Pacific Gas and Electric Company™

Gregory M. Rueger
Senior Vice President–
Generation and
Chief Nuclear Officer

US Mail:
Mail Code B32
Pacific Gas and Electric Company
PO Box 770000
San Francisco, CA 94177-0001

Overnight Mail:
Mail Code B32
Pacific Gas and Electric Company
77 Beale Street, 32nd Floor
San Francisco, CA 94105-1814

415.973.4684
Fax: 415.973.2313

April 26, 2002

PG&E Letter DCL-02-049

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, 2000, through December 31, 2001

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.59, "Changes, Tests, and Experiments," enclosed is the 10 CFR 50.59 Report for Diablo Canyon Power Plant (DCPP) Units 1 and 2. The report includes a summary of all 10 CFR 50.59 evaluations prepared during the interval January 1, 2000, through December 31, 2001.

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 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 an LBIE was not required for the associated change.

10 CFR 50.59 was revised effective March 13, 2001 as established by Federal Register notice (65 FR 77773), dated December 13, 2000. As discussed in Regulatory Issue Summary 2001-03, dated January 23, 2001, the NRC stated that to permit an orderly transition to the revised rule, licensees were allowed to implement the revised rule later than March 13, 2001. In PG&E letter DCL-01-019 dated February 27, 2001, PG&E stated that the revised rule would be implemented at DCPP by August 1, 2001. Actual implementation occurred on July 17, 2001. LBIEs starting with No. 01-037 were performed under the new rule, except No. 01-039, which was initiated under the old rule and was completed under the provisions of the old rule.

IE47

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

Sincerely,

Gregory M. Rueger
Senior Vice President – Generation and Chief Nuclear Officer

jer1
Enclosure
cc: Diablo Distribution
cc/enc: Ellis W. Merschoff
David L. Proulx
Girija S. Shukla

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

January 1, 2000, through December 31, 2001

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

Acronyms

AFD	axial flux difference
AOT	allowed outage time
AR	Action Request
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASW	auxiliary saltwater
CCP	centrifugal charging pump
CCW	component cooling water
CDF	core damage frequency
CET	core exit thermocouples
CFCU	containment fan cooler unit
CIV	containment isolation valve
CLOF	complete loss of forced coolant flow
COLR	Core Operating Limits Report
CRDM	control rod drive mechanism
CTS	Current Technical Specification
CVCS	chemical and volume control system
CWP	circulating water pump
DBA	design basis accident
dc	direct current
DCM	Design Criteria Memorandum
DCP	design change package
DCPP	Diablo Canyon Power Plant
DE	design earthquake
DEH	digital electrohydraulic
DEI	dose equivalent I-131
DFWCS	digital feedwater control system
DNBR	departure from nucleate boiling ratio
DRPI	digital rod position indication
DWI	Direct Work Item
ECCS	emergency core cooling system
ECG	equipment control guideline
EDG	emergency diesel generator
EFPY	equivalent full power years
EOP	emergency operating procedure
ERDS	Emergency Response Data System
ERFDS	Emergency Response Facility Data System
ESF	engineered safety feature
FHARE	fire hazards Appendix R evaluation

Acronyms (continued)

FSARU	Final Safety Analysis Report Update
FW	feedwater
GL	Generic Letter
HELB	high energy line break
HHSI	high head safety injection
HP	high pressure
Hz	hertz
ICW	intake cooling water
ISI	inservice inspection
ITS	Improved Technical Specifications
LA	license amendment
LANL	Los Alamos National Laboratory
LAR	license amendment request
LBIE	licensing basis impact evaluation
LCO	limiting condition for operation
LCV	level control valve
LHUT	liquid hold-up tank
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LTB	load transient bypass
LTOP	low temperature overpressure protection
LTSP	Long Term Seismic Program
MFPCS	main feedwater pump control system
MFW	main feedwater
MOV	motor-operated valve
MPPH	million pounds per hour
MSIV	main steam isolation valve
MSLB	main steam line break

Acronyms (continued)

N/A	not applicable
NCR	Nonconformance Report
NDE	nondestructive examination
NDT	nil-ductility transition temperature
NFPA	National Fire Protection Association
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NSSSS	nuclear steam supply sample system
ODCM	Offsite Dose Calculation Manual
OP	operating procedure
PAM	post-accident monitoring
PASS	post-accident sampling system
PCD	procedure commitment database
PCP	Process Control Program
PG&E	Pacific Gas and Electric Company
PI	pressure indicator
PM	preventive maintenance
PMT	post maintenance test
PORV	power-operated relief valve
PPC	plant process computer
PPC	pressure pulse cleaning
ppm	parts per million
PRA	probabilistic risk assessment
PSRC	Plant Staff Review Committee
PT	pressure transmitter
P/T	pressure/temperature
PTLR	pressure and temperature limits report
QA	quality assurance
QPTR	quadrant power tilt ratio
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RCSVR	reactor coolant system vacuum refill
RG	Regulatory Guide
RHR	residual heat removal
RSE	reload safety evaluation

Acronyms (continued)

RTD	resistance temperature detector
RVLIS	reactor vessel level instrumentation system
RVRLIS	reactor vessel refueling level indication system
RWST	refueling water storage tank
SAR	Safety Analysis Report
SBLOCA	small break loss-of-coolant accident
SBO	station blackout
SCA	scale conditioning agent
SCMM	subcooling margin monitor
SER	Safety Evaluation Report
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SISIP	Seismically Induced Systems Interaction Program
SJAE	steam jet air ejector
SPDS	safety parameter display system
SQA	software quality assurance
SR	surveillance requirement
SRP	Standard Review Plan
SSC	structures, systems, and components
STP	surveillance test procedure
TES	Technical and Ecological Services
TMS	thermocouple monitoring system
TP	temporary procedure
TS	Technical Specification
TSI	Technical Specification Interpretation
TSP	tube support plate
UPS	uninterruptible power supply
USQ	unreviewed safety question
VCT	volume control tank
WOG	Westinghouse Owner's Group

Index of LBIEs

00-001 Back Up Equipment Clarifications to Equipment Control Guideline 18.5, CO_2 Systems

Reference Document No.: ECG 18.5
Rev. No: 5
Reference Document Title: Fire Protection CO_2 System
Safety Evaluation Description:
ECG 18.5.1 documents the operability requirements for the low pressure CO_2 system and associated subsystems. ECG 18.5.1, Action D.1, requires that when a hose reel station is inoperable, backup fire suppression equipment in the affected area be provided. This change clarifies Action D.1 to state: "verify backup fire suppression equipment is available in the affected area. See Table 18.0-5 for recommended backup equipment." In addition, this change adds a list of recommended backup suppression equipment for CO_2 hose reels. A note was also added to Table 18.0-5 to clarify that for turbine building hose reels, 1 hose reel per elevation, per unit is required for operability. Lastly, another note was added to Table 18.0-5 to clarify that it is only necessary to verify water hose reel operable, except for Nos. 10 & 12 where an extra hose shall also be pre-positioned within one hour.

Safety Evaluation Summary:
ECG 18.5.1 documents the operability requirements for the low-pressure CO_2 system and associated subsystems. ECG 18.5.1 requires, when a CO_2 hose reel subsystem station is inoperable, that backup fire suppression equipment be verified available in the affected area. This revision adds additional information to the ECG regarding the recommended backup hose reel station.

This revision is an ECG clarification only and does not change the intent of ECG 18.5. This revision does not reduce or change conditions or required actions currently specified in ECG 18.5.

Conclusion:
This revision to ECG 18.5.1 provides clarification on the recommended fire suppression backup equipment for specific inoperable hose reel stations. This revision does not reduce the level of fire protection currently provided by the Diablo Canyon Power Plant (DCPP) fire protection program. This revision provides clarification to ECG 18.5.1 so that operations can quickly and easily identify the recommended backup equipment for inoperable hose reel stations. Therefore, the DCPP fire protection program and safe shutdown capability is not adversely affected by this change. This activity does not result in an unreviewed safety question (USQ).

00-003 Gaps and Potentially Degraded Penetration Seals in Appendix-A Fire Rated Boundaries

Reference Document No.: FHARE 135
Rev. No: 1
Reference Document Title: Gaps in Appendix-A Fire Rated Boundaries
Safety Evaluation Description:
Fire Hazards Appendix R Evaluation (FHARE) 135, Revision 1 evaluates the acceptability of potentially degraded penetration seals in fire barriers separating 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 smoke detection, automatic CO_2 suppression systems, automatic wet pipe sprinklers, hose stations and portable fire extinguishers, provide assurance that the potentially degraded penetration seals 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:
The penetration seals evaluated by FHARE 135, Revision 1, are located in fire barriers classified as Appendix A to Branch Technical Position (BTP) APCSB 9.5-1 barriers. The fire areas (15 &18), where the high fire loading is contained, do not contain safe-shutdown circuits or equipment.

Based on the available fire-protection features in the affected areas, the additional concern for potentially degraded penetration seals 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 potentially degraded penetration seal configurations. This activity does not result in a USQ.

00-004 Review for ECGs that Incorporate NRC Approved Changes

Reference Document No.: OP1.DC16, FSARU Sect. 16.1
Rev. No: 6 and 12 respectively
Reference Document Title: "Control of Plant Equipment Not Required by the Technical Specifications", and "Technical Specifications and Equipment Control Guidelines"
Safety Evaluation Description:
OP1.DC16 is being revised to eliminate 10 CFR 50.59 and Plant Staff Review Committee (PSRC) reviews of ECG changes that have already been

approved by the Nuclear Regulatory Commission (NRC) (such as the relocation of a Technical Specification (TS) to the ECG program).

Safety Evaluation Summary:
New ECGs or revisions to existing ECGs that implement changes previously approved by the NRC should not require a 10 CFR 50.59 safety evaluation, and consequently not require PSRC review. Since the changes have already been approved by the NRC, they will screen out in the Licensing Basis Impact Evaluation (LBIE) screen process because they do not affect the Safety Analysis Report (SAR) since the license amendment (LA) would be considered part of the SAR. Also, they will not result in USQs in the LBIE review since they have already received NRC review. The PSRC review is unnecessary since the PSRC will have reviewed the license amendment request (LAR), or other licensing submittal, that resulted in the NRC-approved changes.

Conclusion:
Deleting the 10 CFR 50.59 and PSRC reviews of ECG changes that have already been approved by the NRC eliminates unnecessary reviews. These deletions do not affect the NRC approval process, or the requirements of the 10 CFR 50.59 safety evaluation process, or the authority of the PSRC to review plant changes. The deletions only affect reviews that are superfluous to the reviews already required to obtain the NRC approved changes. The deletions do not result in a USQ.

00-005 Remove Abandoned Makeup Water Vacuum Deaerating System

Reference Document No.: Design Change Package (DCP) M-049500
Rev. No: 0
Reference Document Title: Remove Abandoned Makeup Water Vacuum Deaerating System

Safety Evaluation Description:
This change removes the abandoned makeup water vacuum deaerating system to provide additional space that Maintenance would like to have available. The change also cuts and caps the auxiliary steam lines that supply this abandoned system. A new pipe support will be required to support the cut steam lines.

Safety Evaluation Summary:
The makeup water deaerating system is currently abandoned in place and is listed in the Final Safety Analysis Report Update (FSARU) as abandoned in place. This equipment is being removed to free up space in the hot shop that Maintenance would like to have available. This change deletes the references to the makeup water vacuum deaerating system and its associated

equipment from the FSARU.

Conclusion:
This abandoned, design-class II equipment has no accident initiating or mitigating functions and is not described in the TS or their bases, therefore this change does not result in a USQ.

00-008 FSARU 5.2.1.5, Table 5.2-4 and Table 5.2-4A

Reference Document No.: FSARU
Rev. No: 12
Reference Document Title: FSARU 5.2.1.5, Table 5.2-4 and Table 5.2-4A
Safety Evaluation Description:
To implement Improved Technical Specifications (ITS), Current Technical Specification (CTS) 5.7.1 & Table 5.7-1 were moved to the FSARU verbatim in Section 5.2.1.5 and as new Table 5.2-4A in a previous FSARU change. The information contained in new FSARU Table 5.2-4A conflicts with information in existing FSARU Table 5.2-4 and/or does not represent correct information. This change reconciles the differences and merges the tables into a single table. Also, the FSARU text of 5.2.1.5 is modified to more closely implement the intent of ITS 5.5.5, in particular, regarding the distinction between design limit and number of cycles/occurrences.

Safety Evaluation Summary:
Cyclic or transient limitations on the number of occurrences have an ultimate design limit of maintaining the Cumulative Fatigue Usage Factor (the combined effect of all cycle/transients) to less than 1.0 as defined by the American Society of Mechanical Engineers (ASME) Code. There are also Code limits on the developed stresses. Stresses associated with each design cycle are determined based upon parameters [e.g., reactor coolant system (RCS) cooldown rate at 100°F/hr from 550°F to ambient) and the number of cycles established by the Westinghouse Equipment Specification (hence referred to as "design assumptions" and "design cycle"]. These parameters represent conservative assumptions to bound expected plant transient conditions (e.g., RCS cooldown at 80°F/hr from 550°F to 300°F). The actual transient (or partial cycle) creates less stress than the design cycle, yet is counted as one full design cycle. Thus, neither the design assumptions nor the design cycles are necessarily absolute limits, so long as the developed stress and cumulative fatigue usage factor are less than Code allowables – design limit.

ITS 5.5.5 assures that "[a] program provides controls to track the FSARU, Section 5.2 and 5.3, cyclic and transient occurrences to ensure that components are maintained within the design limits." Surveillance Test Procedure (STP) M-55 already exists to track the Table 5.2-4A

(CTS Table 5.7-1) occurrences. Most of the detail of what constitutes an "occurrence" / "cycle" in accordance with Table 5.2-4A is already contained in the FSARU Section 5.2.1.5 discussion of the transient, and is a level of detail that isn't necessary for FSARU Table 5.2-4, or is best located in the program document. Therefore, Table 5.2-4A is deleted.

The text of FSARU Section 5.2.1.5 is modified to delete the wording added by previous relocation of CTS 5.7.1 and replace it with a discussion regarding "design limit" and a program to ensure that the limit is not exceeded to make it more consistent with ITS 5.5.5.

Conclusion:
No USQ is created by this change since:

1. Current licensing bases (CTS Table 5.7-1) and the design-bases supports 12 occurrences of the Inadvertent Auxiliary Spray Actuation transient.
2. The Loss of Load, without immediate turbine or reactor trip, is a more severe transient than a Loss of Load, with turbine trip but without immediate reactor trip transient, as stated in FSARU sections 5.2.1.5.2 and 15.2.7, and was the transient used in accordance with the "design assumptions". Therefore, the cumulative fatigue usage factor remains below the design limit of 1.0.
3. All editorial changes are consistent with the design limit and its assumptions.
4. The FSARU consistently references allowable cumulative fatigue usage factor as Code limit of 1.0. The Safety Evaluation Report (SER) accepts reactor coolant pressure boundary components design loading combinations design limits as "comparable to the criteria recommended in Regulatory Guide (RG) 1.48", which cycles back through the ASME Code to a cumulative fatigue usage factor limit of 1.0.
5. Replacing relocated CTS 5.7.1 with wording more similar to ITS 5.5.5 still assures that the design limit is met.

00-009 Unit 2 Cycle 10 Final Reload Safety Evaluation and COLR Revision 2

Reference Document No.: COLR 2-10
Rev. No: 1
Reference Document Title: Diablo Canyon Unit 2 Cycle 10 Final Reload Safety Evaluation and COLR Revision 1

Safety Evaluation Description:
This revised reload safety evaluation (RSE) documents the NRC approval of LA 136. The amendment allows the use of WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code" to determine core operation limits. This revises the K(z) curve (Figure 7) in the core operating limits report (COLR) for each unit. The new K(z) curve

provides for more operation flexibility than the previous curve in the top portion of the core. This activity is being performed to implement the TS within the 90 day time limit from the NRC SER.

Safety Evaluation Summary:
The implementation of the flat K(z) curve in the TS allows for more operating flexibility during the processing of the monthly flux maps. Since this change does not affect plant operation, or physiclally alter or change the function of structures, systems, or components (SSCs) required to mitigate the consequences of a design-basis accident, the RSE remains valid and unchanged.

Conclusion:
With no change to any systems at the plant and NRC review of the new inputs to the licensing basis under the 10 CFR 50.46 for the small break loss of coolant accident (SBLOCA) analysis, the new K(z) curve in the TS can be implemented. The 50.59 evaluation shows that the RSE is still applicable and no reduction in the margin of safety exists. This activity does not result in a USQ.

00-011 Clarify FSARU Section 6.2.4.1 Regarding Instrument Lines Penetrating Containment

Reference Document No.: FSARU Change Request
Rev. No: 0
Reference Document Title: Clarify FSARU Section 6.2.4.1 Regarding Instrument Lines Penetrating Containment

Safety Evaluation Description:
This change involves a FSARU change to accurately describe the features that provide required protection of containment isolation functions associated with the containment pressure instruments, which includes their closed double-barrier design (already described in the FSARU) and their physical separation and shielding from the effects of missiles, pipe whip, and jet impingement, as afforded by their location. Currently, both the bellows and tubing inside containment, and transmitter diaphragm and tubing outside containment, are described as "enclosed by protective shielding." Since the bellows and transmitter diaphragms are installed in housings provided by manufacturer standard practice, this is not considered an extraordinary feature, for these components. The term "protective shielding" is considered to refer to armored tubing as shown on obsolete drawing 663230-181; this tubing had been previously replaced with stainless steel tubing without armor in accordance with DCPP standard installation practices, in accordance with DCP J-049102 and J-050102, circa 1995.

The statement from the FSARU Section 6.2.4.1.3, "Both the bellows and tubing inside containment and transmitter diaphragm and tubing outside containment are enclosed by protective shielding" has been modified as shown on the FSARU mark-up.

Safety Evaluation Summary:
Protection against missiles and the dynamic effects of pipe rupture (pipe whip/ jet impingement) is provided by virtue of the location of these pressure transmitter (PT) instrument components, and not because these components are "enclosed by protective shielding" as currently described. Missile protection is provided by the crane wall and other credited missile shields inside containment. Also, for both inside and outside containment, these PT components are located outside of the zone of influence of currently postulated high energy line break (HELB) pipe whip/jet impingement, and are therefore separated from HELB dynamic effects hazards. The tubing armor that was previously removed has not been credited as a barrier or shield against these effects, based on a review of HELB documentation, nor would it have been, based on DCPP design-basis requirements of Design Criteria Memorandum (DCM) T-12.

The containment PTs and associated tubing sensing lines meet all the licensing requirements to satisfy the instrumentation and containment isolation requirements.

This change has no adverse impact on DCPP licensing commitments and requirements and does not impact plant operation or operator actions in any way.

Conclusion:
This change does not degrade the protection that is provided these PT instruments against the effects of missiles and the dynamic effects of pipe rupture. The change accurately describes the protection that is provided. This change has no adverse impact on DCPP licensing commitments and requirements, and therefore, this change does not result in a USQ.

00-012 Unit 1 Cycle 10 Final Reload Safety Evaluation and COLR Revision 1

Reference Document No.: COLR 1-10
Rev. No: 1
Reference Document Title: Unit 1 Cycle 10 Final Reload Safety Evaluation and COLR Revision 1
Safety Evaluation Description:
This revised RSE documents the NRC approval of LA 136. The amendment allows the use of WCAP-10054-P-A, "Westinghouse Small Break ECCS

Evaluation Model Using NOTRUMP Code," to determine core operation limits. This revises the K(z) curve (Figure 7) in the COLR for each unit. The new K(z) curve provides for more operation flexibility than the previous curve in the top portion of the core. This activity is being performed to implement the TS within the 90 day time limit from the NRC SER.

Safety Evaluation Summary:
The implementation of the flat K(z) curve in the TS allows for more operating flexibility during the processing of the monthly flux maps. Since this change does not affect plant operation, or physiclally alter or change the function of SSCs required to mitigate the consequences of a design-basis accident, the RSE remains valid and unchanged.

Conclusion:
With no change to any systems at the plant and NRC review of the new inputs to the licensing basis under the 10 CFR 50.46 for the SBLOCA analysis, the new K(z) curve in the TS can be implemented. The 50.59 evaluation shows that the RSE is still applicable and no reduction in the margin of safety exists. This activity does not result in a USQ.

00-013 Special Process "Nondestructive Examination Procedures" Responsibility Transfer

Reference Document No.: FSARU 13.1.1.2.1 and 17.1
Rev. No: 13
Reference Document Title: FSARU
Safety Evaluation Description:
FSARU Chapters 13 and 17 specifically charge the Manager, Technical and Ecological Services (TES) with the responsibility for development, evaluation, qualification, testing, and improvement of nondestructive-examination (NDE) procedures required by the company and with evaluation of these procedures used at DCPP by other organizations. This responsibility is transferred to the Manager, Engineering Services as a result of the TES/ Nuclear Power Generation (NPG) value-improvement project recommendations.

Safety Evaluation Summary:
The transfer of responsibility for development, evaluation, qualification, testing, and improvement of NDE procedures does not constitute a change to, nor a reduction in the breadth of the Quality Assurance (QA) program. The requirements for development and qualification specific to the NDE procedures portion of "Special Processes" will carry forward, while the procedure review and approval process will be controlled by DCPP plant procedures.

Conclusion:
Transfer of managerial responsibility for the NDE procedures portion of "Special Processes" does not reduce QA program commitments. NDE procedures are subject to the same QA, code, and regulatory requirements after the transfer. Since controls related to the quality program remain intact, this change does not result in a USQ.

00-014 Recirculation Sump Screen Modification

Reference Document No.: DCP N-49510
Rev. No: 0
Reference Document Title: Recirculation Sump Screen Modification
Safety Evaluation Description:
DCP N-49510 provides the design to modify the recirculation sump screen and related structures to substantially increase the available surface screen area. This modification will remove the existing inclined grating and associated 1/8 inch x1/8 inch stainless steel mesh, remove a major portion of the weir wall downstream of the inclined grating and remove the 6 inch high curb downstream of the weir wall. These components will be replaced with separate elements consisting of a 6 inch debris curb, a stainless steel grating trash rack, and an extended-surface sump screen fabricated from stainless steel plate perforated with 1/8 inch diameter holes. The new design affords a significant increase in the available screen area. In addition, the new design has design features that are recommended in RG 1.82.

Safety Evaluation Summary:
Ongoing industry evaluations of emergency core cooling system (ECCS) sump screen blockage due to loss-of-coolant accident (LOCA) debris, including fibrous materials, Min-K insulation, paint debris, insulation vapor barrier paper, and fire barrier material, have resulted in a net reduction of DCPP's sump screen head-loss margin. Industry evaluations are expected to continue. Los Alamos National Laboratory (LANL) is supporting the NRC in the resolution of Generic Safety Issue, GSI-191, "Assessment Of Debris Accumulation On PWR Sump Performance." LANL has been tasked to develop a methodology for estimating debris generation and debris transport in PWR containments. The outcome of these actions could have potential adverse impact on DCPP sump calculations and margin. Reconfiguration of the sump screen will significantly increase the available sump screen area.

The new design of the recirculation sump screen incorporates design elements specifically recommended in RG 1.82. In addition, the extended-surface perforated -plate screen surface provides a significant increase in the available screen area. The new sump screen configuration will perform the same design functions as the existing recirculation sump screen configuration to: (1) provide sufficient surface screen area to assure adequate net positive suction head (NPSH) is afforded the ECCS pumps during the recirculation phase of a design-basis LOCA, (2) minimize the effects of air ingestion, and (3) minimize the amount of debris ingested into the ECCS.

Conclusion:
The licensing-basis review of the changes to the configuration of the recirculation sump screen demonstrate that the recirculation sump remains operable during the design-basis events defined in the FSARU. The ECCS system component design for reliability, redundancy, and operation within design and safety limits is not affected by this change. No events that could impact the health and safety of the public are determined to be created by the change in the configuration of the recirculation sump screen. This activity does not result in a USQ.

00-015 Replace Feedwater Pump Speed Control System, Unit 1

Reference Document No.: J-049419

Rev. No: 0

Reference Document Title: Replace Feedwater Pump Speed Control System, Unit 1

Safety Evaluation Description:
This change replaces the existing hybrid analog/digital main feedwater pump control system (MFPCS) 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 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 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 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 input/output (I/O) is also triple-redundant. Failure of a single I/O module, channel, or processor will not cause loss of control.

The upgrade includes linear voltage differential transformers (LVDTs) 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 (MFW) system that comprise the feedwater isolation 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 safety feature (ESF) systems or components associated with detection or mitigation of events associated with the feedwater isolation 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. This activity does not result in a USQ.

00-016 Assessment of Changes in Post-LOCA Dose Apportionment

Reference Document No.: Calculations STA-087 and STA-090
Rev. No: 0
Reference Document Title: "Post-LBLOCA Doses with Cont. Spray Delay time 86.5 sec" and "Margin Leakage Rate From Post-LOCA Recirculation Loop Components Outside Containment"

Safety Evaluation Description:
In Action Request (AR) A0449539, it was identified that there was an error in the assumed spray delay time used to calculate the offsite doses due to post-LOCA containment leakage. This change corrects the doses reported due to containment leakage (increased) and allowable post-LOCA recirculation

leakage (decreased by the same amount) doses as reported in Chapter 15.

Safety Evaluation Summary:
In AR A0449539, it was identified that the containment leakage dose analysis used a spray delay time of 80 seconds instead of the required 86.5 seconds. This error had a small effect on the overall calculated thyroid dose from containment leakage, since the later spray start time resulted in a slightly later start for iodine removal from the containment atmosphere. This resulted in a small increase in the calculated thyroid doses from leakage out of containment. In order to compensate for this increase, a reduction in the allowed leakage from post-LOCA recirculation fluids was also determined.

