance to vent the RHR loops 3 and 4 cold-leg injection lines. Accumulator 1-4 will be isolated immediately prior to the venting process, which will require entry into Action B.1 for TS 3.5.1.

TP TB-0106 will be used for performing a leak test in accordance with Surveillance Requirement 3.4.14.1, since the venting process will cause a small amount of forward flow through check valve SI-1-8818D.

Summary of Evaluation:

TP TO-0110
A large-break LOCA coincident with an isolated accumulator could threaten ECCS acceptance criteria, for example the peak-cladding temperature limit (2200°F). This serious risk mandates a 1-hour TS completion time requirement, so that the probability of occurrence remains very low. This time is reflected in Action Statement B.1 in LCO 3.5.1. In addition, PRA Calculation PRA 01-06 states that the NRC has accepted the risk results in the Westinghouse analysis for extending an inoperable accumulator completion time to 24 hours. (Subsequent to this evaluation, Action Statement B.1 in LCO 3.5.1 was revised to provide a completion time of 24 hours, as approved by License Amendments 160 (Unit1) and 161 (Unit 2) dated August 15, 2003). This venting activity is expected to meet the 1-hour

TS completion time. Although the Accumulator 1-4 water volume is unavailable to contribute to the containment water level in an accident, the RWST level margin compensates for this "lost" water. There is a high degree of confidence shown by a conservative calculation performed by valve component engineering that the Accumulator 1-4 isolation MOV will open following its closure at power. If the valve would not open electrically, the MOV can be opened using the local manual handwheel. Hence, the overall risk is acceptable because of the compensating factors discussed above, and the low probability of occurrence due to the 1-hour completion time requirement.

TP TB-0106
The post-stroke leak rate test of SI-1-8818D is evaluated in this LBIE, because it is an integral part of this corrective maintenance activity. Partially stroking this check valve at power has the potential to lead to degraded seat leakage and intersystem LOCA concerns. This is not expected because the cleanliness requirements of the systems involved (RWST, ECCS) are stringent and the potential for introducing debris into the check valve seat area is low. The RCS first-off check valve, SI-1-8948D, was tested in 1R10 with no leakage recorded, i.e. leak tight. Also, the actual check valve test procedure is benign and must be balanced against the risk of shutting down the unit and establishing the special Mode 4 normal test conditions. The instruments used are installed plant gauges, with the exception of non-intrusive pyrometer readings, and a currently installed pressure gauge (approved as Jumper 01-07). All plant equipment is operated in its normal manner. All monitoring and measurements obtained in the procedure have no impact on plant operation.

Conclusion
The proposed changes do not result in more than a minimal increase in the frequency or consequences of accidents or malfunctions previously evaluated in the FSARU, do not create a new type of accident or malfunction not previously evaluated in the FSARU, and do not impact a fission product barrier or methodology described in the FSARU. Therefore, the proposed changes do not require prior NRC approval.

02-003 Clarify Commitment to RG 1.75

Reference Document No.: DCP E-049605
Rev. No: 0
Reference Document Title: Replace References to RG 1.75

Activity Description:
The change evaluated in this LBIE is the addition of specific criteria for electrical separation in the FSARU and the removal of references and implied commitments to RG 1.75 and IEEE 384:

1. RG 1.75 "Physical Independence of Electric Systems"
2. IEEE 384-1974, "IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits."

DCP E-049605 is being issued to replace references to RG 1.75 in the DCPP design and licensing basis with the actual installed design criteria for separation. The source of the references to RG 1.75 is the FSARU. There are only three topics that reference RG 1.75 in the FSARU: Section 7.2 on the seismic trip system; Section 7.5 on selected PAM instruments; and Section 8.3 for future Class 1E power systems.

Deleting the reference to RG 1.75 for the seismic trip system has no impact on the present design, installation, or maintenance. The NRC stated in SSER 8: "The seismic scram system is of similar design and meets the same criteria as the reactor protection system and is, therefore acceptable." The plant (reactor) protection system is designed to IEEE 279-1971 and IEEE 308-1971. These criteria are being added to the FSARU.