A review of the current licensing basis for post-LOCA doses concluded that the total dose consequences reported in the FSARU included allowable levels of leakage which raised the dose consequences of a LOCA to the regulatory limits established in 10 CFR 100 and General Design Criteria 19.

Conclusion:
The error reported in AR A0449539 was evaluated. The effects of the change result in an increase in one contributor to the post-LOCA dose reported in the FSARU, however the effects of this increase have been found to not increase the total reported post-LOCA dose since there is sufficient creditable margin in the allowable leakage dose values reported in the FSARU. Therefore, there is no impact to the licensing basis due to this error. This activity does not result in a USQ.

00-017 FSARU Review of Telecommunications Equipment Descriptions

Reference Document No.: FSARU
Rev. No: 12
Reference Document Title: FSARU
Safety Evaluation Description:
Section 8.3.1.5.1 (page 8.3-32):
Revised the description of communication circuits to accurately reflect the circuits/conductors used. Removed unnecessary detail regarding the color coding of jackets.
Section 9.5.1.2.11.1 (page 9.5-13):
Revised the section to indicate that the public address system is not installed throughout the entire plant.
Section 9.5.1.2.11.1 (page 9.5-14):
Revised this section to clarify that the "in-plant" radio system consists of the operations and security radio frequencies, not the health physics frequency. Clarified that the system no longer consists of radio-activated pagers

(replaced by the in-plant paging system) and added base radios and control consoles for clarification. Also clarified that the system is a half duplex and not a full duplex system.
Section 9.5.2.2.1 (page 9.5-15):
Revised section to clarify that not all phones in the Pacific Gas and Electric (PG&E) system can dial in fire alarms at the plant. Clarified to state that all "plant" telephones have this function.
Table B-1 (page 9.5B-25):
Clarified that the equipment in the control room is a control console and not a base station radio.
Clarified that the power for the control consoles is from vital power and not from the station batteries. Clarified that the equipment at the hot shutdown panel is a control point and not a base station radio.

Safety Evaluation Summary:
The changes identified have no impact on plant operations or operator actions. The current public address system has adequate coverage to hear plant announcements. The in-plant radio system for operations has not changed; the only clarification was that the health physics radio is not part of the "in-plant" system. The radio-activated pagers were replaced in the 1987-8 time frame with the in-plant paging system. The radios were clarified as being half duplex, which provides the necessary functionality for their service. The power supply change is conservative as the control room control console is powered from vital power as are the hot shutdown panel control points which are additionally backed up by 12 V batteries.

Conclusion:
The changes identified are acceptable and do not constitute a USQ.

00-018 Fire Barrier Penetration Seal Inspection Program Changes
Reference Document No.: ECG 18.7 and STP M-70A
Rev. No: 3 and 4, respectively
Reference Document Title: Fire Rated Assemblies & Inspection of Fire Barrier Penetration Seals
Safety Evaluation Description:
This evaluation discusses proposed changes to the penetration seal inspection program and the bases for those changes. The changes to the penetration seal inspection program are summarized as follows:

1. Performing a visual inspection of 10 percent of the penetration seals, on average, every 18 months, instead of inspecting 100 percent of the seals every 18 months;

2. Reducing the population of penetration seals that require inspection in accordance with STP M-70A;

3. Establishing a process for responding to cases where a degraded seal is identified. This includes: compensatory actions, actions to be taken to determine the cause of the degradation, the scope of the problem and a solution.

4. This evaluation provides the bases for declaring a small population (approximately 23 percent) of ECG 18.7 fire barrier penetration seals operable without current STP M-70A inspections as currently required by ECG 18.7, consistent with the philosophy of a 10 percent sampling program.

Safety Evaluation Summary:
Fire barrier penetration seals are a passive fire protection feature. Because they are passive components, they are highly reliable.

In 1994, all of the DCPP fire-barrier penetration seals were declared inoperable when some of the silicone foam seals were found to not satisfy the design requirements for qualified seals. As such, it was decided that these seals should not be relied upon to fully perform their intended design function. Further, it was concluded that these types of seals were installed in various fire barriers in both Units (1&2). Since the scope of the problem could not be defined at that time, all of the fire-barrier penetration seals were declared inoperable, with compensatory measures established in accordance with ECG 18.7.

Subsequently, a programmatic response was initiated by DCPP to resolve the issues associated with degraded penetration seals. The penetration seal program is now nearing completion, with all of the original scope completed. Since the seal program is transitioning back to a normal maintenance program, from a design verification and repair/replace program, it is necessary to re-evaluate the scope and process used by the penetration seal inspection program.

The basis for changing from a 100 percent surveillance program is summarized below:

- Each seal has had a minimum of two visual inspections performed during the penetration seal re-verification program; which began in 1995.

- Design controls have been enhanced to ensure configuration control.

- Penetration seals are designed for a 40 year life, they are passive components, and are not susceptible to degradation without influence from some type of outside destructive force. A periodic surveillance program is in place to identify degraded seals. This process includes a method to evaluate the cause of a degradation, the scope of the problem, and a solution. The existence of a well defined surveillance program, along with the physical characteristics of the seals, provide the basis for expanding the complete inspection cycle to once every 15 years. (10 percent per 18 months)

- The original Westinghouse Standard TS (NRC NUREG 0452, Revision 5) allowed for a 10 percent penetration seal sampling program.

- The DCPP fire protection licensing basis was reviewed, and no impact to the license basis was identified. Also, this change does not impact any fire protection program commitments.

- A survey of industry operating experience indicates that other plants were originally licensed to, or subsequently changed to, a 10 percent sampling program.

- 100 percent inspection of fire barriers (STP M-70D) have recently been performed. During the course of inspecting fire barriers, an inspector briefly views the material condition of all visible penetration seals. No degraded penetration seals were identified during these inspections.

Due to the length of the penetration seal re-verification project, approximately 5 years, a small population of seals has not been inspected within the required surveillance time period (18 months) specified by ECG 18.7. Therefore, this evaluation concludes that it is acceptable to return penetration seals to a functional status even if the seal has not been visually inspected within the past 18 months.

Conclusion:
Fire protection engineering has determined that it is acceptable to transition to a 10 percent penetration seal sampling program and to return penetration seals to a functional status even if the seal has not been visually inspected within the past 18 months.

The above changes have been evaluated and do not adversely impact the

DCPP fire protection program or the ability of the plant to safely shutdown. This activity does not result in a USQ.

00-019 Revision 1 to ECG 17.1

Reference Document No.: ECG 17.1
Rev. No: 1
Reference Document Title: Auxiliary Salt Water Cross-Tie Valve FCV-601
Safety Evaluation Description:
This revision clarifies ECG 17.1 to reflect that the ECG applies only to FCV-601 and that FCV-601 operability is not driven by auxiliary salt water (ASW) train operability. The wording of Required Action statement B.1.2 is clarified to agree with the commitment PG&E made in response to Generic Letter (GL) 91-13 to perform an operability evaluation in the event FCV-601 is unavailable for greater than 72 hours. Also the Bases have been reformatted to use the ITS format and additional licensing information to make the Bases more useful, and STP references have been removed. All the changes are administrative in nature.

Safety Evaluation Summary:
This revision clarifies ECG 17.1 to reflect that it applies only to FCV-601 and that FCV-601 operability is not driven by ASW train operability. ASW train operability is assessed and tracked separately in accordance with ITS 3.7.8. In response to GL 91-13, PG&E committed to implement an ECG for FCV-601 to assure availability of the valve. Restricting the ECG to address only FCV-601 operability is consistent with the commitments made for assuring operability of FCV-601.

Also the ECG Bases have been reformatted in the ITS format and additional licensing information has been added to make the Bases more useful, and STP references have been removed.

Conclusion:
The proposed changes are administrative in nature and do not result in a USQ.

00-020 Revision 1 to ECG 38.1

Reference Document No.: ECG 38.1
Rev. No: 1
Reference Document Title: Reactor Trip System (RTS) Instrument Response Times
Safety Evaluation Description:

Changes are proposed to ECG 38.1 to: (1) clarify its purpose, (2) incorporate results of an evaluation of Westinghouse Technical Advisory 98-11, "Acceptance Criteria For Time Response Testing," (3) change the description of Functional Unit 10 to be consistent with ITS and the FSARU, and (4) incorporate TS Bases Change 2000-005, dated February 22, 2000. Specifically:

1. Background information has been added to the Bases to explain the purpose of the ECG.
2. Footnote 4 has been added to Functional Units 6 and 7 to specify that the response-time limit for the resistance temperature detector (RTD) sensors is 5 seconds. The overall response-time limit of ≤ 7 seconds for this function, as required by ECG Table 38.1-1, is not changed.
3. The two substeps for single- and two-loop trips for Functional Unit 10 have been deleted to be consistent with ITS Table 3.3.1-1 and FSARU Table 15.1-2. No surveillance test requirements are changed by this change.
4. The words "exclusive of the source range preamplifiers" have been added to Table 38.1-1, Footnote (1), as approved by TS Bases Change 2000-005, dated February 22, 2000.

Safety Evaluation Summary:
Changes (1) through (3) are administrative in nature and do not change any of the technical requirements of the ECG.

Change (4) is a technical change that was approved under TS Bases Change 2000-005, dated February 22, 2000. The justification for excluding the source range preamplifiers from response-time testing is that they do not have any failure mechanisms that could affect response times that would not be detected during routine testing. This justification is consistent with the guidance of IEEE 338-1977.

Conclusion:
The changes proposed for Revision 1 to ECG 38.1 are consistent with LAs 135/135, the FSARU, and TS Bases Change 2000-005. The proposed changes do not result in a USQ.

00-021 Revision 1 to ECG 38.2

Reference Document No.: ECG 38.2

Rev. No: 1
Reference Document Title: Engineered Safety Feature (ESF) Response Times
Safety Evaluation Description:

1. A paragraph has been added to the Bases explaining the purpose of the ECG.

2. ECG Table 38.2-1 has been reformatted to match ITS Table 3.3.2-1 except as noted below:

   a. Reactor trip has not been included in Functional Unit 1 - Safety Injection (SI). This testing is to be deleted in conjunction with ITS implementation.

   b. Feedwater isolation time response is included under Functional Unit 5 rather than Functional Unit 1. The 63 seconds listed is for feedwater isolation from SI, regardless of what initiated the SI.

   c. Phase A isolation time has been included under Functional Unit 3 rather that Functional Unit 1 to provide clear distinction of the three initiating signals for Phase A isolation.

   d. Containment ventilation isolation (CVI) has not been listed under Functional Unit 1. There is no response time requirement for CVI from SI (it is listed as "N.A." in Revision 0 of the ECG). The required response test from high radiation is already included under Functional Unit 9.

   e. Auxiliary feedwater response time has been included under Functional Unit 6 instead of Functional Unit 1. The 60 seconds response time is required for auxiliary feedwater start (motor driven) from SI, regardless of what initiated the SI.

   f. Component cooling water (CCW), containment fan cooler unit (CFCU), and ASW times have been placed in new Functional Units 10, 11, and 12, rather than in Functional Unit 1. These functions are not listed in ITS Table 3.3.2-1 but are required to be response tested from SI. The times listed are from any SI, regardless of the initiator, so need only be listed once.

3. PSRC Interpretation 97-04 has been incorporated into Functional Unit 5 by adding the main feedwater pump trip to 5c.-Safety Injection as approved by LAs 140 (Unit 1) and 140 (Unit 2), dated

February 22, 2000.

4. The description and initiator of Functional Unit 9 have been clarified.

5. Note 9 has been added to clarify functions that were not in the original TS Table 3.3-5.

Safety Evaluation Summary:
Changes are proposed to ECG 38.2 to: (1) clarify its purpose, (2) change the ECG Table 38.2-1 format to match the format of ITS Table 3.3.2-1 except as noted above in the description of changes, (3) delete reactor trip from safety injection time response in Functional Unit 1, (4) incorporate PSRC Interpretation 97-04 into Functional Unit 5, (5) clarify the description of Functional Unit 9, and (6) add note 9. The proposed changes are either administrative or have been previously approved by the NRC.

Conclusion:
The changes proposed for Revision 1 to ECG 38.2 are consistent with LAs 135/135, LAs 140/140, and the FSARU. The proposed changes are either administrative or have been previously approved by the NRC and do not result in a USQ.

00-022 Re-Categorization of Diesel Fuel Oil Transfer Valves and Day Tanks

Reference Document No.: FSARU/TS Bases
Rev. No: 13
Reference Document Title: FSARU 8.3.1.1.13.8, 9.5.4, Figure 3.2-21 and TS Bases 3.8.1.1

Safety Evaluation Description:
This change revises FSARU Sections 8.3.1.1.13.8, 9.5.4, Figure 3-21, and TS Bases 3.8.1.1 to change the description of the diesel fuel oil transfer system to categorize the components downstream of, and including, the level control valve (LCV) manual isolation valves as part of the associated emergency diesel generator (EDG) system.

The day tanks are described in the FSARU and the current TS Bases as part of the fuel oil transfer system. By inference, everything upstream of the day tanks, including the LCVs, would also be considered as part of the fuel oil transfer system in the FSARU and TS Bases descriptions.

Safety Evaluation Summary:

TS 3.8.1.1 includes operability requirements for the diesel fuel oil transfer system. TS 3.8.1.1, Action G, allows one fuel oil train to be inoperable for up to 72 hours. However, if both trains are inoperable, at least one train must be restored to operable status within one hour or a shutdown is required.

The diesel fuel oil transfer system consists of two 50,000 gallon underground storage tanks and two trains of piping and pumps. Each train of the transfer system connects to all six EDGs and is capable of supplying the required fuel oil for all six EDGs. Each EDG is supplied fuel oil from an engine-mounted day tank. A manual isolation valve and an air operated LCV separate each train of the transfer system from the day tanks.

The LCVs have two functions. One function is to regulate level in the associated EDG day tank. The other function is to close to isolate the two trains of the diesel fuel oil transfer system should one train fail. Each LCV is supplied motive air from a separate air receiver associated with the EDG to which it supplies fuel oil. If the air receiver is inoperable, the associated valve would be inoperable. The failure mode of the LCVs is such that if air pressure in the air receivers is lost, the LCVs will close. Based on the current boundaries of the diesel fuel oil transfer system, this would make both trains of the fuel oil transfer system inoperable.

However, the failure of an air receiver, and its subsequent effect on a valve, will only impact the one EDG to which the valves supply fuel oil, since the valves will close and isolate both trains of fuel oil from the EDG and each other. Since the failure of both LCVs to the same EDG cannot affect any other EDG, it should not render both trains of the diesel fuel oil transfer system inoperable.

A review of the licensing basis and discussions with other utilities was conducted to determine NRC acceptance of the diesel fuel oil transfer system boundaries to determine if the components downstream of, and including, LCV manual isolation valves can be categorized as part of the EDG system rather than the diesel fuel oil transfer system.

Conclusion:
Based on the information on the diesel fuel oil transfer system design from other plants, it appears that the DCPP FSARU and Standard Review Plan (SRP) definition of the fuel oil transfer system are based on independent transfer systems for each EDG and the categorization of the LCVs as part of the fuel oil transfer system is inappropriate. Consequently, it is appropriate to categorize the components downstream of, and including, the LCV manual isolation valves as part of the associated EDG system. This activity does not result in a USQ.

00-023 Lesser-Rated Penetration Seals in Selected Barriers

Reference Document No.: FHARE 142
Rev. No: 0
Reference Document Title: Acceptance Criteria for Penetration Seals in Selected Barriers

Safety Evaluation Description:
FHARE 142 Rev. 0 evaluates the acceptability of not inspecting six penetration seals in barriers between the Unit 1 and Unit 2 12 kV switchgear rooms and the outside.

Safety Evaluation Summary:
There are six seals located in 12 kV switchgear rooms that can not be inspected because they are covered by steel angles on the outside and 3M wrap on the inside. FHARE 142, Rev. 0, evaluates these seals and concludes that not inspecting these seals will not have an adverse impact on the DCPP Fire Protection Program.

Conclusion:
FHARE 142 concludes that not inspecting the six penetration seals will not create a potential for a fire to spread from one area to another. Therefore, the seals evaluated by FHARE 142 do not adversely impact the DCPP Fire Protection Program. This activity does not result in a USQ.

00-024 Reactor Vessel and Internals Dynamic Analyses

Reference Document No.: FSARU Chapter 3
Rev. No: 12
Reference Document Title: DCPP Updated Final Safety Analysis Report

Safety Evaluation Description:
The seismic and LOCA dynamic analysis for the reactor vessel and internals components has been revised to account for the Long Term Seismic Program (LTSP) spectra.

Safety Evaluation Summary:
The Westinghouse analyses use the same models for seismic, LOCA and combination (note the Hosgri earthquake and LOCA are not considered to occur simultaneously in the FSARU) as previously used. The same inputs regarding deflection limits and blowdown time are used. The additional spectra for LTSP are applied to the model and resultant loads are combined with the LOCA loads using square-root-of-the-sum-of-squares (SRSS). The results are that the reactor and internals loads are bounded by the allowable loads stipulated in the Generic 4 Loop Stress Report in which generic reactor and internals component imposed loads are below ASME Code stress

allowables.

Conclusion:
The Westinghouse analyses utilize NRC-approved computer programs for dynamic analyses. The results have been verified by PG&E to satisfy DCPP requirements for design-bases information. The verification of the integrity of the reactor vessel and internals system for the LTSP seismic event indicates no USQ has been raised by applying the LTSP criteria.

00-025 NSAL-00-004 Prompt Operability Assessment Compensatory Measures

Reference Document No.: NSAL-00-004
Rev. No: 0
Reference Document Title: Nonconservatisms in Iodine Spiking Calculations
Safety Evaluation Description:
Currently, DCPP TS 3/4.4.8 requires the reactor coolant specific activity to be less than or equal to 1 microcurie/gram. Due to nonconservatisms used in calculating the equilibrium iodine appearance rates for the SG tube rupture (SGTR) accident, administrative limits are required as compensatory measures to ensure that the radiological consequences of SGTR accident will remain bounded by the analysis of record. Specifically, the required administrative limits are that the primary coolant DOSE EQUIVALENT I-131(DEI) specific activities be less than or equal to 0.71 microcurie/gram for a letdown rate of 75 gpm and 0.47 microcurie/gram for a letdown rate of 120 gpm.

Safety Evaluation Summary:
Nonconservative assumptions were used in the calculation of the accident-initiated iodine spiking rates in the primary coolant for the SGTR accident. The accident-initiated iodine spikes are derived from the equilibrium iodine appearance rates. The equilibrium iodine appearance rates are those at which the various iodine isotopes enter the primary coolant system from fuel elements having cladding leaks. These appearance rates are balanced by cleanup or removal through letdown, primary coolant leaks, and radioactive decay so that the primary coolant iodine activity is maintained in an equilibrium. Revising the nonconservative analysis assumptions results in an increase in the equilibrium iodine appearance rates. This will result in increased doses for accidents that model the accident-initiated iodine spikes based on the equilibrium appearance rates. Therefore, Westinghouse has recommended the imposition of administratively-controlled compensatory measures to restrict primary coolant DEI specific activity to less than the current TS value of 1 microcurie/gram. With these administratively-controlled compensatory measures, the radiological consequences for the SGTR

accident will remain within the values currently contained in the analysis of record.

Conclusion:
The nonconservatisms described in Westinghouse NSAL-00-004 impact the SGTR accident. The implementation of the administratively-controlled compensatory measures to restrict the primary coolant DEI specific activity to less than or equal to 0.71 microcurie/gram for a letdown rate of 75 gpm and 0.47 microcurie/gram for a letdown rate of 120 gpm ensures that the calculated dose consequences of the SGTR event will not be increased. Thus, the health and safety of the public and operating staff are not adversely impacted. This activity does not result in a USQ.

00-026 Resin Volume in Chemical and Volume Control System Mixed Bed Demineralizers

Reference Document No.: DCM S-8/FSARU Section 9.3.4
Rev. No: 25/12
Reference Document Title: Chemical and Volume Control System
Safety Evaluation Description:
Chemistry requests an additional 9 cubic feet of resin be allowed in a mixed-bed demineralizer vessel for a total volume of 39 cubic feet. The mixed-bed demineralizers serve as backup cleanup bed to the deborating demineralizers during refueling outages, particularly for cleanup of RCS activity following a forced oxygenation. The additional resin volume will optimize cleanup and minimize radwaste during forced oxygenation. The FSARU and DCM S-8 Rev. 25 describe the mixed-bed demineralizers as having a resin volume of 30 cubic feet per vessel. The mixed-bed demineralizer vessel design (Drawing DC 666210-164 Rev. 2; Note 2) allows for a total of 39 cubic feet of resin in the vessel if it is flushed at end of use rather than regenerated. The FSARU and DCM S-8 must be revised to reflect the use of more than 30 cubic feet in the mixed-bed demineralizer vessels.

Safety Evaluation Summary:
Increase of the resin volume in the mixed-bed demineralizers from 30 to 39 cubic feet will not result in a change to the seismic qualification of the vessels. The demineralizers will continue to operate within the existing design parameters (pressure, flow rate, and temperature). The additional resin volume will slightly increase the pressure drop through the demineralizers but will not adversely affect the operation of the CVCS and the ability of the demineralizers to perform their designed functions.

Conclusion:
This change to the CVCS mixed-bed demineralizers will not increase the probability or the consequences of an accident. It will not increase the probability of occurrence of a malfunction or consequences of malfunction of any equipment. This change will not create any new accident or new kind of malfunction of any equipment. The margin of safety has not changed. Based on the evaluation of the 10 CFR 50.59 LBIE, this change to the FSARU and DCM does not result in a USQ.

00-027 Reactor Trip or Safety Injection

Reference Document No.: Emergency Operating Procedure (EOP) E-0
Rev. No: 24 (Unit 1) and 15 (Unit 2)
Reference Document Title: Reactor Trip or Safety Injection
Safety Evaluation Description:
Changes are proposed to EOP E-0 based on Westinghouse Owners' Group (WOG) Direct Work Items (DWI). In addition, minor changes are made to improve formatting and to accommodate simulator feedback enhancements.

Safety Evaluation Summary:
This revision to EOP E-0, "Reactor Trip or Safety Injection," places seven steps requiring verification of ESF auto actuations on an SI signal, into an Appendix. These steps are performed independently by a licensed operator, rather than by the senior reactor operator (SRO) reading the procedure. This change requires an LBIE for two reasons: (1) control room implementation of the procedure is changed, which, if used incorrectly, could aggravate the consequences of accidents discussed in the SAR, and (2) the changes could adversely impact operator action times discussed in the SAR.

An additional significant change is the incorporation of eight steps into EOP E-0 from EOP E-1.1, "SI Termination." These steps provide earlier termination of charging injection flow in the "inadvertent SI" event.

These changes are a result of WOG DWI DW-96-038. This work item was written specifically to "allow for a more timely SI termination to prevent pressurizer overfill conditions for spurious SI events."

This revision does not introduce any new steps or delete any old steps. The ESF automatic actuation steps are identical to the previously existing steps (with the exception of some minor formatting changes). They will be performed by a licensed operator dedicated to performing the appendix. The steps stand alone; they do not have internal branches or transitions that would require decision making by the operator or that might interfere with the actions of the other control room operators. The steps are simple – verification of status lights and indicators – such that a qualified operator can perform them easily.

A limiting case occurs when the control room staffing is at TS minimum (that is, a Shift Foreman and a licensed operator are available to address the event). The ESF verification steps would have to be performed prior to the stabilization and diagnostic activities. That the ESF verification steps must be performed is a knowledge issue that has been included in the simulator training on the procedure revision.

The purpose of this revision is to improve procedure response. The control room staff is able to address plant stabilization and diagnosis much sooner than before. Validation scenarios indicate that the diagnostic steps will be reached in less time than previously required. In the case of inadvertent SI, the steps required to prevent overfill of the pressurizer (and challenge to the safety valves) are performed without the delay of a procedure-transition tailboard, event classification, and emergency-response organization assignments.

Conclusion:
This revision significantly improves response time to major events without impairing control room effectiveness in mitigating the consequences of an accident. The overall probability of a radiological release will be reduced. Based on the responses of the 10 CFR 50.59 Safety Review, these changes do not constitute a USQ.

00-028 Evaluation of Feedwater Flashing - Main Steam Line Break

Reference Document No.: PGE-99-562
Rev. No: 0
Reference Document Title: Evaluation of Feedwater Flashing - MSLB
Safety Evaluation Description:
Revise MSLB FSARU Section 6.2C.3 to incorporate feedwater (FW) flashing evaluation results as established in Westinghouse letter PGE-99-562. Add text explaining the applicable unisolable, and the potential flashing, volume that must be considered. Add unisolable volumes to FSARU input Table 6.2C-23 and update revised Case 12A results in FSARU Table 6.2C-27.

Safety Evaluation Summary:
In Westinghouse letter PGE-99-523, it was identified that the maximum FW line volume considered for FW flashing effects in the existing DCPP MSLB results and as previously reported in WCAP-13908 may not have been conservative. Based on the Westinghouse revised FW flashing criteria, DCPP established the maximum FW flashing volume in accordance with calculation STA-099. Westinghouse performed an evaluation of the revised

FW flashing effect and documented the results in PGE-99-562. Westinghouse determined that only MSLB Case 12A was impacted. These revised results and basis for FW flashing evaluation need incorporation into the FSARU.

Conclusion:
The methodology for analyzing the MSLB peak pressure and temperature inside containment has not been revised, and there has been no change in any actual or assumed equipment performance credited for mitigation. The analysis has always considered FW flashing effects into the faulted steam generator (SG). This revision only implements a more appropriately conservative FW piping volume that must be considered for flashing, based on the MSLB conditions that can exist.