References to RG 1.75 are being deleted to make the design and licensing basis consistent for safety equipment at DCPP. The separation and isolation criteria for safety equipment and circuits are defined in the FSARU, DCMs, drawings, and procedures. PG&E has not established criteria for RG 1.75 because the RG states in Section D, Implementation, that: 1) the NRC will accept alternative methods and, 2) the NRC staff will use the RG for evaluating plants with construction permit safety evaluations issued after February 1,1974. DCPP safety evaluations for Unit 1 and 2 were issued in 1968 and 1969. RG 1.75 does not apply to the design at DCPP. The separation criteria applied in the plant are being added to the FSARU. Any future changes will be designed to the existing criteria based on IEEE 279-1971 and IEEE 308-1971, which were reviewed and accepted by the NRC.

Summary of Evaluation:
The criteria applied in the separation of safety-related instruments are being added to the FSARU. The references to RG 1.75 and IEEE 384-1974 are being removed. There are no physical changes to the plant and the applied criteria are being captured in the FSARU for future configuration control. Since there are no changes in the DCPP separation distances as accepted by the NRC, there are no effects on accidents and malfunctions previously evaluated in the FSARU, and there is no potential for creation of an event not previously evaluated in the FSARU. With the approved criteria reiterated in the FSARU, there is no change in evaluation methodology. Therefore, the proposed changes do not require prior NRC approval.

02-005 Accident Monitoring Instrumentation, SMM

Reference Document No.: ECG 7.8
Rev. No: 2
Reference Document Title: Accident Monitoring Instrumentation

Activity Description:
This change revises the ECG 7.8, Condition C, completion time for the RCS SMM from 48 hours to 7 days. The SMM is a RG 1.97, non-Category 1, non-Type A instrument and its requirements were relocated from TS as part of implementation of improved TS in 1999. The extension of the completion time to 7 days is based on the 7-day completion time for RG 1.97, Category 1 post-accident instruments controlled by TS 3.3.3.

Summary of Evaluation:
The SMM is not an accident or malfunction initiator. The pertinent 10 CFR 50.59 evaluation criteria are those dealing with consequences of accidents or malfunctions. In the event the SMM becomes inoperable during an accident, the operators are trained to perform manual calculations necessary to determine that plant safety functions are being performed. Therefore, there is no increase in the consequences of an accident or malfunction previously evaluated in the FSARU. Therefore, the proposed change does not require prior NRC approval.

02-006 DCPP Unit 1 Cycle 12 Reactor Core Fuel Load and COLR

Reference Document No.: DCP N-49611
Rev. No: 0
Reference Document Title: DCPP Unit 1 Cycle 12 Core Reload Design Change and COLR

Activity Description:
This DCP incorporates a new fuel-loading pattern for Unit 1 Cycle 12 into the plant design. This design also evaluates and accepts equivalency between the Westinghouse RCCA and Framatome RCCA for use at DCPP. Framatome Advanced Nuclear Power manufactures this equivalent RCCA with a design based on, and justified by, thermal hydraulic analyses, stress analyses, and physics analyses performed by Framatome Technologies. The main differences in the two models are the control rod cladding material, fabrication of the spider assembly and connection of the control rods to the spider, treatment of the cladding surface, and the diameter of the absorber material in the tip region of the control rods. These features are implemented on the Framatome model to enhance wear resistance.

Summary of Evaluation:
The Westinghouse RSE for the core reload, including use of the Framatome RCCA and COLR, includes analyses that verify no previously acceptable safety analysis criteria for any accident are exceeded, that there are no changes required to the plant TS, and that there are no changes that require prior NRC approval. PG&E has reviewed these analyses and concurs with the conclusions of the Westinghouse RSE that the Unit 1 Cycle 12 core reload and COLR do not require prior NRC approval.

02-007 Low Pressure Turbine Outer Cover Removal From Restricted Area

Reference Document No.: TP TD-0203
Rev. No: 0
Reference Document Title: Low Pressure Turbine Cover Handling

Activity Description:
TP TD-0203 will be used to control removal of the Unit 1 low-pressure turbine cover from a heavy load restricted area on the north end of the turbine building on the 140-ft. elevation. As the turbine cover weighs 70 tons, this is a heavy load handling operation. The load handling operation will be conducted while the Unit 1 reactor is defueled, and with movement of irradiated fuel or any other load handling in, or adjacent to, the spent fuel pool area curtailed while the heavy load is suspended over the Unit 1 restricted area. Plant SSCs potentially functioning beneath the overhead load path are back-up SSCs (i.e., emergency diesel power supplies) for removal of decay heat from the spent fuel pool. It is noted that these SSCs are beyond the scope of the DCPP Control of Heavy Loads Program (e.g., safe, cold reactor shutdown) but evaluated to satisfy 10 CFR 50.59 criteria for the temporary load handling procedure.

Summary of Evaluation:
An evaluation of the proposed activity concludes that all eight criteria of the 10 CFR 50.59 evaluation answer negatively. Decay heat removal in the spent fuel pool is not adversely affected by the proposed heavy load handling operation because the electric power to the cooling system is physically separated from the postulated drop area of the load and back-up, non-AC, make-up water sources are available to maintain spent fuel pool level decay heat removal by evaporative cooling. Therefore, the proposed activity does not require prior NRC approval.

02-010 Reverse Osmosis System for RWST (Unit 1 RWST drain line upgrade)

Reference Document No.: DCP N-049578
Rev. No: 0
Reference Document Title: Reverse Osmosis System for RWST Silica Cleanup

Activity Description:
LAs 144 (Unit 1) and 143 (Unit 2) authorize use of a reverse osmosis system to remove silica from each units' RWST in modes when the RWST is required to be operable. The LAs require that the reverse osmosis system suction be connected directly to the RWST drain line and that the drain line contain a flow-limiting device to preserve RWST inventory in case of reverse osmosis system leakage.

The LAs state in part that: "All piping and valves will be designed or qualified to Design Class I/seismic category I up to the discharge of the flow-limiting device or the isolation valve. This will ensure RWST pressure boundary integrity during a seismic event."

A portion of the RWST drain line, upstream of the flow-limiting device, is currently Design Class II, non-seismic category I. This activity will upgrade the existing section of pipe to Design Class I, seismic category I using the methods specified in LAs 144/143 for the refueling water purification piping system upgrade. This LBIE is performed because the RWST drain line upgrade is not specifically included in LAs 144/143.

The analytical inspection and dedication activities to upgrade the RWST drain line are identical to, or more stringent than, those used to upgrade the refueling water purification system piping and are:

- Performance of a pipe stress analysis for the upgraded drain line
- Non-destructive examination of all accessible welds
- Verification of pipe and fitting materials at all accessible locations
- Inclusion into the ISI Program under ASME Section XI, Class 3

Summary of Evaluation:
Upgrading the RWST drain lines to Design Class I, seismic category I will ensure that the pressure boundary integrity of the RWST drain line is maintained during a seismic event, thus preventing a loss of RSWT inventory when the reverse osmosis system is in operation. With the exception of maintaining pressure boundary integrity, the RWST drain line is not credited for safe shutdown or accident mitigation. The RWST drain lines do not impact any other systems and thus cannot create any new failure modes.

Therefore, the upgrade of the RWST drain line using the methodology approved in LA144/143 for the refueling water purification system is not a departure from a method of evaluation approved by the NRC and prior NRC approval is not required.

02-011 Reverse Osmosis System for RWST (Unit 2 RWST drain line upgrade)

Reference Document No.: DCP N-050578
Rev. No: 0
Reference Document Title: Reverse Osmosis System for RWST Silica Cleanup

Activity Description:
LAs 144 (Unit 1) and 143 (Unit 2) authorize use of a reverse osmosis system to remove silica from each units' RWST in modes when the RWST is required to be operable. The LAs require that the reverse osmosis system suction be connected directly to the RWST drain line and that the drain line contain a flow-limiting device to preserve RWST inventory in case of reverse osmosis system leakage.

The LAs state in part that: "All piping and valves will be designed or qualified to Design Class I/seismic category I up to the discharge of the flow-limiting device or the isolation valve. This will ensure RWST pressure boundary integrity during a seismic event."

A portion of the RWST drain line, upstream of the flow-limiting device, is currently Design Class II, non-seismic category I. This activity will upgrade the existing section of pipe to Design Class I, seismic category I, using the methods specified in LAs 144/143 for the refueling water purification piping system upgrade. This LBIE is performed because the RWST drain line upgrade is not specifically included in LAs 144/143.

The analytical inspection and dedication activities to upgrade the RWST drain line are identical to, or more stringent than, those used to upgrade the refueling water purification system piping and are:

- Performance of a pipe stress analysis for the upgraded drain line
- Non-destructive examination of all accessible welds
- Verification of pipe and fitting materials at all accessible locations
- Inclusion into the ISI Program under ASME Section XI, Class 3

Summary of Evaluation:
Upgrading the RWST drain lines to Design Class I, seismic category, I will ensure that the pressure boundary integrity of the RWST drain line is maintained during a seismic event, thus preventing a loss of RSWT inventory when the reverse osmosis system is in operation. With the exception of maintaining pressure boundary integrity, the RWST drain line is not credited for safe shutdown or accident mitigation. The RWST drain lines do not impact any other systems, and thus cannot create any new failure modes.