The NRC margin of safety for containment-response analysis is established based on maintaining the peak post-accident pressure and temperature values below the applicable design limits. The revised FW flashing effect only impacts the MSLB Case 12A results, as presented in the FSARU Table 6.2.C-27. The peak containment pressure and temperature response, which provides the basis for DCPP post-accident equipment qualification, has not changed, and there has been no change or impact on any offsite dose analysis. This activity does not result in a USQ.

00-029 Unit 1 Cycle 10 Final Reload Safety Evaluation and COLR Revision 2

Reference Document No.: COLR 1-10
Rev. No: 2
Reference Document Title: COLR for Diablo Canyon Unit 1 Cycle 10
Safety Evaluation Description:
This revised RSE documents NRC approval of LA 135. The amendment converts the CTS to the ITS. This activity is being performed to implement the ITS within the time limit from the NRC SER.

Safety Evaluation Summary:
The implementation of ITS allows more flexibility to change design limits by expanding the number of operating limits contained in the COLR. This change does not affect plant operation, or physically alter or change the function of SSCs required to mitigate the consequences of a design-basis accident nor initiates a transient or affects the probability of occurrence of any previously analyzed accident.

Conclusion:
With no change to any systems nor any change to the design-basis accident evaluation, the ITS COLR has been expanded to include more operation

limits. The 50.59 evaluation shows that the RSE is still applicable and that no reduction in the margin of safety exists. This activity does not result in a USQ.

00-030 Unit 2 Cycle 10 Final Reload Safety Evaluation and COLR Revision 3

Reference Document No.: COLR 2-10
Rev. No: 2
Reference Document Title: COLR for Diablo Canyon Unit 2 Cycle 10
Safety Evaluation Description:
This revised RSE documents the NRC approval of LA 135. The amendment converts the CTS to the ITS. This activity is being performed to implement the ITS within the time limit from the NRC SER.

Safety Evaluation Summary:
The implementation of the ITS allows more flexibility to change design limits by expanding the number of operating limits contained in the COLR. This change does not affect plant operation, or physically alter or change the function of SSCs required to mitigate the consequences of a design-basis accident nor initiates a transient or affects the probability of occurrence of any previously analyzed accident.

Conclusion:
With no change to any systems nor any change to the design-basis accident evaluation, the ITS COLR has been expanded to include more operation limits. The 50.59 evaluation shows that the RSE is still applicable and that no reduction in the margin of safety exists. This activity does not result in a USQ.

00-031 Addition of Surveillance Requirements to ECG 37.2

Reference Document No.: ECG 37.2
Rev. No: 1
Reference Document Title: Axial Flux Difference (AFD) Monitor Alarm
Safety Evaluation Description:
This change adds the following SRs to ECG 37.2:

- SR 37.2.1 Perform a functional test of the main annunciator alarm function.
- SR 37.2.2 Calibrate excore channels to agree with incore detector measurements.
- SR 37.2.3 Perform CHANNEL CALIBRATION.

The ECG Bases have been updated to discuss these three SRs.

Safety Evaluation Summary:
The requirements of TS 4.2.1.1.a.2 were relocated from CTS 3/4.2.1, "AXIAL FLUX DIFFERENCE," to ECG 37.2, Revision 0 as authorized by LAs 135/135. The remainder of CTS 3/4.2.1 is located in ITS 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)." The relocated CTS 4.2.1.1.a.2 is an SR that requires monitoring the AFD at least once per hour when the AFD monitor alarm is inoperable. Neither CTS 3/4.2.1 nor ITS 3.2.3 has an SR to determine operability of the AFD monitor. Accordingly, there were no SRs to reference or relocate when TS 4.2.1.1.a.2 was relocated to ECG 37.2. Currently STPs I-2C2 and I-2D calibrate a portion of the AFD monitor alarm circuitry. STP I-42 is being revised to include a functional test of the main annunciator alarm function. These three STPs will then provide verification that the AFD monitor alarm is operable. These STPs are consistent with other tests that are currently done and use the same types of test methods and circuit checks that are used for other tests in the surveillance test program. They pose no new operational or maintenance challenges to safety-related equipment.

Conclusion:
The addition of SRs to ECG 37.2 makes the ECG requirements more conservative than when they were in TS. Implementation of the SRs poses no new operational or maintenance challenges to safety-related equipment. Therefore, this change does not result in a USQ.

00-032 Addition of SRs to ECG 37.3

Reference Document No.: ECG 37.3
Rev. No: 1
Reference Document Title: Quadrant Power Tilt Ratio (QPTR) Alarm
Safety Evaluation Description:
This change adds the following SRs to ECG 37.3:

- SR 37.2.1 Calibrate excore channels to agree with incore detector measurements.
- SR 37.2.2 Perform CHANNEL CALIBRATION.

The ECG Bases have been updated to discuss these two SRs.

Safety Evaluation Summary:
The requirements of TS 4.2.4.1.b were relocated from CTS 3/4.2.4, "QUADRANT POWER TILT RATIO," to ECG 37.3, Revision 0, as authorized by LAs 135/135. The remainder of CTS 3/4.2.4 is located in ITS 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." The relocated TS 4.2.4.1.b is an SR that requires calculating the QPTR at least once per 12 hours when the QPTR alarm is inoperable. Neither CTS 3/4.2.4 nor ITS 3.2.4 has an SR to determine operability of the QPTR alarm. Accordingly, there was no SR to reference or relocate when TS 4.2.4.1.b was relocated to ECG 37.3. Currently STPs I-2C2 and I-2D calibrate a portion of the QPTR alarm circuitry. STP I-37-N50 is being revised to verify that the main annunciator system is capable of providing the QPTR alarm. These three STPs will then provide verification that the QPTR alarm is operable. These STPs are consistent with other tests that are currently done and use the same types of test methods and circuit checks that are used for other tests in the surveillance test program. They pose no new operational or maintenance challenges to safety-related equipment.

Conclusion:
The addition of SRs to ECG 37.3 makes the ECG requirements more conservative than when they were in TS. Implementation of the SRs poses no new operational or maintenance challenges to safety related equipment. Therefore, this change does not result in a USQ.

00-033 Valve CVCS-1HCV-142 Trim Replacement
Reference Document No.: DCP N-49496
Rev. No: 0
Reference Document Title: Valve CVCS-1HCV-142 Trim Replacement
Safety Evaluation Description:
Replace the trim set for Unit 1 valve CVCS-1-HCV-142 (Primary System and RCP Seal, Charging Flow Control Valve) from 3/4 inch (flow coefficient (C_v) of 13) to 1-1/4 inch (C_v of 26) trimset. This will reduce system resistance and will add flexibility for operators in controlling pressurizer level at high letdown flow rates. This design change will restore HCV-142 to its previous C_v configuration prior to 1994, as intended by Westinghouse; the previous configuration had functioned acceptably from initial plant startup to 1994. A similar design change was implemented in Unit 2 during 2R9. The increased trim in the Unit 2 valve has resulted in an increase in normal charging flow of approximately 25 percent over its previous rate.

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 low temperature overpressure protection (LTOP) analyses, which considers mass injection challenges to the LTOP. The recently completed LTOP analyses (STA-063 Rev. 2) which support LAs 131/133 (pressure/temperature (P/T) curves through 16 equivalent full power years (EFPY)) include consideration of the larger HCV-142 C_v 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. Operating procedures (OP) L-1, OP L-5 and OP A-6:I have previously been revised to reflect these LTOP limits calculated by STA-063 Rev. 2.

The impact on SBLOCA 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 activity does not result in a USQ.

00-035 Fireproofing On Unistruts Attached to Structural Steel Members

Reference Document No.: FHARE 141
Rev. No: 0
Reference Document Title: Fireproofing On Unistruts Attached to Structural Steel Members

Safety Evaluation Description:
FHARE 141 evaluates the impact that exposed heat transfer items will have on the fire rating of the south walls of fire zones 11-A-1 and 11-B-2. Due to space limitations, fire proofing could not be installed in 11 areas on the I beam at column line C1 in fire zones 11-A-1 and 11-B-2.

Safety Evaluation Summary:
With the exception of the exposed heat transfer items, the steel I beams at column C1, in Diesel Generator areas 1-1 and 1-2, are provided with fireproofing that provides a 3 hour fire rating. The exposed heat transfer items are not provided with any fire proofing because of the space restrictions in the area. Because of this lack of fire proofing, approximately 2.5 sq. in., in 11 locations on each column, will provide a direct heat path to the I beam. However, because of the small surface area created at the unistruts/I beam interface, these items will not impact the 3 hour fire rating of the barriers.

Conclusion:
As a result of the small surface area exposed by the heat transfer components, and the size of the I beam, the resulting increase in temperature would not cause the average temperature of the steel I beam to rise above the ASTM E-119 acceptance criteria of 1000°F. Therefore, not protecting the heat transfer items will not impact the 3 hour fire rating of the barrier. This activity does not result in a USQ.

00-036 Install Reactor Coolant System Hot Sample Panel

Reference Document No.: DCP J-049508
Rev. No: 0
Reference Document Title: Install RCS Hot Sample Panel
Safety Evaluation Description:
This DCP installs the shop-assembled, panel-mounted, RCS hot sample panel on the wall of the Unit 1 Nuclear Steam Supply Sample System (NSSSS) sample cooler room at El. 100. It will connect the panel to the following DCPP fluid systems: the hot-leg sample line, the CCW system, the NSSSS sample sink, the gross-failed-fuel-detector sample return line to the CVCS system, and the 120 volt AC system. The DCP also provides an argon bottle with regulator and connects it to the panel. When this panel is installed, it will provide the capability to collect hot and cold filtered samples from the RCS.

Safety Evaluation Summary:
The RCS hot sample panel added by this DCP only increases the sampling capability of the NSSSS. There is no change to the existing functions or capabilities of the system.

The only licensing commitment in this area is the requirement that the hot-leg sample line to the sample sink be seismically qualified. This is to assure the ability to draw a sample for boron analysis following a Hosgri event. The DCP requires the new panel and the interconnecting tubing to be seismically qualified and to meet all the requirements for PG&E "Class S".

Conclusion:
This design change meets the existing licensing commitments and there is no USQ involved.

00-038 STP M-55, Rev. 9

Reference Document No.: STP M-55
Rev. No: 9
Reference Document Title: Recording of Cyclic Fatigue or Transients
Safety Evaluation Description:
Conversion to ITS has increased the transients that are required to be tracked by this procedure. This revision incorporates all transients described in FSARU, Section 5.2 & 5.3, in accordance with ITS 5.5.5 requirement. A number of clarifications and editorial comments were also made.

Safety Evaluation Summary:
In summary, transients that had not previously been tracked are added to STP M-55.

The number of occurrences of cyclic events or transients have an ultimate design limit of maintaining the cumulative-fatigue usage factor (the combined effect of all cycles/transients) to less than 1.0 as defined by the ASME Code. There are also Code limits on the developed stresses. Stresses associated with each design cycle are determined based upon pressure and temperature parameters (e.g., RCS cooldown rate at 100°F/hr from 550°F to ambient). The number of cycles of various transients to be assumed in the stress/fatigue analysis is established by the Westinghouse Equipment Specification (E-Spec) (henceforth referred to as "design assumptions" and "design cycles"). These parameters represent conservative assumptions to bound expected plant transient conditions (e.g., RCS cooldown at 80°F/hr from 550°F to 300°F). The actual transient (or partial cycle) creates less stress than the design cycle, yet is counted as one full design cycle. Thus, neither the design assumptions, nor the design cycles are absolute limits, so long as the developed stress and cumulative fatigue usage factors are less than Code allowables.

The FSARU consistently references allowable, cumulative-fatigue usage factor as the Code limit of 1.0. SER 0 accepted reactor coolant pressure boundary components design loading combinations design limits as "comparable to the criteria recommended in RG 1.48", which refers back through to the ASME Code to a cumulative fatigue usage factor limit of 1.0.

ITS 5.5.5 reads: "This program provides controls to track the FSARU, Section 5.2 and 5.3, cyclic and transient occurrences to ensure that components are maintained within the design limits." The previous TS for transient limits only listed a sub-set of the "design transients" listed in the E-Spec.

FSARU Table 5.2-4 contains three elements: (1) a complete list of "design transients" and "design cycles" used in the Unit 2 Westinghouse E-Spec, (2) some FSARU Chapter 15 "Faulted Conditions", and (3) two transients not directly from the E-Spec list of transients.

Of note is that the E-Spec list of "design transients" and "design cycles" has gone through various revisions, such that, depending on when the various RCS component's stress/fatigue analyses were performed, different "design transients" and "design cycles" were used. Subsequent analyses were performed in an attempt to bring the original analyses up-to-date to match those listed in FSARU Table 5.2-4. However, it doesn't appear that all analyses were successfully brought fully up-to-date. Because of this, STP M-55 has been revised to use the lesser number of occurrences as "Limits," where applicable. This assures that the cumulative usage factor will remain within Code allowable.

As footnoted in FSARU Table 5.2-4, in accordance with ASME Code, faulted conditions are not included in fatigue evaluations. Therefore, there is no Effect upon the cumulative usage factor by including these transients in STP M-55 scope.

The two transients included in FSARU Table 5.2-4 that are not directly from E-Specs are: (1) Tavg coastdown from nominal to reduced temperature (50 occurrences), and (2) Design Earthquake (DE) (20 occurrences).

WCAP-13457, "Diablo Canyon Units 1 & 2 Tavg/Power Coastdown Program Technical Report," evaluated the Effect upon the structural integrity of RCS components and concluded that, for most RCS components, there was negligible effect upon previously determined usage factors, and for SG divider plate (limiting SG RCS component), the usage factor remained below design limit of 1.0. It should be noted that there is no basis in the WCAP for the limiting the number of coastdowns to 50 occurrences, and that limit is retained (for now) only for consistency with the current FSARU.

Although FSARU Table 5.2-4 currently lists 20 occurrences of a DE, it appears that some of the stress/fatigue analyses for RCS components only analyzed for 5 DE. Therefore, by limiting the number of occurrences to 5, the stress/fatigue will remain within analyzed limits.

FSARU Section 5.3 contains a list of upset conditions that are evaluated in Chapter 15. Three of these transients (a fourth is being deleted by separate FSARU change), "Complete Loss of Forced Reactor Coolant Flow," "Loss of Normal Feedwater," and "Accidental De-pressurization of the Reactor Coolant

System," were not included in the E-Spec, and therefore, were not included in the various RCS component's stress/fatigue analyses. To ensure that the cumulative-fatigue usage factor remains within limits, administrative limits for these transients utilize some of the occurrences of a comparable transient that were included in the stress/fatigue analyses. This ensures that the fatigue cumulative usage factor remains within Code allowable limit.

Conclusion:
These changes will not cause any RCS component to exceed the design limit for cumulative-fatigue usage factor (1.0). Therefore, there is no USQ associated with this change.

00-039 Containment Isolation Valve List

Reference Document No.: AD13.DC1, Attachment 7.7
Rev. No: 7
Reference Document Title: Containment Isolation Valves
Safety Evaluation Description:
This 10 CFR 50.59 safety evaluation is being performed to comply with AD13.DC1, Step 5.16.1. AD13.DC1, Attachment 7.7, now a table in its entirety, has been expanded to include all containment isolation valves. Individual valve information has been expanded to include TS condition and isolation valve type. Information in the eight function lists in the old Attachment 7.7 has been incorporated into the table. The table is now sorted by system and component number, changed from penetration number. Valve service detail has been expanded, since it is now component specific. Unit differences (component number or type) are documented in the table. Components not leak rate tested have been added. The basis for exclusion from surveillance field was eliminated and in its place all V-600 or M-8 leak tests have been added under the surveillance procedure column and, in addition, STPs V-6 and I-1D have been identified as applicable.

Safety Evaluation Summary:
To make the CIV list more useful to Operations, all CIVs are now in the list, and the list sorted by system and valve number. The previous list did not include valves on "closed systems" as defined the FSARU. In addition to the information in the previous list, the valve type and applicable TS condition(s) have been added to the table for each component. This is a complete list of all CIVs. The changes to the list are administrative in nature. The list's purpose is to simplify containment operability determinations when problems or issues arise regarding CIVs or components.

Conclusion:
This change does not constitute a USQ, nor does it have any effect on offsite-dose limits defined in 10 CFR 100. This change has no licensing-basis impact on DCPP.

00-040 Revision 3 to ECG 52.2

Reference Document No.: ECG 52.2
Rev. No 3
Reference Document Title: Technical Support Center ERDS
Safety Evaluation Description:
ECG 52.2 is being revised to reflect the upgrades that have been completed on the Emergency Response Data System (ERDS)

The limiting conditions for operation (LCO), Action Statements, and SRs have been revised to replace the terms "Computer A" and "Computer B" with the terms "Channel - Unit 1" and "Channel - Unit 2."

The reference to a specific STP has been deleted.

The Bases have been expanded using additional licensing material and have been reformatted in the ITS format.

Safety Evaluation Summary:
As a result of reconfiguring the PG&E computer networking system and elimination of the old (Banyan) system, the ERDS function was transferred to the Safety Parameter Display System (SPDS) in 1999. The previous ERDS consisted of one stand-alone computer and associated data collection and transmittal equipment for each unit. The ERDS for each unit was capable of being cross tied to the ERDS for the other unit through a local Ethernet link. The reconfigured ERDS has replaced the stand-alone computers with new software on SPDS. The SPDS host computers for each unit have the capability to be cross tied to the other unit. ECG 52.2, which controls ERDS, is being revised to reflect the reconfigured ERDS. The transfer of ERDS to SPDS results in descriptive changes only to the ECG. No changes in the technical requirements for the LCO, Action Statements, or SRs are necessary.

Removal of the reference to a specific STP and reformatting of the Bases to the ITS format are administrative changes. Expansion of the Bases to incorporate additional licensing material is administrative in nature, based on the fact that the information is included in other approved documents already, and does not change the technical requirements of the ECG.

Conclusion:

The changes proposed for Revision 3 to ECG 52.2 involve descriptive changes, but no technical changes, to the LCO, Action Statements, and SRs. The remaining changes are administrative in nature. The changes do not result in a USQ.

00-041 ITS Bases

Reference Document No.: B 3.8.1, SR 3.8.1.14

Rev. No: 1

Reference Document Title: ITS Bases

Safety Evaluation Description:

Consistent with NUREG-1431, Revision 1, the phrase, "The DG voltage shall be 4160 volts +240 volts and 375 volts within 13 seconds after the start signal. The DG frequency shall be 60 Hz ±1.2 Hz within 13 seconds after the start signal," added to the Bases by DOC-01-30-LG of LAR 97-09 will be removed from the Bases of SR 3.8.1.14.

Safety Evaluation Summary:

The CTS currently requires a timed start for a 24-hour load run. This requirement was moved to the ITS Bases of SR 3.8.1.14 by DOC 01-30-LG during the TS conversion under LAR 97-09. This information is not a part of the NUREG-1431 text for SR 3.8.1.14 nor the Bases of NUREG-1431 for SR 3.8.1.14. There is also no requirement to perform a timed start for a 24-hour load found in RG 1.108 (PG&E is committed to this RG for testing of the emergency diesels). This change will remove the relocated information from the Bases that imply a 24-hour load run must begin with a timed start. A timed start will, however, continue to verify equipment function on a more restrictive frequency than SR 3.8.1.14, since it is still required by several ITS SRs (e.g., the start from standby conditions (SR 3.8.1.7) every 6 months). These required timed starts are consistent with NUREG-1431 and RG 1.108.

Conclusion:

This information may be removed from the Bases without creating a USQ.

00-042 Administrative Control over Reactor Coolant System Pressure Temperature Limits

Reference Document No.: PTLR-1

Rev. No: 0

Reference Document Title: Pressure Temperature Limits Report (PTLR) for Diablo Canyon

Safety Evaluation Description:

The pressure/temperature (P/T) limits imposed on RCS operation have been revised to separate the calculated limits from the ITS.

Safety Evaluation Summary:
The P/T limits are determined from material properties at critical locations in the reactor pressure vessel. The adjusted reference temperature (ART) is the nil-ductility transition temperature (NDT) adjusted for radiation embrittlement. The temperature shift due to embrittlement is ◊RTndt. The heat up and cooldown curves use the fracture mechanics in Appendix G of ASME Section XI and are based on the stress intensity factors for the material at various temperatures. These are also adjusted for ◊RTndt. The method for determining LTOP incorporates the changes in NDT and calculates the effect of various mass injection events. These effects determine the administrative controls on plant equipment which are also described in PTLR-1. The methodology used to calculate the actual limits is very similar to (i.e. was taken from) that used in Westinghouse WCAP 14040-NP-A and previous Westinghouse analyses, which have been reviewed and accepted by the NRC. The specific values are documented in PG&E calculations that have been reviewed by the NRC. The status and variety of the calculations and references to various analyses have caused concerns on the part of the NRC. PG&E has committed that these calculations will be consolidated into a cohesive document to form the bases for the PTLR. Upon NRC review and acceptance of the PTLR bases and methodology, the control of revisions to the PTLR will transfer to PG&E. For the interim, any changes to the RCS pressure and temperature curve or LTOP restrictions will be reviewed and approved by the NRC prior to implementation of the changes. This PTLR relocates RTndt information previously located in various calculations into a common document containing the resultant P/T and LTOP limits.

Conclusion:
The operating limits for RCS pressure and temperature for up to 16 EFPY have been reviewed for safety significance and previously accepted by the NRC in LA 133/131. The relocation of those limits from the CTS to the separate PTLR-1 procedure has been accepted by the NRC in LA 135/135 with the proviso that the NRC has authority over revisions to the P/T limits. The incorporation of this arrangement into the FSARU has been determined to not result in a USQ.

00-044 Requirements for Source Range Audible Indication in Containment

Reference Document No.: TS Bases B 3.9.3

Rev. No: 0

Reference Document Title: Nuclear Instrumentation

Safety Evaluation Description:

Discussion on the audible count-rate indication in containment is added to the Background and Applicable Safety Analysis sections of TS Bases B 3.9.3 in order to clarify the operability requirements of the source range neutron flux monitors. The audible count-rate indication in containment is not required for operability of the source range flux monitors.

Safety Evaluation Summary:

TS 3.9.3 requires that two source range neutron flux monitors be operable to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be operable, each monitor must provide visual indication and at least one of the two monitors must provide audible alarm and count-rate indication in the control room in accordance with the boron dilution analysis. The purpose of the LCO for the source range neutron flux monitors is to assure that DCPP will have at least 30 minutes to detect and terminate a boron dilution event (consistent with NUREG 0800). LA 28/27 SER, dated April 21, 1988, provided NRC approval of DCPP's design. The boron dilution safety analysis credits control room indication but does not credit audible count-rate indication in containment. Therefore, the audible count-rate indication in containment is not required for operability of the source range neutron flux monitors under 10 CFR 50.36. This change removes any implication that the audible count-rate indication in containment is required for operability of the source range neutron flux detectors.

Conclusion:

The audible count-rate indication in containment is not credited in the mitigation of any accident and should not, therefore, be a requirement for operability of the source range neutron flux monitors. This activity does not result in a USQ.

00-045 Heatup of Unit 1 to Mode 3

Reference Document No.: N/A
Rev. No: N/A
Reference Document Title: Heatup of Unit 1 to Mode 3
Safety Evaluation Description:
The LBIE addresses the heatup of Unit 1 to Mode 3 from Mode 5 without the 12 kV buses being supplied from the auxiliary (500 kV) power system. The 4 kV buses would be capable of being supplied from either the 500kV or the startup (230 kV) power systems prior to entry into Mode 4 as required by TS 3.8.1.1.

Safety Evaluation Summary:
The complete loss of forced coolant flow (CLOF) accident is of concern because, if the reactor is not tripped in a timely manner, the loss of flow could result in exceeding departure from nucleate boiling ratio (DNBR). However, this accident is not applicable in those Modes in which the reactor is already tripped.

In Mode 3, the control rods are already fully inserted and only one RCP is required to be operating when the reactor trip breakers are open, as specified in CTS 3.4.1.2 (two RCPs required when reactor trip breakers are closed). Action statement c requires that if no loops are in operation, all activities involving a reduction in boron concentration are to be terminated, and one loop is to be returned to service as soon as possible. The ITS clarifies that all RCS loops may be removed from service for 1 hour out of every 8 hours and that the comparable required action to terminate reductions in boron concentration is to maintain margin to criticality because boron mixing is poor during natural circulation cooling. This does recognize that natural circulation will occur and provide cooling. Additionally, the condition that the RCS would be in following the reactor trip during a CLOF is similar to the condition that the RCS would be in following a loss of the 230 kV system in Mode 3, except that the heat load in the reactor would be significantly reduced.

Since the reactor is already tripped and the plant would be in a similar condition as at the end of the CLOF analysis, but much earlier, the consequences of the CLOF are bounding for this condition. Additionally, startup testing has demonstrated the capability to cool the reactor using natural circulation.

Conclusion:
With Unit 1 operation limited to Mode 3, potentially affected accidents such as the CLOF are not applicable but bound the proposed scenario. Therefore, the proposed change does not create a USQ.

00-046 Containment Isolation Valve List

Reference Document No.: AD13.DC1, Attachment 7.7
Rev. No: 7
Reference Document Title: Containment Isolation Valves
Safety Evaluation Description:
This LBIE is being performed to comply with AD13.DC1, Step 5.16.1. AD13.DC1, Attachment 7.7, is being revised to add additional components discovered in the process of updating FSARU Table 6.2-39 and adds scope to previously approved LBIE 00-039. Most of the changes involved penetrations that had already been exempted from surveillance by previous license agreements that no longer apply under ITS. As a result of capillary fill valves on containment penetrations being added to the list, a detailed review of all penetrations resulted in a few instrument valves being added to the list.