Therefore, the upgrade of the RWST drain line using the methodology approved in LA144/143 for the refueling water purification system is not a departure from a method of evaluation approved by the NRC, and prior NRC approval is not required..

02-012 Upgrade PORV Automatic Actuation Circuitry

Reference Document No.: DCP J-50569
Rev. No: 0
Reference Document Title: Upgrade PORV Automatic Circuitry

Activity Description:
In order to prevent the escalation of the "inadvertent SI at power" accident, the Class II automatic actuation circuitry for the safety-related PORVs (PCV-455C and PCV-456) will be upgraded to Design Class I. The upgrade will involve isolating the pressurizer high-pressure PORV actuation relays (PC-455EX, PC-456EX, PC-457EX and PC-474BX) from the Design Class II portions of the instrument loops (actuating the relays directly from Eagle 21). Then the automatic actuation of the PORV can be credited for ensuring that the PSVs are not actuated during an inadvertent SI at power.

In order to continue supporting the Design Class II pressurizer pressure control scheme, control of PCV-474 is being moved to the PT-455/PT-457 (control via the master controller) and PT-474 (interlock) transmitters. The actuation relay (PC-455IX) will be actuated by the PID controller that previously controlled PCV-455C.

The PT-403A and PT-405A (alternate LTOP) transmitter signals will be processed through Eagle 21. The alternate LTOP transmitter channels from Eagle 21 will be used for LTOP, 8701/8702 interlock, and PI-403A (previously PI-403) indication. The change will also provide control room indication for the PT-405A instrument loop, and PPC indication for both PT-403A and PT-405A (via ERFDS). The RG 1.97 function currently performed by PT-403 and PT-405 (via PR-403 and PI-405) will continue to be performed by PT-403 and PT-405.

Summary of Evaluation:
The changes that affect the licensing requirements are the upgrade of the PORV automatic actuation circuitry and moving the control function for PCV-474 (FSARU Figure 7.7-4). This change adds a safety function for the PORV automatic actuation circuitry to mitigate the consequences of a FSARU Chapter 15 accident. As a result of the new protective function of PCV-455C, PCV-474 will now be used for the pressurizer pressure control function. The licensing requirements for the PORVs will have to be updated to include these new functions along with the associated LCOs and the surveillance

requirements. Since the new protective function has not been previously reviewed by the NRC, it will require NRC review via an LAR. Accordingly, PG&E Letter DCL-02-115 dated September 24, 2002, for NRC approval, submitted LAR 01-08.

Since LAR 01-08 addresses crediting the automatic actuation of the PORVs to mitigate the consequences of the inadvertent SI at power accident, this LBIE only evaluates the licensing impact of using the alternate LTOP transmitters (PT-403A and PT-405A) to permanently perform the LTOP, 8701/8702 interlock, and the PI-403A/PI-405A functions, the upgrade of the automatic controls for PCV-455C/456 and moving the control function for pressurizer pressure control.

The LBIE has evaluated the permanent use of PT-403A and PT-405A for the LTOP, 8701/8702 interlock and PI-403A/PI-405A indication functions, the upgrade of the PCV-455C/456 automatic actuation circuitry, and moving the control function for pressurizer pressure control. The evaluation concludes these changes do not create/delete any new design functions, alter any licensed design functions, or alter the licensed method of performing a design function. Therefore, these changes do not require prior NRC approval.

02-013 Fuel Handling Building Lighting Replacement

Reference Document No.: A0532867
Rev. No: 0
Reference Document Title: DCP/DCN Fuel Handling Building Lighting Replacement

Activity Description:
The proposed DCP/DCN will replace the 49, 1500-watt incandescent light fixtures in the FHB with 47, 400-watt pulse-start metal-halide light fixtures. The pulse-start metal-halide lamps are ANSI tested for the retention of arc tube materials following non-passive failure, and the fixtures will be of an enclosed design. However, each new lamp will contain approximately 43 mg of mercury, which is a restricted material in the FHB.

Summary of Evaluation:
The design features of the pulse-start metal-halide lamps and enclosed fixtures result in an extremely low risk of mercury entering the fuel pool. The evaluated effects of small amounts of mercury in the pool do not result in an increased risk to the health and safety of the public. The reduced relamping frequency over the pools, from 8-12 weeks to 18-24 months, reduces the risk of normal tools and debris entering the pool. This evaluation concludes that the replacement of the light fixtures in the FHB with pulse-start metal-halide

light fixtures can be implemented without prior NRC approval.

02-019 Procedures for Moving Main Generator Rotor

Reference Document No.: See Activity Description
Rev. No: Various
Reference Document Title: See Activity Description

Activity Description:
The following procedures have been revised or written to control the movement of a main generator rotor out of the turbine building for repair and return.

MP M-22.1, Rev.6	"Generator Rotor Handling" (revised)
TP TA-0201, Rev. 0	"Load Path Procedure for Transporting the Generator Rotor within the Plant Site" (new)
MA1.ID14, Rev. 9	"Plant Crane Operating Restrictions" (revised)

For movement of the rotor inside the turbine building, the procedures use a rigging configuration that is different than that described in DCM T-4, Figure 2.2-1, Rev. 2. DCM T-4 specifies suspending the rotor with wire rope slings (placed around the rotor body) from a lifting beam that is suspended from the main crane hooks of the two turbine building cranes. Instead, the proposed procedures specify suspending the rotor directly from the main hooks with Kevlar slings placed around the bearing journals.

NEI 96-07, section 4.2.1.2, states, "For purposes of 10 CFR 50.59 screening, changes that fundamentally alter (replace) the existing means of performing or controlling design functions should be conservatively treated as adverse and screened in." Therefore, these procedure revisions have been screened in for evaluation under 10 CFR 50.59.

Summary of Evaluation:
The proposed procedures for main generator rotor movement do not create more than a minimal increase in the frequency of accidents or malfunctions previously evaluated in the FSARU. The potential accidents or malfunctions caused by a load drop or other load event (e.g., excessive load swing due to a seismic event) are bounded by accidents previously evaluated in the FSARU, and no new failure modes are created. No new accident or malfunction scenarios are created. The proposed activity has no impact on design basis limits for fission product barriers or on methods of evaluation. Therefore, the proposed procedures do not require prior NRC approval.

02-022 Temporary Use of Fluorescent Drop Lights in Mercury Exclusion Areas

Reference Document No.: CF5.ID13
Rev. No: 2
Reference Document Title: Restrictions of Aluminum and Mercury from Plant Areas

Activity Description:
CF5.ID13 is being revised to allow temporary use of specific fluorescent droplights in areas of the containment, auxiliary building, and fuel handling building, where their use was previously excluded. These areas are defined by CF5.ID13 as Category 1 Restriction Areas and Category 2 Restriction Systems. This authorization is supported by:

- A model-specific evaluation of the proposed fluorescent droplights' ability to resist damage (and subsequent mercury release), and
- A program established (within CF5.ID13) to mitigate and evaluate the potential damage to plant equipment in the event of mercury release in a mercury exclusion area.

Using fluorescent droplights is advantageous because they are more rugged and require less maintenance than incandescent droplights. This is particularly important when using lights in high radiation fields (e.g., SG nozzle dam installation).

Summary of Evaluation:
Mercury can degrade alloy and stainless steels and thus have an adverse affect on safety related SSCs. However, a review of the licensing basis indicates that mercury exclusion in plant areas is a plant level restriction not covered by the FSARU or any other licensing document. A safety assessment has been performed that demonstrates that due to the properties of the proposed fluorescent droplights, the probability of mercury escaping a fixture is low. It further demonstrates that in the unlikely event that mercury does escape the light fixture, the amount of mercury released is insignificant and is not likely to cause degradation of plant SSCs. In addition, the proposed procedure revision includes a program that requires immediate cleanup and evaluation in the event mercury is released in a mercury exclusion area. As a result, there is a negligible probability that mercury from temporary fluorescent droplights will degrade safety related SSCs. Therefore, prior NRC approval is not required.

03-001 Control Room Pressurization System Radiation Monitors

Reference Document No.: ECG 23.7
Rev. No: 0
Reference Document Title: Control Room Pressurization System Radiation Monitors

Activity Description:
New ECG 23.7 establishes conditions, required actions, completion times, and surveillance requirements for the control room pressurization system radiation monitors. These radiation monitors provide control room operators a means to measure the activity at each pressurization intake, should control room pressurization be necessary.