Safety Evaluation Summary:
The previous list did not include a few valves on "closed systems" as described in the FSARU or systems in service following an accident, which previously had been exempted from surveillance. This is now considered a complete list of all containment isolation valves. The changes to the list are administrative in nature. The list's purpose is to simplify containment operability determinations when problems or issues arise regarding containment isolation valves or components.

Conclusion:
This change does not constitute a USQ, nor does it have any effect on offsite-dose limits defined in 10 CFR 100. This change has no licensing-basis impact on DCPP.

00-047 Allow Operation in Modes 3-5 with Makeup Mode Selector Switch Out of Service

Reference Document No.: OP B-1A:VII, Unit 1
Rev. No: 19
Reference Document Title: CVCS – Makeup Control System Operation
Safety Evaluation Description:
Makeup Mode Control Switch, 43/MU, is failing, requiring replacement. This procedure provides the instructions to be used by the Operators for RCS makeup control during the time period, approximately two days, that 43/MU will be removed for repair. Under normal operations, the makeup mode is selected using 43/MU, and then the desired operation is initiated using the Makeup Start/Stop switch. Neither of these switches will be active during the repair, forcing Operators to make up to the RCS using the individual valve

controllers. Using this method is essentially the same as manual makeup described in the FSARU Section 9.3.4.2.2.4, except that the flow integrators will not secure the desired evolution at completion. Securing RCS makeup evolutions will require specific operator actions.

PG&E Drawing 663210-162-1 (Westinghouse Instructions for Makeup Control System Failures) describes actions to be taken in the event of failures on the makeup control system. This drawing states that "all remote operated valves and pumps in the RCS makeup control system can be placed in manual control and a boron dilution or make-up operation can be carried out manually from the control room." Therefore, operation of the RCS makeup control system, as described in this procedure revision, is within the system design.

The repairs to 43/MU will be performed in Modes 3-5. This LBIE only addresses SAR discussions for shutdown conditions.

Safety Evaluation Summary:
The major impact of the change on operation of the plant is that there will be no automatic shutoff for dilution or boration flow once the operators initiate the action. Normally, with the 43/MU in place, boration or dilution will secure once the flow integrators have counted down to zero. With the 43/MU switch removed, this function will be lost, and increased Operator attention to the evolution will be required. The loss of this function is mitigated by procedural requirements for increased vigilance during any dilution that operators initiate.

Other functions of the system will not be affected adversely. There will not be automatic makeup capability to the RCS, however, the FSARU discusses loss of this function in Section 9.3.4.2.2.4. The credited means for emergency boration will still be completely functional via CVCS-8104 and changing charging pump suction to the refueling water storage tank (RWST). Due to having emergency boration capability, the ability to maintain shutdown margin will not be compromised.

Conclusion:
Removal of the 43/MU switch has an impact on the initiator for a boron dilution accident. Without the 43/MU switch, no automatic securing or dilution flow will occur. However, this is compensated for by the short time frame for maintenance on the system and the reliance of the safety analysis on source range indication for determining that a dilution accident is in progress. As Operator action is required to terminate a boron dilution accident, this is not an increase in the probability or severity of this accident. Makeup control system operation is not a factor in any of the other accidents analyzed in the SAR. This activity does not result in a USQ.

00-048 Revision 2 to ECG 4.4

Reference Document No.: ECG 4.4
Rev. No: 2
Reference Document Title: Instrumentation - Turbine Overspeed Protection
Safety Evaluation Description:

1. Technical Specification Interpretation (TSI) 86-07 has been incorporated into ECG 4.4 by adding a note to the applicability statement and a discussion of the note to the applicability section of the Bases. As a result, TSI 86-07 may be rescinded.
2. ECG SR 4.4.2 has been changed from "CHANNEL CALIBRATION" to "CHANNEL OPERATIONAL TEST."
3. The Bases have been expanded and rewritten in ITS format.

Safety Evaluation Summary:
A0413787 requests that TSI 86-07 be incorporated into ECG 4.4 and that TSI 86-07 be rescinded. ECG 4.4 is a relocated TS (3/4.3.4.1, "Instrumentation - Turbine Overspeed Protection") that was approved for relocation by LAs 120/118. TSI 86-07 applied to TS 3.3.4.1 and, by this change, is now incorporated into ECG 4.4 by the addition of a note to the ECG applicability statement and a discussion of the note in the applicability section of the ECG Bases. There are no changes to the technical requirements of ECG 4.4 by making this change. As a result of this change, TSI 86-07 may be rescinded.

AR A0473019 requests that ECG SR 4.4.2 be changed from "CHANNEL CALIBRATION" to "CHANNEL OPERATIONAL TEST." This is because the STP that verifies this SR, STP M-21B, "Main Turbine Overspeed Trip Tests," does not include "adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input," as defined by TS (adjustment is done via other procedures). The testing more closely meets the definition of a "Channel Operational Test," in that trip points are verified to be within range. If a trip is not within range, adjustments are made to the P2000 computer or to the mechanical trip mechanism, as appropriate, to bring the trip within range. The trip adjustments are not made with an STP. This change results in no change to the way the STP M-21B is performed, and the STP continues to meet the requirements of the SR, both in its present form as an ECG 4.4 SR, and as it did in the past when the SR was part of TS 3/4.3.4.1. Required adjustments will continue to be made in accordance with other procedures.

The ECG 4.4 Bases have been expanded to include additional licensing information and have been formatted in the ITS format to make them more useful. Expansion of the Bases and use of ITS formatting is administrative in nature, based on the fact that the information is included in other approved documents already and does not change the technical requirements of the ECG.

Conclusion:
The changes proposed for Revision 2 to ECG 4.4 do not change the technical requirements of the ECG and are administrative in nature. They do not result in a USQ.

00-049 Add ASW Pump Room Temperature Monitoring to ECG 23.1

Reference Document No.: ECG 23.1
Rev. No: 2
Reference Document Title: Area Temperature Monitoring
Safety Evaluation Description:

ECG 23.1 has been revised to include the ASW pump rooms as areas requiring temperature monitoring, and additional administrative changes have been made as follows:

1. The ECG LCO and Condition B have been clarified to show that a temperature increase of 16°F over the limit (rather than 30°F) applies to the cable-spreading room.
2. Action B.1.2 has been added to provide an option to perform an evaluation to demonstrate operability of equipment in areas where high temperature limits have been exceeded.
3. Items 31 and 32, ASW Pump No.1 Room and ASW Pump No.2 Room, have been added to Table 23.1-1.
4. The Bases have been expanded and rewritten in the ITS format.

Safety Evaluation Summary:
An evaluation (documented in AR A0389159-A/E17) and performed as part of implementation of DCP J-49233, "ASW Pump Room Temperature Recorder Removal," concluded that ASW pump room temperature monitoring requirements should be added to ECG 23.1. Revision 2 to ECG 23.1 makes this addition. This change is conservative because it adds ASW pump room temperature monitoring requirements to the ECG and does not change any other temperature monitoring requirements.

The addition of Action B.1.2 to Condition B provides an option to address equipment operability in area(s) where temperature limits have been exceeded by 30°F (16°F for the cable spreading room). The purpose of ECG 23.1 is to assure that equipment operability is not lost due to excessive area temperatures. If equipment operability can be shown by successful completion of Action B.1.2, then the purpose of the ECG will have been met.

The ECG Bases have been expanded to include additional licensing information and have been formatted in the ITS format to make them more useful. Expansion of the Bases and use of ITS formatting are administrative in nature, based on the fact that the information is included in other approved documents already, and do not change the technical requirements of the ECG.

Clarifications of the LCO and Condition B to indicate the temperature increase limit for the cable spreading room are administrative in nature and do not change the ECG requirements.

Conclusion:
The proposed revision to ECG 23.1 involves a conservative change that adds requirements to the ECG, a change that provides an option to evaluate equipment operability, and administrative changes that do not change the technical requirements of the ECG. The proposed revision does not result in a USQ.

00-050 Containment Isolation Valve List
Reference Document No.: AD13.DC1 Attachment 7.7
Rev. No: 8
Reference Document Title: Containment Isolation Valves
Safety Evaluation Description:
This 10 CFR 50.59 Safety Evaluation is being performed to comply with AD13.DC1, step 5.16.1. AD13.DC1, Attachment 7.7, is being revised to temporarily add a note next to valves 8885A and 8885B. The two valves, SI-8885A and SI-8885B, are residual heat removal (RHR) test line valves and as a result of these valves being part of the first line of defense to ensure RHR flow separation in the event of a LOCA, these valves will be controlled by temporary procedure (TP) TO-0003. A separate, detailed LBIE, No. 00-054, has been performed for TP TO-0003.

Safety Evaluation Summary:
Unit 1 valves SI-8885A and SI-8885B will have their fuses reinstalled, one valve at a time, and will be opened along with containment isolation valves SI-8871 and SI-8961 while an RHR pump is running on recirculation. This alignment will provide flow and will flush approximately 2,400 gallons of water to a liquid holdup tank (LHUT). Details of this flush and all issues evaluated are detailed in LBIE No. 00-054.

Conclusion:
This change does not result in a USQ, nor does it have any effect on offsite-dose limits defined in 10 CFR 100. This change has no licensing-basis impact on DCPP.

00-051 Containment Isolation Valve List

Reference Document No.: AD13.DC1, Attachment 7.7
Rev. No: 9
Reference Document Title: Containment Isolation Valves
Safety Evaluation Description:
This 10 CFR 50.59 Safety Evaluation is being performed as required by AD13.DC1, step 5.16.1. AD13.DC1, Attachment 7.7 is being revised to delete Note 1 that was added by the previous Rev 8 to allow CIVs 8885A and 8885B to be opened under TP TO-0003, "Flushing RHR Piping in Modes 1, 2 or 3 to Reduce Radiation Levels."

Safety Evaluation Summary:
Revision 7 was never implemented following PSRC approval. Revision 9 will be implemented to comply with ITS and is identical in all respects to Revision 7.

Conclusion:
This change does not result in a USQ, nor does it have any effect on offsite dose limits defined in 10 CFR 100. This change has no licensing-basis impact on DCPP.

00-052 Replacement of Centrifugal Charging Pump 2-1

Reference Document No.: DCP N-050523
Rev. No: 0
Reference Document Title: Replacement of CCP 2-1
Safety Evaluation Description:
This design change is issued to replace CCP 2-1 during Modes 1-3. The replacement pump will consist of a spare casing available at DCPP with a new impeller purchased from Braidwood (stock code 95-3820).

This LBIE is based on this design change being performed after the ITS is implemented. The ITS and FSARU (Revision 14) are used for this evaluation.

Safety Evaluation Summary:
TS 3.5.2 specifies the allowed outage time (AOT) for one train inoperable as 72 hours. Implementation of this design change is expected to exceed the current AOT. An LAR is being submitted to the NRC separately for approval of the AOT extension.

The CCP 2-1 is replaced with equivalent (like-for-like) pump internals having a performance test curve that is within the maximum/minimum safeguards pump curves. As this does not change the system resistance, as addressed below, this activity will not affect the ECCS functions or the performance of the system to mitigate the consequence of a design-basis accident. A post maintenance test (PMT) will be performed to verify that the pump performance curve is within the maximum/minimum safeguards pump curve.

An evaluation was performed (Calculation STA-0122, Rev. 0) to determine the impact of the replacement pump and whether the flow characteristics of the ECCS subsystem would be altered such that a flow-balance test should be performed. For a fluid system, such as the ECCS, the flow distribution can be calculated based on the system resistance and the performance of the pumps in the system. If an actual pump performance curve and the system resistance are known, the flows of a fluid system can be calculated.

The vendor has performed a flow test of the original pump and has provided a certified pump performance curve of the replacement impeller. Using this impeller in a spare casing has negligible impact on the performance curve.

The system resistance for the CCP portion of the ECCS system was obtained during the performance of STP V-15, "ECCS Flow Balance" during 2R9 using the existing CCP 2-1. STP V-15 establishes throttle valve positions and, consequently, system resistance to assure that the ECCS system is performing within the bounds of the safety analyses criteria. There have been no changes to the system since this test that would impact the system resistance.

Conclusion:
This design change is an equivalent replacement (like-for-like) of the existing equipment and will not affect the ECCS functions or the performance of the system to mitigate the consequences of a design-basis accident.

This design change does not change how the ECCS functions or operates. The calculated ECCS pump operating conditions are within the maximum/minimum safeguards flows and pump total-developed head. With the fixed system resistance and the vendor-certified performance curve for the pump (the pump flow verification testing will be performed after pump replacement), the resulting CCP flow distribution was calculated (Calculation STA-122, Rev. 0) and verified to satisfy the FSARU Section 6.3.4.4. Therefore, CCP 2-1 will perform its intended safety-related functions within the required designed minimum/maximum flow limitations. It is concluded that this design change does not affect the ECCS subsystems in any manner, as described in the FSARU, to alter the subsystem flow characteristics such that the performance of a flow balance test during shutdown would be required. This activity does not result in a USQ.

00-053 Moving Spare 25/12 kV Auxiliary Transformer

Reference Document No.: TP TB-0004
Rev. No: 0
Reference Document Title: Moving Spare 25/12 kV Auxiliary Transformer
Safety Evaluation Description:
This safety evaluation is for heavy load lifting and moving of the spare auxiliary 25/12 kV transformer. The scope of the change includes movement of the transformer by an outside contractor. The spare Unit 2 main bank transformer pad will be used as the site location for placement of the transformer.

The transformer will be installed adjacent to an energized 500 KV "A" phase transformer and in proximity to the Standby Startup 22 Transformer. These power sources make up the offsite power sources for Unit 2. The Auxiliary Transformer may be installed with Unit 2 in Modes 1 through 6.

This LBIE is similar to previously approved LBIE No. 98-20, which removed this transformer from service, and LBIE No. 99-108 which removed the spare main bank transformer at power. Conditions that are similar or different from the previous LBIEs are as follows:

Similarities:

1. Contractor and method, as used in LBIE No. 98-20
2. Load path from the parking lot to the plant spare pad
3. Contractor equipment will be used
4. Transformer being moved

Differences:
Location and placement of the crane for offloading; Crane and boom will be positioned to maximize the distance away from the Startup 22 Transformer and also from the energized overhead 500 kV lines. This move will be performed with the transformer drained of oil, which will reduce the weight of the transformer by approximately 24,000 pounds, for a total weight of approximately 86,500 pounds. The crane used by this procedure has a maximum capacity of 400,000 pounds. The lifting of this load is well within the safe-handling limits of the crane and boom.

Safety Evaluation Summary:
The task of installing and removing transformers has been performed previously. The methods and contractor used have been evaluated and accepted. "Lessons Learned" have been applied to capture past experiences and improve the process. The transformer 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 2R9 and are familiar with DCPP controls, procedures, transformers, and equipment.

Civil Engineering has evaluated the need for temporary reinforcement of underground structures that could 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.

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 procedure OM6.ID7 "Activities Near High Voltage Equipment."

Transformer tipping was evaluated. Tipping of the transformer or sliding of the transformer off the transporter will not occur. The size and weight of this transformer is within standard approved-for-highway use transporters. The transformer will be secured to the transporter prior to movement.

Civil Engineering has evaluated the load path. The transporter and tractor loading on the path is slightly greater than H-20 specification (the maximum load for which a normal highway is designed). These structures were evaluated in AR A0460733, evaluation 8 (previously evaluated with larger loads in AR A0367538, evaluation 28).

Previous transformer moves have been successfully performed with transformers approximately the same size and larger.

Conclusion:
The relocation of the existing spare auxiliary transformer does not constitute a USQ and is within the existing licensing basis. The replacement of transformer by the drayage contractor has been thoroughly evaluated by DCPP personnel for effects on the plant, safety, and the environment. The contractor and DCPP have incorporated necessary controls to perform this activity safely and within existing construction standards and codes.

00-054 RHR Flush - Unit 1

Reference Document No.: TP TO-0003
Rev. No: 0
Reference Document Title: Flushing RHR Piping in Modes 1, 2, and 3 to Reduce Radiation Levels
Safety Evaluation Description:
During startup from the Unit 1 forced outage, a crud burst resulted in elevated radiation levels around the RHR system. Prior experience has shown that system flushing is effective in reducing area dose rates.

TP TO-0003 was developed to flush each RHR train and the RHR crossover piping between normally open valves RHR-8716A and RHR-8716B in the Auxiliary Building. This is accomplished by starting RHR Pump 1-1 and placing it on recirculation. SI system test line valves SI-8871 and SI-8961 at penetration 51B are opened, then SI-8885A inside containment off of containment penetration 25 is opened, flushing RHR train one discharge piping through SI system test line piping to the LHUTs. With RHR Pump 1-1 still running, valve SI-8885A is closed, and SI-8885B is opened to flush the RHR cross-tie piping downstream of HCV-637 and HCV-638. RHR Pump 1-1 is then stopped, and RHR Pump 1-2 is started to flush the second RHR train through SI-8885B to complete the flush sequence. The system is then realigned to normal configuration.

Safety Evaluation Summary:
TP TO-0003 provides instructions for flushing out certain portions of the Unit 1 RHR system between the RHR pumps and containment to reduce radiation levels in the vicinity of RHR components. This procedure will be performed in Modes 1-3.

This activity places the plant in an abnormal alignment for Modes 1-3 that involves connection of the RHR system to the SI test lines and flow through containment penetrations 24, 25, and 5B to the LHUTs. RHR suction will be provided by the RWST. RHR piping will be flushed until dose rates in the vicinity of the RHR system drop to acceptable levels as determined by radiation protection personnel while control room operators monitor RWST and LHUT levels.

There will be no impact to achieving the required containment recirculation sump Level of 92.5 feet required for switchover from the ECCS injection phase to cold leg recirculation in accordance with EOP E-1.3, "Transfer to Cold Leg Recirculation". It is estimated that 800 gallons of water will be flushed through each train of RHR and the RHR cross tie piping for a total of 2,400 gallons pumped from the RWST, through the RHR system, into the SI test lines and to the LHUTs. Radiation protection personnel will be monitoring radiation levels in the vicinity of RHR components during the flush. The volume flushed may be more than 2,400 gallons depending on the rate of change of radiation levels. RWST level will be verified to be at a sufficient level to assure level does not drop below the limit of 93 percent during the flush to assure it remains above that required to ensure an adequate margin is maintained. RWST level is currently maintained at approximately 97 percent. Each 1 percent of level in the RWST equates to approximately 4,500 gallons. Therefore even if twice the estimated volume is flushed, the RWST level reduction would be just over 1 percent.

While this evolution is performed, containment isolation valves SI-8885A, SI-8885B (penetrations 24 and 25) will be opened under administrative control. These are remote-manual valves that are treated as sealed-closed valves as specified in FSARU Table 6-2.39. TS LCO 3.6.3, "Containment Isolation Valves," allows locked or sealed-closed valves to be opened on an intermittent basis under administrative control. In accordance with TS and site administrative procedures, the opening of locked or sealed-closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing a person who is in constant communication with the control room at the valve controls, (2) instructing this person to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The effect of adding the current amount of activity (2,400 or more gallons at an activity level of 4 microcuries per milliliter of primarily Co-58) to the LHUT was evaluated with respect to the FSARU Chapter 15 Liquid Holdup Tank Rupture accident (FSARU Sections 15.4.8 and 15.5.25). The FSARU accident assumes complete filling of an LHUT (83,200 gallons) at a rate of 120 gpm for the RCS following operation for and equilibrium core cycle with

1 percent defective fuel. A ruptured LHUT under these conditions will result in a release at the site boundary of approximately ½ percent of the10 CFR 100 limits. The design-basis accident RCS activity level (microcuries per gram) is approximately 40 times the activity currently in the portion of the RHR piping with elevated dose rates. Therefore, as a result of the activity being significantly lower than accident conditions and the relatively low volume of water being transferred to the LHUTs, this activity does not increase the consequences of an accident previously evaluated.

Conclusion:
All SSCs involved would be available to perform their intended safety functions. There in no impact to the ECCS LOCA flow path. There is no impact to the EOPs. There is no impact to previously evaluated accidents. The proposed changes do not result in a USQ.

00-055 Containment Isolation Valve List

Reference Document No.: AD13.DC1, Attachment 7.7
Rev. No: 10
Reference Document Title: Containment Isolation Valves
Safety Evaluation Description:
This 10 CFR 50.59 Safety Evaluation is being performed to comply with AD13.DC1, step 5.16.1. AD13.DC1, Attachment 7.7, now a table in its entirety, has been expanded to include all CIVs. Individual valve information has been expanded to include TS condition and isolation valve type. Information in the eight function lists in the old Attachment 7.7 has been incorporated into the table. The table is now sorted by system and component number, changed from penetration number. Valve service detail has been expanded since it is now component specific. Unit differences (component number or type) are documented in the table. Components not leak-rate tested have been added. The "Basis for Exclusion from Surveillance" field was eliminated and in its place all V-600 or M-8 leak tests have been added under the "Surv Procedure" column and, in addition, STPs V-6 and I-1D have been identified as applicable.

Safety Evaluation Summary:
To make the CIV list more useful to Operations, all CIVs are now in the list and the list is sorted by system and valve number. The previous list did not include valves on "closed systems" as defined in the FSARU. In addition to the information in the previous list, the valve type, and applicable TS Condition(s) have been added to the table for each component. This is a complete list of all CIVs. The changes to the list are administrative in nature. The list's purpose is to simplify containment operability determinations when problems or issues arise regarding CIVs or components.

Conclusion:
This change does not constitute a USQ, nor does it have any effect on offsite dose limits defined in 10 CFR 100. This change has no licensing-basis impact on DCPP.

00-056 DCM S-17B, Revision 17

Reference Document No.: DCM S-17B
Rev. No: 17
Reference Document Title: DCM S-17B, Revision 17
Safety Evaluation Description:
Change the DCM statement that ASW gate motor breakers are normally racked out to a statement that the breakers are administratively controlled in the open position. This will also result in a change to procedure commitment database (PCD) commitment T35190 and OP E-5:I.

Safety Evaluation Summary:
By maintaining the ASW gate breakers (which are in the vital 480 volt load centers in the auxiliary building) open, spurious operation of the gates due to fire will be prevented.

Conclusion:
Opening the breakers provides the same protection as racking the breakers out and can be considered an equivalent action. This activity does not result in a USQ.

00-057 Revise FSARU, Section 2.3.3

Reference Document No.: FSARU, Section 2.3.3
Rev. No: 13
Reference Document Title: Onsite Meteorological Measurement System
Safety Evaluation Description:
Revise FSARU, Sections 2.3.3.1, 2.3.3.2, 2.3.3.3, and 2.3.3.4, to remove explicit detail pertaining to meteorological monitoring instrumentation. The level of detail presently in the FSARU is not required to support the design basis of the system.

Safety Evaluation Summary:
The FSARU currently describes the output voltage and current values of each meteorological instrument processor. It also unnecessarily describes the wind

speed sensors as cup type which precludes upgrading to a more reliable type of sensor.

Conclusion:
This description of the meteorological instruments in FSARU, Section 2.3.3 does not require the current level of detail to support the design-basis requirements of the SER, NUREG 0654, and RGs 1.23 and 1.97. The descriptive detail contained in the FSARU prevents upgrading the instruments and may result in the system not conforming the FSARU description. This FSARU change does not affect the Emergency Plan, as no equipment changes are being made, and the same detailed description of the meteorological monitoring instrumentation does not exist in the Emergency Plan. This activity does not result in a USQ.

00-058 Replace RVLIS and TMS, Unit 2

Reference Document No.: J-050434
Rev. No: 1
Reference Document Title: Replace RVLIS and TMS, Unit 2
Safety Evaluation Description:
This change removes the existing reactor vessel level instrumentation system (RVLIS) and thermocouple monitoring system (TMS) processor chassis, display and related hardware from Post Accident Monitor (PAM)-3 and PAM-4 cabinets and installs new processors and displays. This upgrade disconnects the plant process computer (PPC) from the core exit thermocouples (CETs) and from the TMS and the Emergency Response Facility Data System (ERFDS) from the CETs. The PPC will be provided with the same data through 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.

Safety Evaluation Summary:
The new processor and display offer enhanced features such as automatic self-test and diagnostics, greater flexibility, improved operator interface, 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.

The new TMS/RVLIS architecture includes features to mitigate the effects of the unlikely event that either or both trains experience halt of a required application or lockup of a processor - failures that could delay access to information used by the operators to assess accident progress, and thus secondarily affect accident recovery and consequences. Following such an event, the operators will be alerted to the failure condition and the system will restart automatically without intervention. The existing systems do not restart automatically. In the extreme case that both trains fail due to a common cause and the automatic restart feature fails in both trains, limited data will continue to be available from the nonsafety-related subcooled margin monitor (SCMM), which is not affected by this upgrade.

Software changes in the PPC, Validyne server, and the ERFDS are performed in accordance their respective SQA plans 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 cannot increase the probability, consequences, or possibility of an accident or malfunction previously evaluated in the SAR. Margin of safety, as discussed in any TS, is not affected because the RVLIS and TMS have no protection or actuation function. This activity does not result in a USQ.