NEI 96-07, Section 4.2.1.2, states, "For purposes of 10 CFR 50.59 screening, changes that fundamentally alter (replace) the existing means of performing or controlling design functions should be conservatively treated as adverse and screened in." Since ECG controls represent a change in the way the radiation monitors are controlled, this activity has been conservatively screened in.

Summary of Evaluation:
Creation of ECG 23.7 does not involve an accident initiator, impact a design basis limit for a fission product barrier, nor does it affect a method of evaluation. The ECG does not impact malfunctions previously evaluated in the FSARU, or create a malfunction with a different result than previously evaluated in the FSARU. Therefore, the proposed change does not require prior NRC approval.

03-003 SG Pressure/Temperature Limits

Reference Document No.: ECG 4.3
Rev. No: 3
Reference Document Title: Steam Generator Pressure/Temperature Limitation

Activity Description:
ECG 4.3 is being revised to incorporate limits for leak testing performed on the secondary side of the SGs. The change will allow the secondary side of the SGs to be pressurized above 200 psig at a minimum temperature of 60°F, subject to a maximum pressure limit of 834 psig for SG 1-1 and 1052 psig for the remaining seven SGs at DCPP. The applicability for these new limits is for the duration of leak tests performed in conjunction with SG maintenance and inspection. No change to the FSARU is required since the minimum temperatures for pressurization of either side of the SG are not specified.

ECG 4.3 in its present form is specific to SG operation, hydrostatic testing, and ISLT. It does not address the SG secondary leak test performed to identify leaking tubes with the primary side open to atmosphere. Protection of the SG against brittle fracture is a design basis requirement; thus, the ECG is to be revised to include such testing. While the original ECG temperature/pressure limits are easily met by primary coolant heating during operation, hydrotesting, and ISLT, similar means for heating the SG structure are unavailable during the secondary leak test.

The original ECG 4.3 temperature/pressure limitations are based on a Code-specified evaluation of SG structural material, Charpy V-notch, or dropweight testing performed at temperatures at least 60°F below the "lower of the vessel hydrotest temperature or the lowest service metal temperature." Because the Charpy tests for DCPP SG materials were performed at 10°F, the original Code permits a minimum metal temperature of 70°F for hydrotest, ISLT, and operation. These requirements are preserved as ECG 4.3.1 in the ECG revision; new limits are set as ECG 4.3.2 only for the SG secondary leak test.

At PG&E's request, Westinghouse provided WCAP-13141, "Technical Basis for Determination of Secondary Side Pressure Test Temperatures for Diablo Canyon Units 1 and 2 Steam Generators," dated December 1991, as the technical basis for the SG secondary side pressure test. The WCAP-13141 analysis, based on linear elastic fracture mechanics and specific to the DCPP SGs, determines allowable pressure limits for a range of temperatures based on the imposed stress intensity factor and fracture toughness in the SG secondary side metal. The methodology is similar to that used in calculation of the primary system pressure/temperature limits with some exceptions, such as the size of the evaluated flaw and the exclusion of thermal stresses. The WCAP-13141 analysis determines a lower bound pressure limit of 60°F for all DCPP SGs, and secondary side pressure limits of 834 psig for SG 1-1, and 1052 psig for the balance. Within these limits, the SG structure is not subject to brittle failure.

The applicability for this change is limited to leak testing during which the SG is out of service and not required to support any TS regarding RCS loop or other SSC operability.

Summary of Evaluation:
The FSARU, SER, SSERs, Westinghouse, and other correspondence were reviewed during the development of this ECG revision. Several instances of the ECG requirements were identified in SSERs 3, 4, and 6, and in NRC Inspection reports IR 86-29 (U1) and IR 86-27 (U2); these are all in the context of referencing the TS that was the predecessor of the ECG. In all cases, the references indicated that the limits were protecting against brittle

fracture of the SGs.

This change does not affect other DCPP licensing bases because the structural integrity of the RCS, including the SGs, is unchanged. The bases for DCPP TSs, as well as other licensing commitments, are maintained.

The 10 CFR 50.59 evaluation concludes that the revised ECG 4.3 pressure and temperature limitations for the secondary leak test provides the required protection of the integrity of the SG structure, including primary and secondary side pressure boundaries. Because both sides of the pressure boundary are maintained, there is no increase in the probability of an accident, a malfunction, the possibility of a new malfunction or accident, or increase in consequences of an accident previously evaluated. Therefore, the proposed change does not require prior NRC approval.

03-006 Use of an SI Pump for Boration in Lieu of a Charging Pump

Reference Document No.: See Activity Description
Rev. No: 1
Reference Document Title: CVCS - Reactivity Control Systems - Boration Systems - Flow Path – Shutdown (ECG 8.5) / CVCS - Reactivity Control Systems - Boration Systems - Charging Pumps – Shutdown (ECG 8.6) / FSARU Section 9.3.4.3.1 & 15.2.4.2 (3)

Activity Description:
ECGs 8.5 and 8.6 are being revised to allow an SI pump to be credited as part of the boration flow path in Mode 6 with the reactor vessel head closure studs fully de-tensioned, in place of a charging pump. This activity screened in because it is a procedure change that fundamentally alters the existing means of performing or controlling a design function. This change is being incorporated into the FSARU.

Summary of Evaluation:
Use of an SI pump as part of the boration flow path will assure that borated water can be injected into the RCS at a rate that assures timely boration of the RCS when required, and will not cause a cold overpressurization event of the RCS in Mode 6, since it will only be used for this purpose when the reactor vessel head closure studs are fully de-tensioned. SI pumps produce sufficient head to borate the reactor core. Use of an SI, pump instead of a charging pump for boration head, will not adversely affect the plant. Therefore, this change does not require prior NRC approval.

03-007 Extension of Turbine Valve Test Interval from Quarterly to Semi-Annually

Reference Document No.: ECG 4.4
Rev. No: 4
Reference Document Title: Instrumentation - Turbine Overspeed Protection

Activity Description:
The proposed change will increase the test interval for turbine valve (high pressure turbine stop and control valves, reheat stop, and intercept valves) testing from quarterly to semi-annually based on the updated probabilistic analysis of turbine missile ejection event reported in WCAP-16054, "Probabilistic Analysis of Reduction in Turbine Valve Test Frequency for Nuclear Plants with Siemens-Westinghouse BB-95/96 Turbines," dated April 2003.

Periodic valve testing requires a temporary power reduction that results in lost electrical generation. In addition, an inadvertent reactor trip can become more likely during the transient power reduction and increase. Therefore, less frequent turbine valve testing results in fewer plant transients.

Summary of Evaluation:
Westinghouse recently re-evaluated the impact of extending the turbine valve test interval beyond the current three months on the annual probability of turbine missile ejection due to overspeed, following the applied basic methodology described in the previously NRC-approved 1987 Westinghouse report WCAP-11525, and using the updated BB-95/96 turbine valve failure rates and system separation frequency.

Based on the review of the results of the new analysis (WCAP-16054) and PG&E's own and industry experience with the BB-95/96 turbine operations, PG&E has concluded that extending the turbine valve test interval to six months will not:

1) Have an adverse effect on the frequency of occurrence of the turbine missile ejection event,
2) Create a new type of turbine valve failure mechanism,
3) Increase the valve failure rates, or
4) Change PG&E's initial conclusion described in FSARU 3.5.2.2.1 that turbine missile ejection is not considered a credible event because of its low probability and there is no radiological consequence involved.

Therefore, the proposed change does not require prior NRC approval.

03-008 Emergency Assessment and Response System

Reference Document No.: ECG 52.4
Rev. No: 0
Reference Document Title: Emergency Assessment and Response System (EARS)

Activity Description:
The EARS system is being placed under ECG control in response to dose assessment problems, and problems with EARS being out of service. PG&E has determined that EARS needs to be placed under the same level of control as the other emergency response facility systems; SPDS (ECG 52.1), ERDS (ECG 52.2), and Data and Recall Recorder Subsystem (ECG 52.3).

This LBIE evaluates implementation of ECG 52.4 for the TSC and EOF EARS stations. The ECG considers each EARS station to consist of one train each of (1) the radiation monitoring system data acquisition system, (2) the meteorological data acquisition system, and (3) the meteorological information and dose assessment system. Action statements, completion times, and surveillance requirements are established for EARS that are consistent with ECG 52.1, 52.2, and 52.3.

Summary of Evaluation:
This activity screened in because the creation of a new ECG is a fundamental change in how an SSC design function is controlled, and it is conservatively being treated as adverse to be evaluated under 10 CFR 50.59.

Implementation of ECG 52.4 will provide more stringent controls on EARS and greater assurance that EARS will be available to perform its emergency assessment and response functions. This new ECG will not adversely affect the plant. Therefore, this change does not require prior NRC approval.