00-060 Unit 1 Power Uprate Design Change

Reference Document No.: DCP N-049516
Rev. No: 0
Reference Document Title: Unit 1 Uprate
Safety Evaluation Description:
The Unit 1 uprate DCP does not require any physical modifications of plant equipment and only requires the rescaling of several major plant operating parameters and the associated documentation changes to reflect the new Unit 1 uprated operating conditions. These changes are:

- Increase in the 100 percent core thermal power level from 3,338 MWt to 3,411 MWt.
- Increase in the 100 percent nuclear steam supply system (NSSS) thermal power level from 3,350 MWt to 3,425 MWt.
- Increase in the maximum nominal RCS Tavg from 576.6 °F to 577.3 °F.
- Increase the program full power RCS Tref from 572.76°F to 573.49°F.
- Increase in the full power pressurizer level from 59.8 percent to 60.7 percent.
- Revise the high pressure (HP) turbine full power first stage pressure

from 540.5 psia to 558.0 psia.

- Revise the power level at which the programmed digital electrohydraulic (DEH) control of the turbine governor valves is transferred from full arc admission to 75 percent arc admission.

Safety Evaluation Summary:

The Unit 1 uprate DCP does not require any physical modifications of plant equipment and only requires the rescaling of several major plant operating parameters and the associated documentation changes to reflect the new Unit 1 uprated operating conditions. The Unit 1 uprate DCP is comprised of two types of parameter rescalings, those that require prior NRC approval prior to implementation and those that are being implemented in accordance with 10 CFR 50.59 in order to maintain thermal efficiency at the uprated conditions.

The rescaling of the maximum nominal RCS Tavg from 576.6°F to 577.3°F and the associated full power change in pressurizer level from 59.8 percent to 60.7percent and the defined new core thermal power level of 3,411 MWt require prior NRC approval for implementation. Therefore, the implementation of these revised values must be authorized by the NRC approval established in the SER and LA issued for the Unit 1 uprate LAR 99-03.

The rescaling of the program RCS Tref from 572.76 °F to 573.49 °F, the HP turbine full power first stage pressure from 540.5 psia to 558.0 psia, and the increase in NSSS rated power from 3,350 to 3,425 MWt are changes which do not impact any safety analysis or margin of safety and may be implemented via 10 CFR 50.59 as documented in this LBIE. These parameters are being rescaled to reflect the revised RCP heat input calculation and to maintain the Unit 1 thermal efficiency.

In addition, in order to maximize the reliability of the first stage HP blades, the current power level of 30 percent at which the DEH program is transferred from full arc (single valve) admission to 75 percent arc admission (sequential valve) will be revised to an appropriately higher power level.

Conclusion:

In summary this LBIE provides two conclusions.

1. The NRC SER and LA for LAR 99-03 will constitute prior NRC approval for the increased core thermal power license limit from 3,338 to 3,411 MWt, the increase in maximum design RCS Tavg from 576.6 to 577.3 °F, and the increase in the full power pressurizer level from 59.8 to 60.7 percent as established in the Unit 1 uprate DCP N-049516.
2. All other changes implemented in accordance with DCP N-049516 (RCS Tref, HP 1st stage pressure, NSSS rating, and turbine control

valves) do not impact any safety analyses or margins of safety and may be implemented in accordance with 10 CFR 50.59 since they do result in a USQ.

The acceptability of this LBIE is contingent upon receipt of an NRC SER and LA as requested in LAR 99-03.

00-061 Containment Isolation Valve List

Reference Document No.: AD13.DC1, Attachment 7.7
Rev. No: 11
Reference Document Title: Containment Isolation Valves
Safety Evaluation Description:
In order to make the successful RHR flush performed in Unit-1 in June 2000 under TP TO-0003 a permanent option for either unit, Note 5 is being added to AD13.DC1, Attachment 7.7, for valves 8885A & 8885B. This will allow these valves to be opened under the control of OP B-2:IX, a permanent plant procedure, which has all the same provisions for assuring ECCS flowpath separation and maintaining critical ECCS parameters as did TP TO-0003.

This 10 CFR 50.59 Safety Evaluation is also being performed as required by AD13.DC1, step 5.16.1, since any change to this list constitutes a change to the facility. Note 4 is also being added to AD13.DC1, Attachment 7.7, for those 5 remote-manual valves identified in FSARU Table 6.2-39 as requiring administrative controls and to be treated the same as sealed-closed valves. These valves are 8823, 8824, 8843, 8885A and 8885B. This change simply transfers the requirement from FSARU Table 6.2-39, note 22, to AD13.DC1. Also, asterisks are being added for 6 vent valves to authorize opening these valves in Modes 1 through 4, on an intermittent basis, under administrative control in order to support the performance of STPs. These 6 valves are RHR-937, SI-114, SI-115, VAC-79, VAC-80, and VAC-81. These valves were not previously in the AD13.DC1, Attachment 7.7, prior to ITS implementation, but all are opened routinely under associated STPs at power.

Safety Evaluation Summary:
<u>Addition of Note 5 associated with RHR Flush:</u>
OP B-2:IX provides instructions for flushing certain portions of the RHR system between the RHR pumps and containment to reduce radiation levels in the vicinity of RHR components. This procedure may be performed in Modes 1 through 3 in either unit.

This activity places the plant in an abnormal alignment for Modes 1-3 that involves connection of the RHR system to the SI test lines and flow through containment penetrations 24, 25, and 51B to the LHUTs. RHR suction will be provided by the RWST. RHR piping will be flushed until dose rates in the vicinity of the RHR system drop to acceptable levels, as determined by radiation protection personnel, while control room operators monitor RWST and LHUT levels. The procedure suggests that unless indicated otherwise, a 1 percent rise in LHUT level is to be used as a guideline for the flush. 1 percent of a LHUT represents about 830 gallons. RWST level will be monitored during the flush to assure this tank never drops below the minimum 93 percent required by SR 3.5.4.2. Each 1 percent level in the RWST equates to approximately 4,500 gallons. RWST level is typically maintained at approximately 97 percent. Radiation protection personnel will also be monitoring radiation levels, as needed, in the vicinity of RHR components during the flush.

There will be no impact to achieving the required containment recirculation sump level of 92.5 feet required for switchover from the ECCS injection phase to cold leg recirculation in accordance with EOP E-1.3, "Transfer to Cold Leg Recirculation."

While this evolution is performed, containment isolation valves SI-8885A, SI-8885B (penetrations 24 and 25) will be opened under administrative control. These are remote-manual valves that are treated as sealed-closed valves as specified in FSARU Table 6-2.39. TS LCO 3.6.3, "Containment Isolation Valves," allows locked or sealed-closed valves to be opened on an intermittent basis under administrative control. In accordance with TS and site administrative procedures, the opening of locked or sealed-closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing a person who is in constant communication with the control room at the valve controls, (2) instructing this person to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The FSARU Chapter 15 Liquid Holdup Tank Rupture accident (FSARU Sections 15.4.8 and 15.5.25) assumes complete filling of an LHUT (83,220 gallons) at a rate of 120 gpm from the RCS following operation for an equilibrium core cycle with 1 percent defective fuel. A ruptured LHUT under these conditions will result in a release at the site boundary of approximately ½ percent of the 10 CFR 100 limits. The design-basis accident RCS activity level (microcuries per gram) is approximately 40 times the activity that existed in June 2000 in Unit-1 in the portion of the RHR piping with elevated dose rates. The LHUTs are designed for a maximum fill rate of 550 gpm (Ref. DCM S-8). Also the fill rate for the LHUT using the flowpaths described in OP B-2:IX of 25-35 gpm is small compared with the 120 GPM accident analysis rate above. Also, the 2,000 - 3,000 gallons expected from OP B-2:IX will have no overall impact on an LHUT. Therefore, as a result of the activity being significantly lower than accident conditions and the relatively low volume of water being transferred to the LHUTS, this activity does not increase the consequences of an accident previously evaluated.

Addition of Note 4 for Remote Manual Valves Identified by Note 22 in FSARU Table 6.2-39:
The requirement to remove the fuses and administratively control remote-manual valves 8823, 8824, 8843, 8885A, and 8885B has been met by writing an administrative clearance for these valves and removing their control power fuses while the plant is at power. Currently Clearances 61964 (U-1) and 63388 (U-2) meet this requirement. Adding this note to AD13.DC1 makes this FSARU requirement more visible to Operations. This change involves restating a requirement already addressed in the FSARU.

Adding Asterisks (*) for Vent Valves RHR-937, SI-114, 115, VAC-79, 80 and 81:
Asterisks are being added for 6 vent valves to authorize opening these valves in Modes 1 through 4 on an intermittent basis under administrative control in order to support the performance of STPs. These six "containment isolation valves" were not in the AD13.DC1 CVI list prior to ITS implementation. The original list of asterisked valves consisted of 22 manual or remote-manual valves and was located in TS 3.6.3. It consisted of process flow-isolation valves only, no vents or drains, but it did include one flow isolation valve, SI-161on Pen 51B. Opening these valves was allowed at power for testing, maintenance, or other activities, as long as the provisions of administrative control are followed. In 1992, the CVI list was relocated to AD13.DC1 in accordance with LAs 73/72. One additional process-flow CVI, MS-902 (SG nitrogen supply), was added to the list in 1995. The basis for adding this valve to the list was to allow normal letdown to be reestablished on loss of instrument air to containment by use of Abnormal Operating Procedure, AP-9. Non Conformance Report (NCR) N0002036 (July 1997) was the result of an NRC inspector's question about whether a normally-closed root valve was included in the monthly surveillance required by TS 4.6.1.1. This NCR resulted in DCPP expanding the term "containment isolation valve" to include all vents and drains on most penetrations, which included all normally-closed valves on closed systems inside containment. It did not include penetration vents and drains outside containment on systems required to be in service post accident. The current list of CVIs in AD13.DC1, approved for ITS implementation, includes all penetration boundary valves consistent with this expanded definition of "containment isolation valves". Opening VAC-79, 80, and 81 at power is required by TS SR 3.6.3.7 to perform surveillances.

Conclusion:
Components being used for this RHR flush may be called upon during a design-basis accident (DBA). The minimum RWST and containment recirculation sump levels required for a DBA would be maintained and would not be challenged by the flow to the LHUTs, based on a dedicated operator monitoring RWST level and automatic closure of containment penetration 52B isolation valves at the initiation of a DBA. Therefore, containment recirculation sump level is not compromised. All SSCs involved would be available to perform their intended safety functions. Removal of power from SI test line boundary valves removes required operator actions and the consequences of inadvertent operator actions. There is no impact to the ECCS LOCA flow path. There is no impact to the EOPs. There is no impact to previously evaluated accidents.

For the 6 vent valves that will be allowed to be opened intermittently under administrative controls, their STPs will have the OP O-12 controls added to them. OP O-12 revision is not required because it already covers manual containment isolation valves. This activity does not result in a USQ.

00-062 Lead Shielding Request TSR 00-0008

Reference Document No.: TSR 00-0008

Rev. No: 0

Reference Document Title: Lead Shielding Request for Shielding on Unit 1 Line 279 and 280 (RHR Supply to Spray Headers)

Safety Evaluation Description:
Temporary Shielding Request (TSR) 00-0008 will allow plant personnel to install and remove temporary lead shielding on lines 279 and 280 (RHR supply to spray headers 1,2,3,4) in the auxiliary building on the 100 foot elevation in the penetration area. This shielding is required for ALARA concerns for the crew performing work activities in the area and general area dose reduction. Piping has been seismically evaluated to acceptable design criteria along with the additional weight of lead blankets (Ref. Piping study calculation 8-110 study log 1098 and 8-117 study log 1099).

Safety Evaluation Summary:
The impact of lead shielding on the piping qualification, operation, and seismic interaction have been evaluated as acceptable.

Conclusion:
Based on the discussions in the LBIE, it is concluded that the installation of lead shielding on the piping as described in TSR 00-0008 is acceptable. The

lead shielding does not have any adverse impact on piping structural integrity, or operation, nor ant seismic interaction with any targets in the vicinity. This activity does not result in a USQ.

00-063 Removal of CWP Trip from ICW

Reference Document No.: DCP E-49524
Rev. No: 0
Reference Document Title: Removal of CWP Trip from ICW
Safety Evaluation Description:
The automatic circulating water pump (CWP) trip from low intake cooling water (ICW) pressure is being eliminated by this design change. Operators will manually initiate a trip of a CWP on loss of ICW flow, based on the existing low-pressure alarm, the stator high temperature alarm, and PPC motor temperature indication.

Safety Evaluation Summary:
Deletion of the CWP on low ICW pressure will not change any safety analysis in the FSARU. There is no safety function for the CWP trip on low ICW pressure. It was installed only for protection of the Class II CWP motor. The ICW low-pressure alarm (which will not be deleted), stator temperature alarm, and temperature indication on the PPC provide sufficient information to determine if a CWP needs to be manually tripped. This manual action will provide sufficient protection for the CWP motors.

Conclusion:
There is no safety impact from deleting the CWP trip from low ICW pressure. The reference to the CWP trip on low ICW pressure in the FSARU is descriptive only. This activity does not result in a USQ.

00-064 ASW Pump Discharge Vacuum-Relief Valves

Reference Document No.: TS Bases 3.7.8
Rev. No: 0
Reference Document Title: Auxiliary Saltwater System (ASW)
Safety Evaluation Description:
This change revises the Bases for TS 3.7.8, "Auxiliary Saltwater System," to add additional information regarding the requirements for operability of an ASW train. The change discusses how the operability of the ASW pump discharge vacuum-relief valves affects the operability of the ASW system.

Each ASW train has a vacuum-relief system consisting of two vacuum-relief valves (check valves) that function to prevent water hammer in the system piping during an ASW pump trip and restart transient. At least one vacuum-relief valve must be operable for the associated ASW train to be operable. Check valves are passive components and, unless otherwise specified, are not required to be considered in meeting the single-failure criterion. The second vacuum-relief valve on each header ensures reliability of the function. If both vacuum-relief valves on a single header are inoperable, water hammer during an ASW pump trip-and-restart transient could affect both ASW trains, unless the ASW header cross-tie valve is closed and the ASW pump breaker or direct current (dc) control power switch is opened for the affected ASW train, precluding the potential for water hammer in the train.

Safety Evaluation Summary:
The vacuum-relief valves are considered passive components with respect to the single failure criterion, and a failure of a vacuum-relief valve to function need not be postulated. Therefore, at least one ASW pump discharge vacuum-relief valve is required to be operable for the associated ASW train to be operable. This will ensure that in the event of a single failure coincident with a design-basis event that one ASW train will be operable and able to perform its safety function. If both ASW vacuum-relief valves on one ASW train are inoperable, water hammer during an ASW pump trip-and-restart transient could affect both ASW trains, unless the ASW header cross-tie valve is closed and the ASW pump breaker or dc control power switch is opened for the affected ASW train, precluding the potential for water hammer in the train. These actions, which limit the impact to a single train, are consistent with actions taken to comply with TS 3.7.8 for other failures affecting the operability of a single train.

This Bases change is consistent with the plant design and licensing bases, and consistent with current plant operation. There is no change to the design, function, or operation of the ASW pump discharge vacuum-relief valves, or of the ASW system.

Conclusion:
This change, by describing how the operability of the ASW pump discharge valve affects ASW train/system operability, provides assurance that the ASW system will be operable and able to perform its required function during anticipated operational occurrences and postulated accidents. This activity does not result in a USQ.

00-065 Requirements for Operability of ASW Pump Discharge Vacuum-Relief Valves
Reference Document No.: ECG 17.4
Rev. No: 0

Reference Document Title: ASW Pump Discharge Vacuum-Relief Valves
Safety Evaluation Description:
This change creates a new ECG which establishes a limiting condition for operation, action requirements, and SRs for operability of the ASW pump discharge vacuum-relief valves under the controls of the ECG program specified in Administrative Procedure OP1DC16, "Control of Plant Equipment Not Required by the Technical Specifications."

Safety Evaluation Summary:

Each ASW train has a vacuum-relief system, consisting of two vacuum-relief valves (check valves) on each ASW supply header, which functions to prevent water hammer in the ASW system piping during an ASW pump trip-and-restart transient. Vacuum-relief is necessary to assure ASW system integrity and operability of the ASW system. The operability of the ASW system ensures that sufficient cooling capacity is available for continuous operation of safety-related equipment during normal and accident conditions.

The vacuum-relief valves are considered passive components with respect to the single failure criterion, and a failure of a vacuum-relief valve to function need not be postulated. Therefore, at least one ASW pump discharge vacuum-relief valve is required to be operable for the associated ASW train to be operable. This will ensure that in the event of a single failure coincident with a design-basis event that one ASW train will be operable and able to perform its safety function. The ECG limiting condition for operation requires that two ASW pump discharge vacuum-relief valves be operable on each ASW pump discharge header to provide redundancy, so that at least one ASW pump discharge vacuum-relief valve is operable to prevent water hammer in the system piping during an ASW pump trip-and-restart transient.

If one vacuum-relief valve in either ASW train is inoperable, it shall be restored to operable status within 7 days. Separate condition entry is allowed for each ASW train, because a single inoperable vacuum-relief valve would not impact the other train. If both vacuum-relief valves in one ASW train are inoperable, action is to be taken immediately to either isolate the impact to one train by closing the cross connect valve and preventing the associated ASW pump from automatically starting, or declare both trains inoperable. Action shall then be taken in accordance with the TS for either one or both trains inoperable.

Conclusion:
This change creates a new ECG which establishes a limiting condition for operation, action requirements, and SRs for operability of the ASW pump discharge vacuum-relief valves under the controls of the ECG program

specified in Administrative Procedure OP1DC16. The requirements are consistent with the plant design and licensing bases and consistent with current plant operation. There is no change to the design, function, or operation of the ASW pump discharge vacuum-relief valves or of the ASW system. The requirements provide additional assurance that the ASW pump discharge vacuum-relief valves will be maintained operable and that the ASW system will be operable and able to perform its required function during anticipated operational occurrences and postulated accidents. This change does not result in a USQ.

00-066 Storage of Flammable Liquids in Plastic Bottles

Reference Document No.: FHARE 144

Rev. No: 0

Reference Document Title: Storage of Flammable Liquids in Plastic Bottles

Safety Evaluation Description:

The DCPP Fire Protection Program uses the National fire Protection Association (NFPA) Codes for guidance. In accordance with NFPA 30, "Flammable and Combustible Liquids Code," a flammable liquid must be stored in an "approved" container. Presently, no small approved containers are available, therefore, small quantities (up to 16 ounces) of flammable liquids are being stored in plastic containers that are not approved. FHARE 144 evaluates the use of plastic containers for storage of flammable liquids.

Safety Evaluation Summary:

NFPA 30 requires the use of "approved" storage containers for flammable liquids. However, upon further investigation into the code, Chapters 4 and 5 both allow for the "incidental use and storage" of flammable liquids. This allows small quantities of these liquids to be used and stored for performing tasks that are subordinate to the primary occupancy classification. Therefore, the process used to store small quantities (16 ounces or less) of flammable liquids in high-density plastic bottles at DCPP meets the intent of NFPA 30, which is our commitment. The fire hazard due to flammable liquids and other combustible materials was evaluated in the plant fire hazard analysis. This evaluation documents that ample fire protection features are available throughout the plant to mitigate the effects of a fire and ensures that safe shutdown can be achieved and maintained.

Conclusion:

The process used to store small quantities of flammable liquids in plastic bottles at DCPP meets the intent of NFPA 30 and is safe. No new fire hazards have been created. No new ignition sources have been introduced to the plant. The results of the fire protection safe-shutdown analysis have not been impacted. This change does not increase the risk that a fire will damage other equipment important to the safe and reliable operation of the plant. This activity does not result in a USQ.

00-067 New Fuel Elevator Modification

Reference Document No.: DCP N-49494
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, nozzle, and other parts 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 proposed change limits 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 (SFP). 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 SFP surface.

The design features are similar to that required for the SFP bridge hoist that are specified in the design-bases discussion in the FSARU.

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 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 FSARU. The fuel repair work is to be performed on one assembly at a time and in conjunction with no other fuel movement within the SFP or the containment. The shielding provided by the SFP 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 that 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 leads to the conclusion that the probability of the fuel-handling accident remains "very small," as described in Section 15.4.5.1 of the FSARU. 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 USQ is created by the use of the modified new fuel elevator to handle a spent fuel assembly for fuel repair.

00-069 TS Bases SR 3.5.2.3

Reference Document No.: TS Bases SR 3.5.2.3
Rev. No: 0
Reference Document Title: TS Bases SR 3.5.2.3
Safety Evaluation Description:
Eliminate the parenthetical qualification regarding checking full ECCS pump casing vents (for non-running pumps) because the status of pump operation is not a concern for checking full using ultrasonic testing. This change will eliminate cycling of ECCS pumps during TS 3.5.2.3 surveillance activities as long as suitable testing methods are used.

Safety Evaluation Summary:
The TS SR 3.5.2.3 Bases read, "...alternates to venting the accessible system high points, can be employed to provide this assurance, such as ultrasonic testing the vent lines of the ECCS pump casings (for non-running pumps) and accessible high point vents."

Review of original ITS Bases submittals, NRC requests for additional information, and interviews with personnel involved found that DCPP added the words, "such as...." of our own accord, without NRC prompting, specifically to allow use of ultrasonic testing to verify full. There are no notes nor recollections of involved personnel of what prompted DCPP to add the parenthetical note, except that there had been contamination experiences venting running pumps and venting running CCPs was not required due to the self-venting design (i.e., the running pump was always considered to be full). DCPP subsequently changed the STP to shut down and vent a running CCP after the NRC clarified that the TS in effect at that time required venting the pump (the old TS did not allow alternate means of verifying that the system was full of water).

Some methods of verifying the vent lines full could be impaired by pump operational status, but ultrasonic testing is a method that is conservatively affected. Ultrasonic testing of a vent line containing a frothy gas-water mixture or all gas will yield a distinctly different signal than that of a water-filled vent line, resulting in the operator being required to shut the pump down and vent the location.

Conclusion:
It is acceptable for ECCS pumps to continue running when ultrasonically testing the vent lines to verify the pump and piping is full. This activity does not result in a USQ.

00-070 Change to ECG 7.8 Bases, Accident Monitoring Instrumentation

Reference Document No.: ECG 7.8
Rev. No: 1
Reference Document Title: Accident Monitoring Instrumentation
Safety Evaluation Description:
ECG 7.8 provides requirements for the accident monitoring instrumentation. SR 7.8.3 requires that a channel calibration be performed every 24 months on the RCS subcooling margin monitor, power operated relief valve (PORV) position indicator (temperature element), and safety valve position indicator (temperature element) channels. This change revises the Bases for SR 7.8.3 to include additional information on performing the channel calibration on these channels. This change is a recommended enhancement to the ECG 7.8, SR 7.8.3.

Safety Evaluation Summary:
The proposed change is administrative because it only enhances the Bases for ECG 7.8, SR 7.8.3, to provide additional information regarding the performance of channel calibrations for those accident monitoring instrumentation channels that include resistance temperature detectors (RTDs) and/or thermocouple sensors. The added information is consistent with the TS definition of "channel calibration" included in TS 1.1. The change does not revise ECG requirements for operability of these channels, nor does it change the design, operation, or method of testing the channels as specified in plant procedures.

Conclusion:
Based on the considerations discussed above, the change is acceptable, and there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Based on the evaluation in LBIE Section 1, a USQ is not involved.

00-071 Centrifugal Charging Pump Backup Firewater Cooling AOT and Inspection Changes

Reference Document No.: ECG 8.3
Rev. No: 1
Reference Document Title: Centrifugal Charging Pump (CCP) Backup Firewater Cooling

Safety Evaluation Description:
ECG 8.3 controls the operability of backup firewater cooling for the CCPs. In accordance with probabilistic risk assessment (PRA) Calculation C-9 R6, the completion time for Required Action A.2 is being reduced from 30 days to 72 hours, and SR 8.3.1 is being changed from 31 to 7 days. Other changes include:

- The conditions, and corresponding actions, are being condensed from three conditions to one, since the completion time is essentially the same regardless of the number of CCPs that firewater cooling can supply.
- Three references are being added to the Reference section.
- The ECG 8.3 LCO, Applicability, and Actions are being changed in order to be consistent with the changes above.

The ECG 8.3 Background and Applicable Safety Analyses sections will reflect the PRA bases for the completion time and surveillance frequency changes.

Safety Evaluation Summary:
A PRA calculation found backup firewater cooling to be a significant factor in preventing a LOCA caused by a loss of RCP seal injection. The PRA calculation recommended a decrease in the completion time for restoration of backup firewater cooling and an increase in frequency of the surveillance that inventories the fire hoses necessary to install backup firewater cooling. These changes provide additional controls and restrictions to assure that the CCPs are available if CCW and ASW are lost, by assuring that firewater cooling is available to the CCPs.

Conclusion:
The ECG 8.3 revision will increase the reliability of CCP backup firewater cooling. This revision does not impact any commitments contained in the DCPP licensing basis and does not result in a USQ.

00-072 Performance of Preplanned EDG Preventive Maintenance in Modes 1, 2, 3, and 4

Reference Document No.:
Rev. No: 0
Reference Document Title: Performance of Preplanned EDG Preventive Maintenance in Modes 1, 2, 3, and 4

Safety Evaluation Description:

This change revises PCD commitment T36047 in order to allow preplanned preventive maintenance (e.g., those inspections recommended by the manufacturer), which requires the incapacitation of an EDG, to be performed in Modes 1, 2, 3, and 4 in accordance with the maintenance rule and the TS.

This change revises ECG 21.3, "Miscellaneous EDG Functions," Revision 0, to delete the Note modifying ECG SR 21.3.3 which states: "This Surveillance shall not be performed in MODES 1 and 2." SR 21.3.3 requires that "each EDG be subjected to an inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service." The change also deletes reference to performance during refueling and adds a restriction on the mode(s) in which the inspection can be performed, if post-maintenance testing requires performance of a TS SR.

This change revises TS Bases 3.8.1, "AC Sources - Operating," Revision 0, for SRs 3.8.1.8, through 3.8.1.14, and 3.8.1.16, through 3.8.1.19 to explicitly state the modes in which preplanned maintenance shall be performed, if the SR is required to be performed to demonstrate operability after completion of the preplanned maintenance.

Safety Evaluation Summary:

<u>LAs 44 and 43, and ECG 21.3</u>

LAs 44 and 43, for Units 1 and 2, respectively, dated October 4, 1989, revised the DCPP TS to change the EDG AOT from 72 hours to 7 days. In its safety evaluation, the NRC concluded that the risk associated with plant operation with a 7 day AOT was acceptable, because the baseline risk (no maintenance) associated with plant operation with a 7 day AOT and 6 EDG configuration (2.017E-04/yr CDF) was the same as or less than the risk associated with plant operation with a 72 hour AOT and 5 EDG configuration (2.078E-04/yr CDF). Prior to installation of the sixth EDG, this change (the increase in AOT from 72 hours to 7 days) would apply only to the swing EDG for performance of preplanned preventive maintenance. After the sixth EDG was installed and operational, the 7 day AOT would apply to all EDGs for unplanned maintenance only. The CDF associated with plant operation with a 7 day AOT and 6 EDG configuration is 9.6999E-5/yr for EDG on-line maintenance (based on 1997 PRA model).

PG&E committed, with the addition of the sixth EDG, to perform preplanned preventive maintenance that requires the incapacitation of an EDG only during cold shutdown and/or refueling outages. The NRC safety evaluation for LAs 44 and 43 states that performance of preplanned maintenance only during plant shutdown and/or refueling met the guidelines of RG 1.93, "Availability of Electric Power Sources," and was acceptable. RG 1.93 describes operating procedures and restrictions acceptable to the NRC staff that should be implemented if the available electric power sources are less than required by the LCO. In addressing preplanned preventive maintenance, RG 1.93 states, "The operating time limits delineated above (in the RG) are explicitly for corrective maintenance activities only. The operating time limits should not be construed to include preventive maintenance activities which required the incapacitation of any required electric power source. Such activities should be scheduled for performance during cold shutdown and/or refueling periods." This position was consistent with the philosophy that TS LCOs and associated actions were, in part, developed to address random failures of plant SSCs, and to provide a reasonable time to effect repairs before plant shutdown was required. This commitment is reflected in the ECG 21.3 SR 21.3.3 note and in the ECG Bases.

More recently, TS limiting conditions for operations and associated actions have been used by nuclear power plant licensees for the deliberate removal of SSCs from service to perform on-line maintenance. The benefits of performing maintenance activities during power operations include increased system and plant reliability, reduction of plant equipment and system material condition deficiencies that could adversely impact plant operations, and reduction of work scope during plant refueling outages. However, if maintenance is performed at power without proper controls and careful consideration of risk, overall risk could be significantly increased because of the reduced capability to mitigate the consequences of an accident or transient compared to the risk that occurs from expected random equipment failures.

<u>Maintenance Rule</u>
The removal of SSCs from service to perform on-line preventive maintenance, and specific criteria for the acceptability of performing such maintenance, is addressed in 10 CFR 50.65. 10 CFR 50.65 was recently revised to add 10 CFR 50.65(a)(4), which requires:

> "Before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk

that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety." [64 FR 38551, July 19, 1999, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"].

This requirement is effective November 28, 2000, and becomes legally binding on DCPP at that time.

The intent of 10 CFR 50.65(a)(4) is to require that licensees perform assessments before maintenance activities are performed on SSCs covered by the maintenance rule, and to manage the increase in risk that may result from the proposed activities.

Using the 1997 PRA model, EDG 1-2 is the most limiting from a CDF perspective. It can be removed from service for 145 hours (about 6 days) per year and be risk insignificant (risk insignificant means there is a <E-6 increase in the baseline CDF). If the 1-2 EDG is taken out of service for the full TS allowed 7 days (168 hours), the CDF would increase by approximately 1.16E-6/yr above the of baseline CDF of 9.6999E-5/yr, in which case risk management actions must be established in accordance with the maintenance rule. This is still less than the CDF the NRC used as a benchmark in LA 44 and 43 (i.e., 2.078E-04/yr CDF for a 72 hour AOT for unplanned maintenance with zero preplanned maintenance on the shared EDG for the 5 EDG configuration).

<u>Technical Specifications</u>
Performing the assessment required by 10 CFR 50.65(a)(4) does not relieve DCPP from compliance with its operating license (including the TS) and other applicable regulatory requirements. TS LCO 3.8.1, "AC Sources -Operating," requires that three EDGs, capable of supplying the onsite Class IE power distribution subsystem(s), be operable. With one EDG inoperable because of the maintenance, Required Action B.4 requires that the inoperable EDG be restored to operable status within 7 days and 10 days from discovery of failure to meet the LCO. If the inoperable EDG cannot be restored within that Completion Time, the plant will be placed in a Mode in which the LCO does not apply. While in the required action, a single failure of another SSC is not required to be postulated. Therefore, the remaining two EDGs are assumed to be available to supply power to at least one complete train of ECCS equipment to mitigate the radiological consequences of an accident that could result in potential offsite exposures.

<u>Station Blackout</u>
Removing one EDG from service in order to do preplanned maintenance will

not have an impact on the reliability assumed in the station blackout (SBO) analysis. The SBO analysis for DCPP assumes the loss of all offsite power, concurrent with the failure of two EDGs on one unit. The DCPP SBO analysis considered the reliability of the EDGs to be greater than or equal to 0.95. Performance criteria established in accordance with the maintenance rule will assure that reliability and availability assumptions for the EDGs are maintained.

Summary

Based on the above, the proposed changes to commitment T36047 to allow preplanned maintenance on an EDG to be performed in accordance with maintenance rule requirements, to ECG 21.3 to delete the mode restrictions for SR 21.3.3, and to the TS 3.8.1 Bases to provide restrictions on the modes in which preplanned maintenance can be performed, are acceptable. Compliance with the maintenance rule (assessment of the risk and management of the risk associated with the performance of preplanned preventive maintenance on an EDG) and compliance with the TS will ensure that performance of preplanned preventive maintenance on an EDG in Modes 1,2, 3, and 4 will be appropriately controlled, that the risk associated with performing such maintenance is not significantly increased, and that the risk is managed.

Conclusion:

Based on the considerations discussed above, there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The performance of preplanned preventive maintenance on an EDG in Modes 1, 2, 3, and 4 will be conducted in accordance with the DCPP operating license (including the TS) and the applicable regulations, and no changes to these requirements are necessary.

Based on the analysis in LBIE Section 1, a USQ is not involved.

00-074 Reduced Power Level for High Swell Conditions

Reference Document No.: OP O-28 / OP AP-7

Rev. No: 7/24

Reference Document Title: Intake Management / Degraded Condenser

Safety Evaluation Description:

This LBIE evaluates revisions to operating procedures OP O-28 and AP-7. These procedures are being revised to enhance unit availability by operating at a recommended reduced power level of 25 percent during periods of increased potential "kelp attacks." This will minimize the possibility of a reactor trip in the event that the condenser becomes unavailable.

Previously, these procedures only discussed power level in terms of the P-9 setpoint. OP O-28, step 6.4.10 recommended reducing power below the P-9 setpoint for high swell conditions, while AP-7 used the P-9 setpoint power level as the decision basis for initiating a reactor trip for degraded condenser conditions.

The P-9 setpoint is being raised from the compensatory action value of 15 percent to its original licensed value (in accordance with LAs 30/29) of 50 percent. However, operating at this higher P-9 value of 50 percent would be expected to result in a direct challenge to the pressurizer PORVs and main steam safety valves (MSSVs) following a loss of the condenser. Evaluations of plant response indicate that no challenges to the PORVs and/or MSSVs would be expected at power levels at or below 25 percent. Therefore, these procedures are being revised to ensure that a reduced power level of 25 percent is recommended for high swell conditions to provide assurance that a controlled shutdown can be accomplished and that the shutdown would not challenge the pressurizer PORVS or the MSSVs, in the event the condenser becomes unavailable.

Safety Evaluation Summary:
The reduced power level has been selected to provide assurance that a controlled shutdown can be performed without challenging the pressurizer PORV or MSSVs in the event the condenser becomes unavailable. This change is being incorporated into operating procedures as a recommended power level for specific plant conditions. This LBIE establishes that this mode of operation does not impact the existing P-9 licensing basis, nor does it represent a new condition or mode of operation which must be evaluated, since it remains conservatively bounded by the existing DCPP safety analyses.

Conclusion:
In summary, this LBIE concludes that establishing a recommended reduced power level in accordance with OP O-28 and AP-7 is not required by, and does not impact, any DCPP design or licensing bases, nor does it create any new operating conditions or scenarios not already analyzed, and therefore, these procedure changes do not constitute a USQ.

00-076 Containment Isolation Valve List
Reference Document No.: AD13.DC1
Rev. No: 12
Reference Document Title: Containment Isolation Valves
Safety Evaluation Description:

Modify the Containment Isolation Valve list, Attachment 7.7 by:

a. Removing the asterisks (*) from the table for selected manual or remote-manual valves.
b. Adding a general note that allows administrative control under OP O-12 for any normally-closed, manual, containment isolation valve.
c. Adding K-10B2 to Surv/Ops Sealing Procedure column for all valves covered by that procedure and deleting the reference to STP I-1D.
d. Correcting the unit identifier for Unit-2 Valves 9354B, 9355A/B, and 9356A/B.

Safety Evaluation Summary:
To allow operations to apply TS 3.6.3 consistently, AD13.DC1, Attachment 7.7, is being revised to eliminate asterisk (*) statements that allowed intermittent opening under administrative control to a limited number of containment isolation valves and to add a general note to allow intermittent opening of any normally-closed, manual, or remote-manual, containment isolation valve as allowed by TS 3.6.3.

Conclusion:
This change does not constitute a USQ, nor does it have any effect on offsite dose limits defined in 10 CFR 100. This change has no licensing-basis impact on DCPP.

00-078 Lead Shielding In Accordance with DCP P-49542

Reference Document No.: P-49542
Rev. No: 0
Reference Document Title: Lead Shielding Request for Shielding on Unit 1 RHR Piping Nest at El. 100 Feet in Aux. Bldg.
Safety Evaluation Description:
The design change under DCP P-49542 will allow personnel to install temporary lead shielding on RHR piping nest on the 100 foot elevation in the Aux. Bldg. penetration area. This shielding is required for ALARA concerns for the crew performing work activities in the area and general area dose reduction. Piping has been seismically evaluated to acceptable design criteria along with the additional weight of lead blankets (Ref. Piping Calculation 8-103 rev. 12 and M8-103 rev. 6)

Safety Evaluation Summary:
The impact of lead shielding on the piping qualification, operation, and seismic interaction have been evaluated as acceptable.

Conclusion:
Based on the discussions in this LBIE, it is concluded that the installation of lead shielding on the piping as described in DCP P-49542 is acceptable. The lead shielding does not have any adverse impact on piping structural integrity or operation, nor any seismic interaction with any targets in the vicinity. This activity does not result in a USQ.

00-080 Steam Generator Pressure Pulse Cleaning with Scale Conditioning Agents

Reference Document No.: MRS-SSP-1119-PGE/PEG
Rev. No: 0
Reference Document Title: Steam Generator Pressure Pulse Cleaning with Scale Conditioning Agents
Safety Evaluation Description:
This LBIE addresses DCPP Unit 1 SG Pressure Pulse Cleaning (PPC) with coincident application of Scale Conditioning Agent (SCA) to the SG secondary side fill water. The PPC/SCA activity is to be performed on SGs 1-2 and 1-3 only during the 1R10; however, this LBIE is intended to be applicable to all DCPP Unit 1 SGs.

PPC/SCA is a maintenance process performed as part of the refueling outage SG tube-bundle cleaning activities. The process is designed to remove scale and deposits from the SG tubes and the tube support plates, to provide increased heat transfer, higher steam pressure, and greater generation.

Safety Evaluation Summary:

This LBIE is based upon the detailed evaluation of the proposed PPC/SCA process provided in Westinghouse SECL-00-124.

Process and Hardware Description
The PPC system introduces a volume of high-pressure nitrogen gas into the water-filled secondary side of the SG through 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 break up and dislodge the scale and deposits residing on the SG tubes and on the tube support plates (TSP). An SCA has been added to the SG fill water to increase the efficacy of the PPC process by softening and partially dissolving the scale and deposits. The PPC process is implemented in the SG when the SG secondary side is filled, or in the process of being filled and/or drained, with water containing SCA. PPC is followed by the sludge-lancing process which removes the dislodged scale and deposits from the SG.

Scale Conditioning Agent
The SCA to be used in the DCPP SGs contains Free EDTA, Ascorbic Acid, Triethanolamine, and a surfactant. The SCA formulation has been evaluated by test and found acceptable for effects on SG tube bundle and structural materials. A post-PPC/SCA passivation treatment, using a hydrazine soak (a normal part of the post-outage startup), will be used to preclude the potential for oxidized-copper species to coat the heat transfer tubes on plant startup. The SCA will be introduced to and discharged from the SGs according to a fill/drain schedule that is based on internal tube support plate elevations, and pulsing will be performed as appropriate to each fill level. Following the PPC/SCA activity, the SG will be refilled with demineralized water and pulsed during the drain cycle. Sludge-lancing activities will follow the PPC/SCA in order to remove as much dislodged scale as possible.

The SCA has been tested for potential effect on the magnetite deposits existing in the tube-TSP crevices. Results of these tests indicate there is no removal of these dent-causing deposits and, thus, no potential effect on the locked-tube bundle analysis.

Structural Evaluation
The effects of the PPC energy release into the SG secondary side have been evaluated with respect to the SG structure, including the shell, tube support plates, and tubes. All evaluations demonstrate acceptability of the process. Specific to the heat transfer tubes, 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 in the limiting, non-thinned tube was calculated to be 0.040 for pulsing at a 10-second interval for 58 hours, as specified in the Procedure. Calculated fatigue usage for previous Unit 1 PPC events is 0.0607, and fatigue usage for all other design events is 0.011. Thus, although PPC contributes significantly to total fatigue usage, the total, cumulative, fatigue usage including the 1R10 PPC is, at 0.1117, well below the limit of 1.0. For the limiting cold-leg thinned tube, total, cumulative, fatigue usage is 0.220. On this basis, future applications of PPC are judged acceptable.

Radiological and Safety Issues
SG tube secondary side deposits loosened and/or removed are potentially radioactive, thus normal ALARA practices will be followed throughout the PPC/SCA process. Openings to atmosphere in each SG are limited to that provided by the main steam line and the 10 percent dump valve. The motive nitrogen gas expelled via this route, once it has passed through the entire SG elevation and the steam dryers, has insufficient energy to carry water droplets

or particulates.

Conclusion:
The use of the PPC with SCA system in the DCPP Unit 1 SGs 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 result in a USQ.

00-082 Scaffold & Shielding Over RHR Recirculation Sump in Mode 1-4

Reference Document No.: A0515749
Rev. No: 0
Reference Document Title: Shielding For Unit 1 RHR Recirculation Sump
Safety Evaluation Description:
This LBIE applies to building scaffold inside the Unit 1 containment annulus in Modes 1 through 4 above the RHR recirculation sump to allow hanging lead shielding on the RHR pipes. There will be no planking installed on the scaffold, and the lead will not be installed until Mode 5. Building scaffolding at power, prior to placing RHR in service in Mode 4, will save an estimated 300 mRem for the job. This evaluation does not address the installation of the lead.

Safety Evaluation Summary:
Scaffold will be erected inside the Unit 1 containment annulus in Modes 1 through 4 above the RHR recirculation sump to allow later hanging lead shielding on the RHR pipes. This evaluation is based on the following precautions:

- There will be no planking installed on the scaffold.
- Lead will not be installed until Mode 5.
- The scaffold tubing will not have paint on it, only galvanizing.
- Personnel working above the sump will not carry any small objects capable of passing through the sump screen (no ty-wraps, pens, small objects).
- A Seismically Induced Systems Interaction Program (SISIP) qualified engineer will accompany the scaffold crew into containment during the erection.

Building scaffolding at power, prior to placing RHR in service in Mode 4, will save an estimated 300 mrem for the job.

ISSUES:
<u>Seismic Interaction</u>

The FSARU states that DCPP has evaluated the interaction of Design Class II equipment with Design Class I to ensure the continued operability of the Design Class I equipment. The SISI program at DCPP provides this assurance on a continuing basis and is appropriately applied to this scaffold erection. A SISIP qualified engineer will accompany the scaffold construction crew to verify that placement and bracing does not affect SISIP targets. A pre-, in-progress, and post SISIP evaluation will be performed by the engineer, and he will approve the structure at the end of the job. Additionally, this activity is in accordance with AD7.ID5 for scaffolding built inside SISIP-sensitive areas.

Hydrogen Generation

Because the total amount of galvanizing does not exceed FSARU analysis limits the scaffold erection does not present a hydrogen generation concern.

Dropped Hardware During Scaffold Erection

The concern about dropping poles or knuckles during erection is with the RHR sump and possibly putting a hole in the screens. The sump screens are adequately protected by a layer of grating that will prevent puncture of the screen if a pole, tool, or knuckle were to drop during construction. This is an intentional design aspect of the sump grating and screen.

RHR Recirculation Sump Blockage

The scaffold will be constructed in the containment annulus, which eliminates it from LOCA jet concerns. No planking will be installed on this scaffold, thereby eliminating them as a sump blockage concern. Only poles and knuckles will be installed, which do not float. Small materials (e.g., pens, pencils, things in PC pockets, ty-wraps, etc.) are prohibited above the sump. Tools and equipment brought into containment will be controlled in accordance with STP M-45B. The poles will not be painted, which will prevent addition of debris that could block the sump. The structure will be braced to prevent SISIP concerns and therefore prevent structure movement during a seismic event that could create sump blockage.

Conclusion:

By following existing plant procedures and observing the above bulleted requirements, the scaffold structure erected in Mode 1 through 4 will be within the existing design and license requirements. This activity does not result in a USQ.

00-083 Delete Trip of CWP 2-1 & 2-2 on Low ICW Pressure

Reference Document No.: DCP E-050524
Rev. No: 0
Reference Document Title: Delete Trip of CWP 2-1 & 2-2 on Low ICW Pressure
Safety Evaluation Description:
The automatic CWP trip from low Intake ICW pressure is being eliminated by this design change. Operators will manually initiate a trip of a CWP on loss of ICW flow based on the existing low pressure alarm, the stator high temperature alarm, and the PPC motor temperature indication.

Safety Evaluation Summary:
Deletion of the CWP on low ICW pressure will not change any safety analysis in the FSARU. There is no safety function of the CWP trip on low ICW pressure; it was installed only for protection of the Class II CWP motor. The ICW low pressure alarm (which will not be deleted), stator temperature alarm, and temperature indication on the PPC provide sufficient information to determine if a CWP needs to be manually tripped. This manual action will provide sufficient protection for the CWP motors.

Conclusion:
There is no safety impact from deleting the CWP trip from low ICW pressure. The reference to the CWP trip on low ICW pressure in the FSARU is descriptive only. This activity does not result in a USQ.

00-086 Pyrocrete Enclosure Thickness

Reference Document No.: FHARE 145
Rev. No: 0
Reference Document Title: Pyrocrete Enclosure Thickness
Safety Evaluation Description:
On the east wall of the Unit 1, 12 kV Switchgear Room (Fire Area 10), the bus duct to 4 kV Bus D bus duct runs adjacent to Enclosure 10-28-6, such that an adequate thickness and configuration to comply with the 2 hour tested configuration cannot be obtained. According to the fire endurance test, a Pyrocrete thickness of 2 inches, with an air gap of 2 inches between the interior edge of the Pyrocrete and the conduit, is required to provide the necessary 2 hour fire rating for the enclosure. However, because of the interference with the 4 kV Bus D bus duct, a 4 inch by 18 inch section is only capable of obtaining a 1-1/2 inch thickness of Pyrocrete which would be positioned against the 4 inch conduit inside the enclosure. While a test report for Pyrocrete 241 does show that the 1-1/2 inch thickness that we can obtain will provide a 2 hour rating, this configuration does not match the tested fire endurance test for the enclosure. Since this 4 inch by 18 inch section does

not comply with an approved tested configuration, this configuration was analyzed in FHARE 145 to determine if it will withstand the hazard associated with the area and to determine if the configuration will affect the ability to safely shut down the plant in the event of a fire in the fire area.

Safety Evaluation Summary:
FHARE 145 evaluates the adequacy of the Pyrocrete thickness, at the interface between Enclosure 10-28-6 and the 4 kV Bus D bus duct, to ensure the Bus F circuits located in the enclosure are adequately protected for the hazards in the area. With the exception of a 4 inch by 18 inch section, at the interface to Enclosure 10-28-6, and the 4 kV Bus D bus duct, sufficient Pyrocrete thickness is provided to provide a 2 hour fire rating. However, due to the location of the 4 kV Bus D bus duct, insufficient clearance is available to obtain the necessary Pyrocrete thickness and configuration to comply with the 2 hour tested configuration.

Conclusion:
While Enclosure 10-28-6 will have a small area (4 inch by 18 inch) that does not comply with the tested configuration, an adequate thickness of Pyrocrete will be installed to provide reasonable assurance that a fire in the area will not impact the safe shutdown circuits in the enclosure. As noted in FHARE 145, the fire loading for this area is less than 45 minutes, with a majority of that fire loading located on the 76 ft elevation. This fact, coupled with the fact that this reduction in Pyrocrete area is approximately 10 ft above the floor on the 85-ft elevation, the possibility of a fire of significant nature propagating to this area of the enclosure is remote. Therefore, because of the low combustible loading in the area, the location of the reduced Pyrocrete cross-sectional area, and the presence of an automatic smoke detection system, this configuration is expected to withstand the hazards of the area. Redundant safe-shutdown circuits will remain free from fire damage, and the ability of the plant to achieve and maintain safe shutdown will not be affected. This activity does not result in a USQ.

00-088 Unit 1 Cycle 11 Reactor Core Fuel Load and COLR

Reference Document No.: DCP N-49537

Rev. No: 0

Reference Document Title: Unit 1 Cycle 11 Reactor Core Fuel Load and COLR

Safety Evaluation Description:
This DCP incorporates the new fuel loading pattern for Cycle 11 into the plant design. This core reload supports the increase in power associated with the Unit 1 uprate project.

Safety Evaluation Summary:
The RSE identified limits on Cycle 11 Operation, in concluding that there were no USQs resulting from the Unit 1, Cycle 11 core design. These limits are:

1. Cycle 10 shutdown is at a burnup between 21,970 MWD/MTU and 23,220 MWD/MTU.
2. Cycle 11 burnup is limited to a maximum of 1,000 MWD/MTU beyond end-of-full-power capability.

   In addition, during coastdown:

   a. The Cycle 10 extended operation is achieved by reducing the 100 percent power Tavg to a temperature no lower than 565°F and then proceeding to a power reduction coastdown.
   b. The Lo-Lo Tavg setpoint and the Tavg program for rod control and steam dump systems are not changed.
3. 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 rod control system will be placed in the manual mode during the coastdown and the nuclear instruments will be adjusted daily, based on calorimetric power.
4. The RCS chemistry boron/lithium control program for reducing corrosion product transport in the RCS allows lithium concentrations of up to 3.5 ppm with a corresponding pH of 7.05 to 7.1.

Conclusion:
The evaluation performed in the Westinghouse RSE for Cycle 11 and the additional evaluations performed as a part of this design change ensure that the nuclear fuel for Cycle 11 is designed in accordance with the proper licensing and design-bases and that no impact on nuclear safety results from its implementation. The impact of the new lithium chemistry concentration limits has been evaluated to maintain corrosion effects below design limits for fuel cladding, SG tubes, and other RCS components. The safety significance of reuse of fuel assemblies with top nozzle spring screws susceptible to

cracking has been determined to be low such that conditional use for one fuel cycle is justified. This activity does not result in a USQ.

00-089 COLR for Diablo Canyon Unit 1 Cycle 11

Reference Document No.: COLR 1-11
Rev. No: 0
Reference Document Title: COLR for Diablo Canyon Unit 1 Cycle 11
Safety Evaluation Description:
The COLR is updated for every cycle to reflect the new core design. The FQ margin penalties in excess of 2 percent per 31 EFPD change, as well as the load follow W(z) factors.

Safety Evaluation Summary:
The RSE identified limits on Cycle 11 Operation, in concluding that there were no USQs resulting from the Unit 1, Cycle 11 core design. These limits are:

1. Cycle 10 shutdown is at a burnup between 21,970 MWD/MTU and 23,220 MWD/MTU.

2. Cycle 11 burnup is limited to a maximum of 1,000 MWD/MTU beyond end-of-full-power capability.

    a. In addition, during coastdown:
    b. The Cycle 10 extended operation is achieved by reducing the 100 percent power Tavg to a temperature no lower than 565°F and then proceeding to a power reduction coastdown.
    c. The Lo-Lo Tavg set point and the Tavg program for rod control and steam dump systems are not changed.

3. 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 rod control system will be placed in the manual mode during the coastdown and the nuclear instruments will be adjusted daily, based on calorimetric power.

4. The RCS chemistry boron/lithium control program for reducing corrosion product transport in the RCS allows lithium concentrations of up to 3.5 ppm with a corresponding pH of 7.05 to 7.1.