03-010 Change Containment Closure Commitment Under Severe Weather Conditions

Reference Document No.: AD8.DC54
Rev. No: 8
Reference Document Title: Containment Closure

Activity Description:
Procedure AD8.DC54 is being revised to limit the required containment closure in the event of a severe weather warning. The procedure currently requires containment closure of all penetrations that are open directly to the outside atmosphere when a severe weather warning is in effect. The existing procedure is more conservative than the PG&E LAR or the SE provided by

the NRC in support of LA 155/155. The proposed procedural change will limit closure to the equipment hatch, which is consistent with the NRC's basis for issuance of LA 155/155 as documented in the NRC SE.

Procedure AD8.DC54 was originally revised to implement LA 155/155. However, there was an inconsistency within LA 155/155 as to what the severe weather closure requirement was. Per the LA itself, all penetrations open to the outside atmosphere would require closure. However, the PG&E LAR, the current fuel handling accident analysis, or the NRC SE provided in support of LA 155/155, did not support this. The PG&E LAR and the NRC SE only required the equipment hatch to be closed. The technical basis for requiring closure in severe weather is the potential effect of wind-driven missiles. The equipment hatch is the only penetration open directly to the outside atmosphere that could be affected by a wind-driven missile. All other penetrations either open into another building or their exposed area is so small that a missile entering them is not credible.

Summary of Evaluation:
The primary 10 CFR 50.59 criteria affected by the proposed change are the two that address increases in the frequency of an accident and the likelihood of a malfunction previously evaluated in the FSARU. The proposed change only revises what is required to be closed to protect processes and equipment inside containment from external weather during periods when the containment is allowed to be open. The change provides the same level of defense against the potential release of fission products as the previous requirements, and does not increase the frequency of any accident or malfunction. The remaining six 10 CFR 50.59 criteria also answer "No." Therefore, the proposed change does not require prior NRC approval.

03-014 Turbine Control Replacement

Reference Document No.: DCP J-049625
Rev. No: 0
Reference Document Title: DEH/P2000 Turbine Control Replacement

Activity Description:
This design change replaces the existing Westinghouse DEH/P2000 turbine control system with a new system manufactured by Triconex. The existing control system is digital, but has an analog system as a backup. This existing analog system is also used for overspeed protection. The replacement system is completely digital. Since the critical function performed by the turbine control system is to protect the turbine from overspeed, this design change is treated as a digital upgrade, and has been reviewed in accordance with NRC Regulatory Issue Summary 2002-22, Use of EPRI/NEI Joint Task Force Report, "Guideline on Licensing Digital Upgrades: EPRI TR-102348,

Revision 1, NEI 01-01: A Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule."

Summary of Evaluation:
The primary 10 CFR 50.59 criteria affected by the proposed change are the two that address increases in the frequency of an accident and the likelihood of a malfunction previously evaluated in the FSARU. Using the guidance provided by the EPRI/NEI Joint Task Force report, the proposed change does not increase either the frequencies of any accident or the likelihood of any malfunction previously evaluated in the FSARU. The remaining six 10 CFR 50.59 criteria also answer "No." Therefore, the proposed change does not require prior NRC approval.

03-015 Manual Makeup with Reactor Coolant Makeup Control System Impaired

Reference Document No.: OP B-1A: VII (Unit 1)
Rev. No: 27
Reference Document Title: CVCS - Makeup Control System Operation

Activity Description:
In order to perform switch and relay replacements in the reactor coolant makeup control system, power will be removed from the system. This will render the Makeup Mode Control Switch MU43 and Start/Stop Switch MU1, and their associated features, non-functional. A new section has been added to OP B-1A:VII (Unit 1) to permit manual operation of flow control valves and pumps to allow boration and/or dilution while these components are out of service. During this time the automatic functions of the primary makeup water and boric acid flow integrators will be non-functional, flow will not automatically stop when the integrators reach zero, as during normal operation.

Summary of Evaluation:
The only accident that can potentially be initiated in this configuration is the uncontrolled boron dilution accident discussed in FSARU Section 15.2.4. The FSARU describes the opening of the primary makeup water flow control valve as the initiator of the event. The analysis relies on the operator to manually close the valve to terminate such an event, regardless of its initiator, in response to indications and alarms that will still be available to the operator. The frequency of occurrence of this event will not increase since more than a single action is needed to initiate a dilution, the evolutions performed by the system will be under increased administrative controls, and the operator will be sensitized to the absence of the automatic features. Therefore, the proposed change does not require prior NRC approval.