Conclusion:
The evaluation performed in the Westinghouse RSE for Cycle 11 and the additional evaluations performed as a part of this design change ensure that the nuclear fuel for Cycle 11 is designed in accordance with the proper licensing and design-bases and that no impact on nuclear safety results from

its implementation. The impact of the new lithium chemistry concentration limits has been evaluated to maintain corrosion effects below design limits for fuel cladding, SG tubes and other RCS components. The safety significance of reuse of fuel assemblies with top nozzle spring screws susceptible to cracking has been determined to be low such that conditional use for one fuel cycle is justified. This activity does not result in a USQ.

00-090 Containment Isolation Valve List

Reference Document No.: AD13.DC1
Rev. No: 13
Reference Document Title: Control of the Surveillance Testing Program
Safety Evaluation Description:
Modify the Containment Isolation Valve List, Attachment 7.7, by modifying Note 5, to allow opening valves 8885A and 8885B on an intermittent basis under administrative control in Mode 4 under STP V-5A2. The note previously allowed opening these valves only under OP B-2:IX – the RHR flushing procedure, which can be performed in Modes 1 though 4.

Safety Evaluation Summary:
The proposed change will allow performing STP V-5A2, "ECCS Check Valve Leak Test, Post-Refueling/Post-maintenance Valves 8948A-D and 8818A-D", in Mode 4, which opens valves 8885A and 8885B. These valves are containment isolation valves that are normally closed in Modes 1 through 4.

Conclusion:
This change does not constitute a USQ, nor does it have any effect on offsite dose limits defined in 10 CFR 100. This change has no licensing-basis impact on DCPP.

00-091 Revise FSARU Section 5.5 to Allow RCP Vibration Signal Defeat

Reference Document No.: FSARU Section 5.5
Rev. No: 14
Reference Document Title: Revise FSARU Section 5.5 to Allow RCP Vibration Signal Defeat
Safety Evaluation Description:
This change documents the acceptability of defeating the signals from the RCP vibration monitoring probes, turning the RCP vibration computer off. The operator actions due to abnormal RCP indications are clarified. This FSARU change further documents that the RCP vibration computer is not required for safe plant operation.

Safety Evaluation Summary:
This change recognizes practices at DCPP that minimize nuisance alarms in the control room. Occasionally an RCP monitoring probe or other component will malfunction. Jumpers have been used in the past to document this practice. This change will document in the FSARU that these signals may be defeated, without performing a separate LBIE on each occasion that such an action is required.

The use of vibration monitoring in diagnosing, evaluating RCP malfunctions, and shutting down an RCP is clarified to match existing procedural guidance.

The alarm processing for RCP vibration alarms is clarified in the FSARU, to recognize that these alarms are not required for safe plant operation, and that the RCP vibration computer does not drive the main annunciator alarms.

FSARU Chapter 15, Section 15.4.4, analyzes the consequences of a single locked RCP rotor. The section assumes that the RCP rotor instantaneously seizes. No credit is taken for the vibration monitoring system. Hence, the system does not affect consequences or probability of events as described in the FSARU.

Conclusion:
Since the RCP vibration monitoring system is only for condition monitoring and does not perform a safety function that is relied upon in the SAR or license basis, individual signals or the RCP vibration monitoring computer may be defeated or turned off to eliminate nuisance alarms or for maintenance purposes. This activity does not result in a USQ.

00-092 Control of the Surveillance Testing Program
Reference Document No.: AD13.DC1
Rev. No: 14
Reference Document Title: Control of the Surveillance Testing Program
Safety Evaluation Description:
All changes are contained in Attachment 7.7, "Containment Isolation Valves."

1. Specify panel isolation valves that can be used as CIVs when the normally-closed instrument CIVs are opened. This will reduce TS entries.
2. Delete electrical penetrations and welded-cap spare penetrations from the list – they are not CIVs and do not require surveillance in

accordance with SR 3.6.3.3 or 3.6.3.4. Deleted the accompanying Note 1.

3. Delete electrical penetration plant instrument (PI) root valves from the list – they are not CIVs (reference NCR N0002036).
4. Add clarifying note as to application of "administrative controls" when opening valves in the table.

Safety Evaluation Summary:
This change will allow instrument calibrations and repair of instrumentation manifold equipment to be performed in Modes 1 through 4 without entering the Action of TS 3.6.3 for opening a normally closed containment isolation valve. This is because a qualified upstream valve will be closed prior to opening the valve, maintaining full operability of the containment penetration. This change does not apply to valves on penetrations that are local leak rate tested.

Deletion of the electrical penetrations and PI root valves (which the NRC has agreed are not CIVs) is being performed to simplify the table and restore it to its specified function – listing the CIVs.

Conclusion:
This change is of an administrative nature and does not result in a change to the FSARU. However, any changes to Attachment 7.7 require performance of an LBIE. This activity does not result in a USQ.

00-094 Safe Shutdown Analysis for Fire Areas 4-A and 4-B (Chem. Laboratory and Access Control Area)

Reference Document No.: FHARE 117, Rev. 0, DCP M-049536
Rev. No: 0
Reference Document Title: Safe Shutdown Analysis for Fire Areas 4-A and 4-B (Chem. Laboratory and Access Control Area)
Safety Evaluation Description:
FHARE 117, Rev. 0, was generated to evaluate unsealed penetrants in the 1-hour rated ceiling in Fire Area 4-A, and not crediting the 2-hour rated walls that separate Fire Areas 4-A and 4-B. This FHARE evaluates the impact on the deviation from Appendix R, Section III.G.2 requirements that were approved in SSERs 23 and 31 and provides the basis for DCP M-049536.

Safety Evaluation Summary:
The non-rated penetration configurations in the 1-hour ceiling and in the 2-hour barrier separating Fire Areas 4-A and 4-B were determined to not affect the conclusions in the approved Appendix R Section III.G deviations in SSERs 23 and 31. The configurations will not adversely affect the ability to ensure that damage to redundant trains of safe-shutdown systems would be limited such that the ability to achieve and maintain safe shutdown can still be achieved. DCP M-049536 is implementing the modification to the configuration of the fire rated barriers described in FHARE 117.

Conclusion:
Based on the low combustible loading in both Fire Areas 4-A and 4-B, the additional barriers installed around Bus G circuits above the 1-hour ceiling by DCP M-049536, the fire rated enclosure around the diesel fuel oil and ASW pump room fan circuits, the available fire protection features, and the location of redundant components, the non-rated penetration seal configurations would not affect the ability to achieve and maintain safe shutdown. This activity does not result in a USQ.

00-095 Reactor Coolant System Vacuum Refill
Reference Document No.: N-050532
Rev. No: 0
Reference Document Title: Reactor Coolant System Vacuum Refill
Safety Evaluation Description:
This LBIE addresses permanent modification to the plant, temporary use of the RCS vacuum refill (RCSVR) equipment, and the acceptability of using the RCSVR assist during Mode 5.

1. Permanent modifications to the Plant:

   A new 1 inch RCSVR connection with dual isolation valves is added on the 6 inch PORV inlet header with provisions to connect a 1-1/2 inch vacuum hose. The duel isolation valves provide part of the reactor coolant pressure boundary (RCPB) during Modes 1 though 4. In Mode 5, these valves provide the connection to the RCSVR components.

2. For temporary use of the RCSVR components:

   Temporary hoses and a skid-mounted vacuum pump with a high-efficiency particulate air filter will be used to connect the RCSVR components to the RCS. The new valves will be used with a 1-1/2 inch vacuum hose to connect the vacuum pump to the Pressurizer volume and a 2 inch vacuum hose will connect the remaining RCS by means of an existing 2 inch connection on the RCS Loop 1 Cold Leg, which includes two RCPB valves.

Safety Evaluation Summary:
DCP N-050532 allows the modification of the RCPB to add a 1 inch connection and use an existing 2 inch RCPB connection to use a RCSVR process during Mode 5.

The use of the RCSVR process does not impact the operating license, licensing basis, or the design basis of the plant. An alternate method of refilling the RCS during an outage is provided. This method uses a vacuum assist to evacuate the air/gasses from the RCS during the fill process. This process is added as an alternate method to the FSARU.

The revision of the inservice inspection (ISI) boundary drawings and the piping schematics is part of the FSARU figures. However, this is acceptable since the change relates only to a non-licensing basis of the SAR. The design installation and operation of the RCPB connections are in accordance with the applicable design classifications and meet the description of the system in the SAR.

Conclusion:
In summary, the use of the RCSVR will not impair the safety function or performance of the reactor vessel, reactor internals, control rod drive mechanism (CRDM) system analysis, level monitoring systems, RCP seals, pressurizer, SGs, tanks, pumps, heat exchangers, filters, demineralizers, valves, nor RCS piping. The use of the RCSVR will not adversely affect the safe operation of the plant. The operation of the RCSVR does not represent a potential USQ nor does it require a change to plant TS.

00-096 New RVRLIS and PPC Multiplexer in Containment

Reference Document No.: DCP J-050525
Rev. No: 0
Reference Document Title: New RVRLIS and PPC Mux in Containment
Safety Evaluation Description:
This design change replaces the existing Delta-P reactor vessel refueling level indication system (RVRLIS) narrow and wide range transmitters with four transmitters. The four transmitters sense pressurizer vent space pressure (wide range reference leg), reactor head pressure (narrow range reference leg), reactor water level (narrow range variable leg) and refueling pool water level (wide range variable leg). The signals from these transmitters are routed

to a new PPC multiplexer which will be installed in the containment. The PPC will use the signals from the four new transmitters to determine wide and narrow range refueling levels and initiate main annunciator system (MAS) alarms similar to the existing system. The new system also provides an ultrasonic level signal from the hot leg to the PPC. The ultrasonic level signal initiates from a level sensing device adapted for the application by ISI.

Safety Evaluation Summary:
This design modification installs new RVRLIS transmitters, a PPC multiplexer in containment, fiber optic feedthroughs in Containment Penetration 25, some wiring in cabinet PK010 in the Auxiliary Building in order to connect the PPC to the MAS, and interconnecting wiring and raceway in the containment and Auxiliary Building.

The PPC will display: wide range refueling pool/vessel level, narrow range vessel level, pressurizer pressure, reactor vessel head pressure, and the ultrasonic level (mid loop only). The PPC will initiate alarms due to high/low wide range level, high/low narrow range level, low ultrasonic level, high pressure in the pressurizer, and high pressure in the reactor vessel. These will alarm on a new alarm window PK02-22 dedicated for refueling outages. Advantages with the new system include: shorter tubing lengths resulting in quicker response times, pressurizer and reactor vent space pressure signals which support vacuum refill, and the installation of the PPC multiplexer in containment that supports other projects including a more robust and permanent pressurizer safety valves' loop seals' temperature monitor.

Conclusion:
The new RVRLIS system meets all licensing requirements while providing user friendly and diverse reactor vessel refueling level indication. The installation of the system allows for future monitoring capabilities by installing a PPC multiplexer in the containment. The installation provides for additional fiber optic data to support other related outage needs. This activity does not result in a USQ.

00-097 Hydrogen Recombiner
Reference Document No.: FSAR Update
Rev. No: 13
Reference Document Title: Hydrogen Recombiner
Safety Evaluation Description:
This change corrects the containment hydrogen concentration limit statement from "3.5 percent" to "4 percent." The current licensing basis for control of combustible gas concentrations in containment following a LOCA is 4 percent.

However, the text in FSARU Section 6.2.5 has statements describing the hydrogen concentration limit as 3.5 percent and flammability limit of 4 percent. This FSARU change is intended to clarify the licensing basis for the hydrogen concentration to be 4 percent.

Safety Evaluation Summary:
The proposed changes to the FSARU discussion concern the allowable hydrogen concentration and operation of the hydrogen recombiners and are being made to reflect the actual regulatory guidance included in RG 1.7 and the SRP. The actual FSARU change revises paragraphs (2) and (4) of Section 6.2.5.1, and the seventh paragraph in Section 6.2.5.2.2 and the last two sentences in the first paragraph of Section 6.2.5.3.1.4.

RG 1.7, Revision 2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," provides acceptable assumptions for evaluation the production of combustible gasses following a LOCA. In that RG the acceptable limit for hydrogen concentration is 4 v/o. This RG states that the 4 v/o limit would insure that there would be no burning of hydrogen within the containment.

The SRP, Section 6.2.5, "Combustible Gas Control In Containment," Sub-Section 4, states: "The proposed operation of the combustible gas control equipment, excluding containment atmosphere dilution (CAD) systems, is acceptable if there is an appropriate margin (e.g., on the order of 0.5 v/o) between the limiting hydrogen concentration limit and the hydrogen concentration at which the equipment would be actuated." The concentration limit that is referred to in the SRP is from RG 1.7 and equals 4 v/o.

Revision 0 of the FSARU, Section 6.2.5.3.1.4 "Conclusion," provided that starting a recombiner at 3.5 v/o (at approximately 30 days) or earlier will ensure that the bulk containment hydrogen concentration will not reach the low flammability limit of 4 v/o.

In June 1997, Westinghouse provided a report titled, "Post-LOCA Hydrogen Generation Evaluation From Radiolysis," which was the result of a request from PG&E. The assumptions in that report and evaluation included increased material volumes because of in containment storage requirements for some additional materials such as scaffolding. The result of that evaluation demonstrated that a single recombiner placed in service at the 3.5 v/o limit, would continue to maintain the containment hydrogen concentration below the lower flammability limit of 4 v/o.

Revision 12 of the FSARU, Section 6.2.5.3.1.4, "Conclusion", incorporates a change that included the June 1997 Westinghouse report results. That change maintained the statement about bringing in a recombiner at 3.5 v/o and maintaining the containment hydrogen concentration below the lower flammability limit of 4.0 v/o.

The DCPP licensing basis has always been, and continues to be, the lower flammability limit. The administrative limits established and proceduralized have always protected, and continue to protect, that licensing limit. The clarifications provided by this request do not change the level of protection or the licensing-basis limit. They are provided to reduce potential misunderstanding between the actual licensing limit and the administrative limits.

Conclusion:
Based upon the fact that the actual licensing basis for hydrogen concentration limit remains unchanged at 4 percent in the proposed text, it can be concluded that no licensing basis is impacted for this proposed FSARU change. This activity does not result in a USQ.

00-098 Recirculation Sump Screen Modification

Reference Document No.: DCP N-50510
Rev. No: 0
Reference Document Title: Recirculation Sump Screen Modification
Safety Evaluation Description:
DCP N-50510 provides the design to modify the recirculation sump screen and related structures to substantially increase the available surface screen area. This modification will remove the existing inclined grating and associated 1/8 inch x 1/8 inch stainless steel mesh, remove a major portion the weir wall downstream of the inclined grating and remove the 6 inch high curb downstream of the weir wall. These components will be replaced with separate elements consisting of a 6 inch debris curb, a stainless steel grating trash tack, and an extended-surface sump screen fabricated from stainless steel plate perforated with 1/8 inch diameter holes and a 40 percent free area. The new design affords a significant increase of the available screen area. In addition, the new design has features that are recommended in RG 1.82.

Safety Evaluation Summary:
Ongoing industry evaluations of ECCS sump screen blockage due to LOCA debris, including fibrous materials, Min-K insulation, paint debris, insulation vapor barrier paper, and fire barrier material, have resulted in a net reduction of DCPP's sump screen head loss margin. Industry evaluations are expected to continue. LANL is supporting the NRC in the resolution of Generic Safety Issue, GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." LANL has been tasked to develop a methodology for

estimating debris generation and debris transport in PWR containments. The outcome of these actions could have potential adverse impact on DCPP sump calculations and margin. Reconfiguration of the sump screen will significantly increase the available sump screen area.

The new design of the recirculation sump screen incorporates design elements specifically recommended in RG 1.82. In addition, the extended-surface perforated plate screen surface provides a significant increase in the available screen area. The new sump screen configuration will perform the same design functions as the existing recirculation sump screen configuration to: (1) provide sufficient surface screen area to assure adequate NPSH is afforded the ECCS pumps during the recirculation phase of a design-basis LOCA; (2) minimize the effects of air ingestion; and (3) minimize the amount of debris ingested into the ECCS.

Conclusion:
The licensing-basis review of the changes to the configuration of the recirculation sump screen demonstrates that the recirculation sump remains operable during the design-basis events defined in the FSARU. The ECCS system component design for reliability, redundancy, and operation within design and safety limits is not affected by this change. No events that could impact the health and safety of the public are determined to be created by this change to the configuration of the recirculation sump screen. This activity does not result in a USQ.

00-099 Replace RVLIS and TMS Processors, Signal Conditioners, and Displays, Unit 1

Reference Document No.: DCP J-049434
Rev. No: 0
Reference Document Title: Replace RVLIS and TMS Processors, Signal Conditioners, and Displays, Unit 1
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 CETs and from the TMS (reference junction RTD and hottest T/C data) and ERFDS from the CETs. The PPC will be provided with the same data through a fiber-optic data link through the existing Validyne server. The ERFDS will be provided with the same data through 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.

Safety Evaluation Summary:
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 enhanced features such as automatic self-test and diagnostics, greater flexibility, improved operator interface, 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.

The new TMS/RVLIS architecture includes features to mitigate the unlikely event that either or both trains experience failure of a required application or lockup of a processor - failures that could delay access to information used by the operators to assess accident progress, and thus affect accident recovery and consequences. Following such an event, the operators will be alerted to the failure condition and the system will restart automatically without intervention. The existing systems do not restart automatically. In the extreme case that both trains fail due to a common cause and the automatic restart feature fails in both trains, limited data will continue to be available from the nonsafety-related SCMM, which is not affected by this upgrade.

Software changes in the PPC, Validyne server, and the ERFDS are performed in accordance their respective SQA plans 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 cannot increase the probability, consequences, or possibility of an accident or malfunction previously evaluated in the SAR. Margin of safety, as discussed in any TS, is not affected because the RVLIS and TMS have no protection or actuation function. This activity does not result in a USQ.

00-100 Class I Outdoor Water Storage Tanks - Removal of Delaminated Layer of Concrete

Reference Document No.: DCP C-049530
Rev. No: 0
Reference Document Title: Class I Outdoor Water Storage Tanks - Delaminated Concrete Removal

Safety Evaluation Description:
This LBIE documents that the dimensional change on design drawing No. 463987 showing the concrete/shotcrete to be a minimum of 10 inches thick provides the structural capacity to meet the licensing requirements. Drawing No. 463987 is in the FSARU as Figure 3.8-65, Sheet 2 of 2.

Safety Evaluation Summary:
The function of the concrete shell on the outside of the tanks is to provide security protection and structural integrity to resist the forces due to a seismic event. The minimum thickness for security is 8 inches. Calculations 52.21.9 and 52.21.10 verified that the minimum thickness of 10 inches will provide the structural integrity to resist the seismic events and tornado. The work will be to remove the loose concrete from the tank. This is generally above approximate elevation 126 ft where the concrete was placed using the shotcrete method. Where the cover over the rebar is less than 2 inches, the concrete will be coated to prevent direct exposure to the weather. The work does not affect the operation or the function of the tanks.

Conclusion:
The concrete shell of a minimum of 10 inches on the walls provides all the requirements to meet the FSARU requirements for security and to resist the forces from a seismic event. When the concrete delaminated layer on top of the tanks is removed, the thickness of the concrete shell will be less than 8 inches. Security shall be notified prior to the removal and will provide the compensatory measures as required. The change to the drawing does not impact the ability of the outdoor water storage tanks to perform their safety-related functions. This activity does not result in a USQ.

00-101 Class I Outdoor Water Storage Tanks - Delaminated Concrete Removal

Reference Document No.: DCP C-050530

Rev. No: 0

Reference Document Title: Class I Outdoor Water Storage Tanks - Delaminated Concrete Removal

Safety Evaluation Description:

This LBIE documents that the dimensional change on design drawing No. 463987 showing the concrete/shotcrete to be a minimum of 10 inches provides the structural capacity to meet the licensing requirements. Drawing No. 463987 is in the FSARU as Figure 3.8-65, Sheet 2 of 2.

Safety Evaluation Summary:

The function of the concrete shell on the outside of the tanks is to provide security protection and structural integrity to resist the forces due to a seismic event. The minimum thickness for security is 8 inches. Calculations 52.21.9 and 52.21.10 verified that the minimum thickness of 10 inches will provide the structural integrity to resist the seismic events and tornado. The work will be to remove the loose concrete from the tank. This is generally above approximate elevation 126 ft where the concrete was placed using the shotcrete method. Where the cover over the rebar is less than 2 inches, the concrete will be coated to prevent direct exposure to the weather. The work does not affect the operation or the function of the tanks.

Conclusion:

The concrete shell of a minimum of 10 inches on the walls provides all the requirements to the meet the FSARU requirements for security and to resist the forces from a seismic event. When the concrete delaminated layer on top of the tanks is removed, the thickness of the concrete shell will be less than 8 inches. Security shall be notified prior to the removal and will provide the compensatory measures as required. The change to the drawing does not impact the ability of the outdoor water storage tanks to perform their safety-related functions. This activity does not result in a USQ.

00-102 Revise Main Feedwater and Main Steam Instrument Scaling

Reference Document No.: J-050558

Rev. No: 0

Reference Document Title: Revise Main Feedwater and Main Steam Instrument Scaling

Safety Evaluation Description:

This change will modify the scaling for main steam and main feedwater flows in the Digital Feedwater Control System (DFWCS), Eagle 21 Process Protection System, PPC, SPDS, ERFDS, ERDS and associated indicators and recorders in various locations to conform to the 120 percent NSSS design flow specified in the DFWCS Functional Requirements. This modification does not affect the operation or function of the DFWCS, Eagle 21 or the information systems (PPC, SPDS, ERDS and ERFDS).

This change also modifies the scaling of ERDS. The ERDS data stream is not otherwise affected. Changes to the ERDS require NRC notification within 30 days of the modification in accordance with NUREG 1394, Rev. 1, "Emergency Response Data System (ERDS) Implementation." Prior approval is not required by NUREG 1394, Rev. 1

Safety Evaluation Summary:
This design change modifies the scaling for main steam and main feedwater flows in the DFWCS, Eagle 21 PPC, SPDS, ERFDS, ERDS and associated control board indicators and recorders from the current value of 4.2 million pounds per hour (MPPH) to a new value of 4.5 MPPH. The functional requirements for the DFWCS state that the steam and feed flows are scaled to 0-120 percent of the NSSS design rating. The Unit 2 NSSS design rating is 3.74 MPPH. The value of 4.5 MPPH is 120 percent (rounded up) of the DCPP Unit 2 design NSSS rating. This modification does not affect the operation or function of the DFWCS, Eagle 21 or the information systems (PPC, SPDS, ERDS and ERFDS). All calculations and control actions are performed in engineering units. This change will only result in a lower raw input signal value corresponding to the same flow rate in engineering units. Actual design flow rates are not affected.

This change is intended to reduce the likelihood of one or more DFWCS loops failing to manual during certain feedwater system transients. The operator will be able to devote full attention to mitigating the transient. The risk of challenge to plant protection system is not increased. Therefore, the change does not have any impact on any accident analysis as described in the FSARU Chapters 6 and 15.

The affected monitoring systems (PPC, SPDS, ERFDS) do not perform any control or protection functions and are directly not used for accident mitigation. Main feedwater and main steam flows are RG 1.97 functions (FSARU Table 7.5-6) but are only used to verify that the process lines have been properly isolated following an accident. This scaling change will not affect the use of the main feedwater and main steam flow instrumentation following an accident.

Conclusion:
This change does not affect any FSARU Chapter 6 or 15 accident analysis. The main steam and main feedwater flows are not associated with (and thus will not adversely affect) any ESF systems or components associated with detection or mitigation of any design-basis events. The upgrade will not increase probability, frequency, or consequences of evaluated events or equipment malfunctions. A USQ is not involved or created by this change.

00-103 Revise Main Feedwater and Main Steam Instrument Scaling (Rev. 1)

Reference Document No.: J-050558
Rev. No: 1
Reference Document Title: Revise Main Feedwater and Main Steam Instrument Scaling (Rev. 1)

Safety Evaluation Description:
This change will modify the scaling for main steam and main feedwater flows in the DFWCS, Eagle 21 Process Protection System, Plant Process Computer (PPC), SPDS, ERFDS and associated indicators and recorders in various locations to conform to the 120 percent NSSS design flow specified in the DFWCS Functional Requirements.

This change also modifies the scaling of ERDS. The ERDS data stream is not otherwise affected. Change to the ERDS requires NRC notification in accordance with NUREG 1394, Rev. 1, "Emergency Response Data System (ERDS) Implementation."

Revision 1 addresses replacement of the existing Westronics analog recorders FR-510, -520, -530 and –540 with Yokogawa digital paperless videographic recorders.

Safety Evaluation Summary:
This design change modifies the scaling for main steam and main feedwater flows in the DFWCS, Eagle 21 PPC, SPDS, ERFDS, ERDS and associated control board indicators and recorders from the current value of 4.2 MPPH to a new value of 4.5 MPPH. The functional requirements for the DFWCS state that the steam and feed flows are scaled to 0-120 percent of the NSSS design rating. The Unit 1 uprate 100 percent RTP flow is 3.71 MPPH per loop [Ref STA-114]. The new value of 4.5 MPPH is 120 percent (rounded up) of the DCPP Unit 1 uprate 100 percent loop flow rate. This modification does not affect the operation or function of the DFWCS, Eagle 21 or the information systems (PPC, SPDS, ERDS and ERFDS). All calculations and control actions are performed in engineering units. This change will only result in a lower raw input signal value corresponding to the same flow rate in engineering units. Actual design flow rates are not affected.

This change is intended to reduce the likelihood of one or more DFWCS loops failing to manual during certain feedwater system transients. The operator will be able to devote full attention to mitigating the transient. The risk of challenge to plant protection systems is not increased. Therefore, the change does not have any impact on any accident analysis as described in the FSARU Chapters 6 and 15.

The affected monitoring systems (PPC, SPDS, ERFDS) do not perform any control or protection functions and are directly not used for accident mitigation. Main feedwater and main steam flows are RG 1.97 functions (FSARU Table 7.5-6) but are only used to verify that the process lines have been properly isolated following an accident. This scaling change will not affect the use of the main feedwater and main steam flow instrumentation following an accident.

Conclusion:
This change does not affect any FSARU Chapter 16 or 15 accident analysis. The main steam and main feedwater flows are not associated with (and thus will not adversely affect) any ESF systems or components associated with detection or mitigation of any design-basis events. The upgrade will not increase probability, frequency, or consequences of evaluated events or equipment malfunctions. A USQ is not involved or created by this change.

01-001 VCT H_2 Supply/Relief Pressure Setpoint Range Change

Reference Document No.: DCP N-49531 & N-50531

Rev. No: 0

Reference Document Title: VCT H_2 Supply/Relief Pressure Setpoint Range Change

Safety Evaluation Description:

The subject design change will revise the hydrogen gas supply pressure to the volume control tank (VCT). In particular, the hydrogen supply pressure control valve PCV-955 setpoint range will be lowered from 15 - 35 psig to 15 - 26 psig. The associated VCT relief pressure (PC-190 setpoint) will be lowered from 17 - 37 psig to 17 - 28 psig to reflect past design objective of maintaining relief pressure 2 psi above supply pressures.

In addition, FSARU Appendix 9.5A requires revision to correct actions pertaining to opening RWST suction valves SI-8805A and SI-8905B. Specifically, the evaluations will be corrected to note the valves may be damaged by fire, thus requiring local manual opening.

Safety Evaluation Summary:

INPO OE-9961, "Pre-Fire Plan Inadequacies (Appendix R)," identified a fire condition at Beaver Valley 1 which can render the CCPs inoperable. Specifically, fire damage to VCT isolation valve control circuitry may render the valving inoperable to the extent that the remote manual valve closure function, and valve interlock feature with the RWST suction valving, are lost. Normally, a CCP takes suction from the VCT, with automatic swapover to the RWST on VCT low level. During a postulated fire, the VCT cannot be isolated (via manual valve closure) from the CCP within the short time it takes to empty the VCT. Assuming that both the RWST and VCT are aligned to CCP suction and the VCT's hydrogen blanket pressure is greater than RWST static pressure, the CCP will draw suction from the VCT versus the RWST. Once emptied, the VCT's hydrogen gas space is drawn into the pump's suction, resulting in pump cavitation, thus affecting pump operability. According to DCPP's fire hazards analysis (FSARU Appendix 9.5A and 9.5G) and calculation N-061, this condition also applies to DCPP, as current design permits VCT pressures in excess of RWST pressure.

Conclusion:

In resolution of INPO OE-9961, this DCP will lower the hydrogen cover gas supply and relief pressure range to the VCT to ensure tank pressure remains below RWST static pressure. The lower hydrogen pressure adds an additional engineering-design feature to help ensure the CCP takes suction from the RWST versus VCT when CCP suction is aligned to both tanks. In this case, the VCT will not empty and hydrogen gas will not enter pump suction, regardless of VCT isolation valving position. This activity does not result in a USQ.

01-003 ECG 23.6

Reference Document No.: ECG 23.6
Rev. No: 0
Reference Document Title: 480 Vac Class I Switchgear Ventilation
Safety Evaluation Description:
This change creates new ECG 23.6 to place the 480 Vac Class I switchgear ventilation system under ECG controls. The 480 Vac Class I switchgear ventilation system is risk significant as determined by the PRA and endorsed by the Maintenance Rule Expert Panel.

Safety Evaluation Summary:
The 480 Vac class I switchgear ventilation system is risk significant, as determined by the PRA and endorsed by the Maintenance Rule Expert Panel. As such, the system is scoped under the Maintenance Rule (10 CFR 50.65) and is subject to the new requirements of 10 CFR 50.65(a)(4), effective November 28, 2000. The new paragraph requires that, prior to performing maintenance activities on risk significant SSCs, the increase in risk from the proposed maintenance activities be assessed and managed. This new ECG will provide controls to assure system unavailability is corrected in a timely manner. The ECG will also provide a mechanism for tracking unavailability of the system, which is a requirement under the Maintenance Rule. The system will not be operated or maintained differently than in the past.

Conclusion:
Implementation of ECG 23.6 will provide additional controls on the 480 Vac Class I switchgear ventilation system to maintain its availability. The change is conservative and does not result in a USQ.

01-006 Primary to Secondary Leakage Monitoring/Air Ejector Gaseous Effluent Monitors

Reference Document No.: ECG 2.1 / ECG 39.2
Rev. No: 0 / 3
Reference Document Title: Primary to Secondary Leakage Monitoring/Air Ejector Gaseous Effluent Monitors
Safety Evaluation Description:
The proposed change will rescind ECG 39.2 Rev. 3, "Air Ejector Gaseous Effluent Monitors," and relocate the requirements for RM-15 and RM-15R to the new ECG 2.1, Rev. 0, "Primary to Secondary Leakage Monitoring." Additionally it will add operability, surveillance, and action requirements for the Steam Jet Air Ejector (SJAE) wide-range flow rate channel, FIT-81, to

ECG 2.1 Rev. 0. The Bases have been updated accordingly.

Safety Evaluation Summary:
Currently, ECG 39.2 ensures the operability of RM-15 & RM-15R, the radiation monitors that monitor the flow stream from the SJAE and used in monitoring primary to secondary leakage. This change is an administrative change that moves the requirements of ECG 39.2 to a new ECG that will include requirements for the SJAE wide-range flow rate channel, FIT-81. The SJAE wide-range flow rate channel monitors the air exhaust flow rates and is used in conjunction with RM-15 or RM-15R to determine the primary-to-secondary leakage. This change does not change the design nor does it affect the operation of any of these components. The new requirements for the SJAE wide-range flow rate channel provide additional assurance that it is operable and capable of performing its safety function.

Conclusion:
FIT-81 and RM-15/15R are needed to ensure the requirements of TS 3.4.13 are met. Since they both monitor the same process, it is most efficient to combine them into one process-oriented ECG. This is consistent with other DCPP TSs and ECGs. This change does not result in a USQ.

01-007 Reactivity Control Systems – Position Indication System - Shutdown

Reference Document No.: ECG 41.1
Rev. No: 1
Reference Document Title: Reactivity Control Systems – Position Indication System - Shutdown
Safety Evaluation Description:

1. TSI 94-10 has been incorporated into ECG 41.1.

2. The Applicability statement has been revised by adding the words "...and rods capable of being withdrawn."

3. The two maintenance procedures that are credited in TSI 94-10 for determining the functionality of digital rod position indication (DRPI) are being upgraded to STPs and are used as the basis for two new SRs, SR 41.1.1 and SR 41.1.2.

4. SR 41.1.3 (previously SR 41.1.1) states that it is the same SR as TS SR 3.1.7.1. The wording of SR 41.1.3 has been revised to match the wording of SR 3.1.7.1, and the frequency of SR 41.1.3 (24 months) has been revised to match the frequency of SR 3.1.7.1 (Once prior to criticality after each removal of the reactor vessel head), which is more conservative.

5. The ECG 41.1 Bases have been expanded and rewritten in ITS format.

Safety Evaluation Summary:

AR A0502970 requests that TSI 94-10 be incorporated into ECG 41.1 and that TSI 94-10 be rescinded. ECG 41.1 is a relocated TS (3/4.1.3.3, "Reactivity Control Systems - Position Indication System - Shutdown") that was approved for relocation by LAs 120/118. TSI 94-10 applied to TS 3.1.3.3, and by this change, is now incorporated into ECG 41.1. As a result of this change, TSI 94-10 may be rescinded.

The addition of the words "and rods capable of being withdrawn" to the Applicability statement provides additional flexibility for performing maintenance. It allows the reactor trip breakers (RTBs) to be closed without entering ECG 41.1, as long as the rods are incapable of being withdrawn (such as by deenergizing the rod control motor generator sets). This condition is consistent with similar conditions specified in TS for other systems, such as TS 3.3.1 "RTS Instrumentation," Table 3.3.1-1, note (f) that specifies the condition "With the RTBs open or all rods fully inserted and incapable of withdrawal."

The ECG 41.1 Bases have been revised to reflect the provision of TSI 94-10 that DRPI rods on bottom lights are an acceptable means of verifying all rods are fully inserted, even though DRPI has not been declared operable. The two procedures that are credited in TSI 94-10 for verifying the functionality of DRPI are being upgraded from maintenance procedures to STPs and are used as the basis for two new SRs SR 41.1.1 and 41.1.2 that will be used to verify the functionality of DRPI.

The rewording of SR 41.1.3 to make it identical with TS SR 3.1.7.1 is administrative in nature and also results in a more conservative frequency by replacing "24 months" with "Once prior to criticality after each removal of the reactor vessel head."

Conclusion:

The changes proposed for Revision 1 to ECG 41.1 involve relocating a previously approved TSI to the ECG, providing additional flexibility in the Applicability statement of the ECG consistent with TS, rewording an SR to match TS, formatting changes, and the addition of information taken from other approved documents. The changes do not result in a USQ.

01-010 Revision 3 to ECG 4.4

Reference Document No.: ECG 4.4
Rev. No: 3
Reference Document Title: Instrumentation – Turbine Overspeed Protection
Safety Evaluation Description:
The proposed change will add a note to the applicability of ECG 4.4 that permits the turbine to be returned to service under administrative control, solely to perform testing required to demonstrate turbine overspeed protection system operability. The Bases portion of the ECG further explains that this note is consistent with TS 3.0.5.

Safety Evaluation Summary:
This change adds a note permitting the turbine to be returned to service under administrative control for operability testing. Overspeed testing of the turbine requires it to be run at greater than1800 rpm. Testing the turbine overspeed protection System at greater than 1800 rpm is controlled by DCPP procedure STP M-21B, satisfying SR 4.4.2. The testing performed under STP M-21B is only the final verification performed. Various functional tests are performed prior to turbine operations, e.g., MP I-1.36-1, "Main Turbine Control Integrated Functional Test." In OP C-3:II, "Main Unit Turbine – Startup," the turbine is tripped at 550 rpm on the initial turbine roll up and the turbine valves are walked down to verify that all the turbine valves closed. The turbine overspeed trip test (simulated) of STP M-21A is performed after the unit reaches synchronous speed.

Conclusion:
This change adds a note to ECG 4.4, consistent with TS 3.0.5, to permit the turbine to be placed under administrative control to allow operability testing. This change does not result in a USQ.

01-011 Post-LOCA Containment Hydrogen Generation Evaluation

Reference Document No.: STA-137
Rev. No: 0
Reference Document Title: Post-LOCA Containment Hydrogen Generation Evaluation
Safety Evaluation Description:

FSARU Section 6.2.5 and Tables 6.2-41, 42, 43, 44, and 45 and Figures 6.2-25, 26, 27, 28, and 29 were changed to reflect the new post-LOCA containment hydrogen generation analysis in support of keeping the scaffolding inside containment (AR A0507209, Evaluation No. 07).

Safety Evaluation Summary:
The report (Westinghouse letter PGE-00-560 with calc note CN-REA-00-81) of this evaluation presents the results of post-LOCA hydrogen production and accumulation inside containment and provides a means of conversion between contingency inventories of corrodible materials (i.e., Zn and Al). Some of the Westinghouse recommended changes to the FSARU are not incorporated into this proposed FSARU change, because the recommended changes are arbitrary in nature and do not reflect any real changes as a result of this evaluation.

Conclusion:
This post-LOCA containment hydrogen generation evaluation does not have any adverse impacts on the DCPP licensing basis. The results of the new evaluation are consistent with the current licensing basis. This activity does not result in a USQ.

01-012 Revisions to ECG 18.7, STP M-70A, STP M-70C

Reference Document No.: ECG 18.7, STP M-70A, STP M-70C
Rev. No: 5 / 4 / 5
Reference Document Title: ECG 18.7, Fire Rated Assemblies
Safety Evaluation Description:

1. ECG 18.7 is being revised to update the bases section and incorporate the ITS format to resolve human factors issues.

2. ECG 18.7 is being revised to add an additional compensatory measure for certain penetration seals that will allow a temporary repair of qualified penetration seals in lieu of implementing a continuous fire watch.

3. ECG 18.7 is being revised to add an option to install the portable detection system, in conjunction with an hourly firewatch, for inoperable fire rated assemblies that do not have fire detection on at least one side of the assembly.

4. ECG 18.7 is being revised to add an additional compensatory for fire doors. The new action will provide administrative control over non-functional fire doors.

5. STP M-70A is being revised to include instructions on the use of temporary penetration seal repairs.

6. STP M-70A is being revised to update and clarify existing text and instructions. The most significant change to this procedure is the addition of instructions on the implementation of the new penetration seal temporary repair.

7. STP M-70C is being revised to add an additional instruction on the use of temporary administrative controls on non-functional fire doors by installing STOP signs on both sides of the door.

8. STP M-70C is being revised to add additional information on the inspection of heating, ventilating, and air conditioning doors with dogs.

9. STP M-70C is being revised to add an inspection sheet for non-preventative maintenance program doors.

Safety Evaluation Summary:
ECG 18.7 provides positive procedural control over fire rated assemblies. The FSARU states: "For each fire hazard, a suitable combination of prevention, detection and suppression capability, and ability to withstand the effects of a fire shall be provided." Therefore, in the event that a fire rated assembly becomes inoperable, appropriate compensatory measures must be taken while the assembly is being restored to an operable status. Typically, compensatory measures, such as fire watches and fire detection or the detection capability of the automatic fire suppression, are used. The proposed changes to ECG 18.7, STP M-70A, and STP M-70C include adding two new additional compensatory measures and adding an alternative method for providing fire detection.

In order to improve ECG 18.7, a new ECG 18.7 has been created. The new ECG:

1) uses the ITS format;
2) improves clarity;
3) establishes links between the fire-protection ECGs;
4) provides additional information for verifying flow switch operability; and,
5) provides useful information regarding system operability.

The ITS format was developed to: increase user acceptance, improve access to information, provide human factors guidance, and to promote consistency in content, format, and style.

In addition, the scope of ECG 18.7 has been clarified and expanded to include hatches located in fire-rated barriers and credited cable-tray firestops.

Conclusion:
ECG 18.7 ensures that fire damage is limited, such that one train of safe shutdown equipment necessary to achieve and maintain safe shutdown is always available. The proposed changes to ECG 18.7, STP M-70A, and M-

70C do not eliminate any fire-rated assemblies from the scope of the ECG and do not reduce the level of fire protection at DCPP. Therefore, the ability to safely shut down the plant has not been adversely affected. This activity does not result in a USQ.

01-015 HELB Barrier Requirements

Reference Document No.: DCP N-049565
Rev. No: 0
Reference Document Title: HELB Barrier Requirements
Safety Evaluation Description:
Revise the HELB Barrier between Area H and K in the Auxiliary Building to detail barrier requirements at elevation 140 ft and above, involving certain sections of the central stairwell walls in the Auxiliary Building. The location of the HELB barrier was previously defined in Chron 199592, a reference of DCM T-12. While this document identified a HELB barrier on Elevations 85, 100, and 115, the barrier extending up the central stairwell along the same wall as shown on the lower elevations was not detailed. This design change also includes the documentation of the qualification of SSCs (structure, doors, penetration seals) that form the HELB barrier, to withstand the required HELB pressure as established by existing analyses. The existing Area H/K compartment-pressurization analyses (Calculation M-493 Revision 2) establish the required pressure that the HELB barrier between Area K and the mild environment of Area H must withstand.

Safety Evaluation Summary:
FSARU Figures 3.6-10 and 3.6-14 are affected because the associated drawings (515944 and 515948) are updated by the design change to correctly depict the boundaries of Area 20 (the Auxiliary Building central stairwell) at the 73 and 140 foot elevations, in accordance with the existing Area H/K compartment pressurization analyses. Additionally, FSARU Figures 9.5 F-10 and 9.5 F-11 are affected because the associated drawings (515571 and 515572) are updated by the design change to detail the HELB barriers between Area H and K at the 140 foot elevation and above, according to Chron 236662. The only impact to the FSARU is the changes to these figures. The Area H/K compartment pressurization analyses that form the basis for the location of HELB barriers and required qualification pressures, have not been revised. This design change includes the correct depiction of one of the subcompartment boundaries as already contained in the analysis (for Area 20), and the resulting definition of the HELB barrier along the same wall as previously defined but not detailed. The design documents have been updated to consistently describe the Area 20 compartment designations, and resulting HELB barrier requirements, to meet existing DCPP licensing commitments and pressurization-analysis design basis.

Conclusion:
As evaluated by the 10CFR 50.59 safety evaluation, this change to the FSARU figures does not result in a USQ. A LA is not required to implement this change.

01-018 ECG 7.7, Revision 3

Reference Document No.: ECG 7.7
Rev. No: 3
Reference Document Title: Reactor Vessel Head Vents
Safety Evaluation Description:
SR 7.7.c. has been modified by deleting the phrase "during venting" and adding the phrase "or with the core off loaded (no Mode)."

Safety Evaluation Summary:
ECG SR 7.7c has been revised to delete the words "during venting" to allow an alternate method of performing the surveillance test. Also, the option to perform the SR with the core off loaded (no Mode) will allow scheduling flexibility.

The reactor head vent valves were added as a post-TMI item as required by NUREG-0737, item II.B.1. NUREG-0737, page II.B.1-2, states, "...the size of the reactor coolant vents is not a critical issue." Therefore, flow rate through the vents was not an important design consideration. The important safety function is to allow the venting of noncondensible gases and/or steam from the RCS that could inhibit natural circulation core cooling.

The original TS SR for the flow path was implemented by venting gas through the vent valves during the RCS venting process, which removes air from the reactor following reassembly. The test process was revised in 1991 to pressurize the head-vent piping with nitrogen up to the first off isolation valve, and then to verify the flow path by observing the pressure in the piping drop as it enters the reactor head. Use of this reverse flow into the reactor head is acceptable, because there are no check valves between the reactor head and exhaust of the vent valves. The flow path from the reactor head to the vent valves and associated piping is an unused part-length CRDM housing, which is essentially a pipe. Performing the test with the head on the reactor vessel could cause RCS pressure to increase. However, there is no risk of overpressurizing the RCS, since the nitrogen supply is regulated to between 200 and 300 psig. This pressure range is significantly lower than the LTOP setpoint of 435 psig. In addition, the STP provides controls to ensure that the correct nitrogen supply is used to pressurize the piping and that it is isolated from the head vent piping at the nitrogen-supply bottle prior to opening the flow path into the reactor head. The addition of the option to perform the test with the core off loaded provides flexibility for scheduling the test. Since the revised test methodology does not use RCS gas for verifying flow, the reactor vessel head does not need to be installed on the reactor vessel to perform the test.

Conclusion:
The proposed changes to ECG SR 7.7.c. retain the flow verification testing requirements necessary to establish the operability of the reactor vessel head vent valves and provide scheduling flexibility for performing the test. The changes do not result in a USQ.

01-019 Unit 1 10 Percent Load Rejection with LTB Inhibited for PMT 03.11

Reference Document No.: PMT 03.11
Rev. No: 0
Reference Document Title: Unit1 10 percent Load Rejection with LTB Inhibited for PMT 03.11
Safety Evaluation Description:
DCP J-49419 replaced the Lovejoy turbine speed control system with a digital Woodward turbine speed control system. As part of the PMT of the new design, PMT 03.11 was written to provide a MFW pump turbine control system stability test for a 10 percent step-load transient at 100 percent rated thermal power. The load transient bypass (LTB) is inhibited for the test and a 10 percent reactor thermal power step-load decrease is performed at 2,400 mw/min. In accordance with OP L-4 and OP C-3 III. Proper operation of the MFW pumps is verified, and the plant is restored to normal. The test was written as a special test as defined in AD13.ID1.

Safety Evaluation Summary:
PMT 03.11 is performed at approximately 100 percent reactor power. The test involves observing and documenting the effect of a Category 1 design-based load transient. All plant equipment will be in auto, controlling operations and the response of the transient. The LTB feature will be inhibited from actuation as to preclude undesirable effect of actuation. Operator action during the test is to monitor plant parameters and only if necessary respond to unanticipated response as with any test performed. Procedure precautions and tailboard address the critical plant parameters and responsibilities during the test.

The performance of this test does not impact any licensing commitments or requirements as this is a plant feature not required for safe shutdown.

The LTB relay actuation is not credited for function or operation during an accident or event and is removed during this test as a conservative measure as not being required for the transient induced by the test.

Conclusion:
The 10 CFR 50.59 safety evaluation has demonstrated that no USQs have been created as a result of performance of PMT 03.11.

01-022 Revision to ECG 64.1

Reference Document No.: ECG 64.1
Rev. No: 2
Reference Document Title: Electrical Equipment Protective Devices - Motor-Operated Valves Thermal Overload Protection and Bypass Devices

Safety Evaluation Description:
The Bases for ECG 64.1 states the following: "A list of the ECG-controlled MOV thermal overload protection and bypass devices is maintained in the Diablo Canyon plant procedures. The administration of the list shall be conducted in accordance with Section 50.59 of 10 CFR Part 50 and the provisions in the Administrative Controls Section of the TS. Records of the changes to the valve list are maintained, and an annual report is made that includes a brief description of changes and a summary of the safety evaluation of each in accordance with 10 CFR 50.59."

This change deletes the last sentence so that the requirements of 10 CFR 50.59 will govern maintenance of records of changes to the list of ECG-controlled motor operated valve thermal overload protection and bypass devices, and reporting of those changes to the NRC.

Safety Evaluation Summary:
LAs 79 and 78, for DCPP Units 1 and 2 respectively, relocated TS Table 3.8-1, "Motor-Operated Valves Thermal Overload Protection and Bypass Devices," to DCPP procedures in accordance with the guidance provided in GL 91-05, "Removal of Component Lists from Technical Specifications," dated May 6, 1991. The list is currently located in AD13.DC1. In proposing the changes issued in LAs 79 and 78, PG&E stated that any changes to the table would constitute a change to the facility, and thus would be subject to the provisions of 10 CFR 50.59. This intent was stated in the TS Bases issued with the LAs.

LAs 120 and 118, for DCPP Units 1 and 2, respectively, authorized relocation of the TS requirements for motor-operated valves thermal overload protection and bypass devices (TS 3/4.8.4) to an ECG (ECG 64.1). In approving the relocation, the NRC stated that the technical requirements relocated to licensee controlled documents (ECGs) would be controlled in accordance with 10 CFR 50.59.

Because the change is administrative in nature, it will not impact operation of the plant. No plant equipment or accident analyses will be affected. Additionally, the change will not relax any criteria used to establish safety limits, will not relax any safety systems settings, nor will it relax the bases for any limiting condition for operation.

Conclusion:
The change to the ECG Bases is an administrative change for consistency with the regulations, does not impact the operation of the plant, and will not adversely impact the health and safety of the public. This activity does not result in a USQ.

01-023 Vacuum Refill Procedure OP A-2:IX (Unit 2)

Reference Document No.: OP A-2:IX
Rev. No: 0
Reference Document Title: Reactor Vessel - Vacuum Refill of the RCS
Safety Evaluation Description:
The new operations procedure for vacuum refill of the RCS provides guidance for refilling the RCS and forming a bubble in the pressurizer under vacuum conditions. With the plant in Mode 5 and the RCS pressure boundary intact, the desired vacuum (absolute pressure) is established, and the RCS will be filled to approximately 70 percent pressurizer level. Once the pressurizer has been filled to 70 percent level, the RCS temperature will be raised to 150-160 degrees F and the pressurizer heaters will be energized to begin pressurizing the RCS. When a steam bubble has been formed in the pressurizer and the appropriate pressure has been reached, the reactor coolant pumps can be started.

Safety Evaluation Summary:
The modified operating instructions contained in OP A-2:IX for refilling the RCS under vacuum conditions and for operation of the pressurizer and reactor coolant pumps in Mode 5, provide an alternate method of filling and venting the RCS. This procedure can be used in place of OP A-2:I, "Reactor Vessel – Filling and Venting the RCS," when plant conditions make the use of the vacuum-refill method more desirable.

Conclusion:
The use of the proposed, modified operating procedure for filling and venting the RCS and forming a bubble under vacuum will not impair the safety function or performance of any plant equipment. Based on the above information, the use of the vacuum-refill process will not adversely affect the safe operation of the plant, does not represent a potential USQ, and does not require a change to plant TSs.

01-024 CFCU Time-Delay Relay Replacement

Reference Document No.: DCP E-050547
Rev. No: 0
Reference Document Title: CFCU Time-Delay Relay Replacement
Safety Evaluation Description:
Time-delay relays are used in the control circuits for the CFCUs to provide appropriate start sequencing following various postulated plant events such as LOCA, MSLB, and loss of offsite power (LOOP). The previously installed Agastat relays were the subject of a 10 CFR 21 report and became obsolete since the original vendor no longer supports this product line. Replacement Allen-Bradley relays were installed for these devices. At the same time, another time-delay relay (with a time-delay setting of 0 seconds) was removed. A new interposing relay (Potter & Brumfield) was installed to simplify circuitry and maintain the control circuit functions as before. All of these modifications are physically contained within the motor control centers (MCCs) and auxiliary relay panels of the 480 V vital switchgear. The changes implemented by this DCP maintain the same time-delay settings and CFCU functions.

Safety Evaluation Summary:
The CFCUs are used for the mitigation of such accidents/events as LOCA, MSLB, and LOOP. All of the modifications made under this DCP maintain the present design, ratings, qualification, installation, performance, function, and failure modes of the current equipment. The inputs, assumptions, and performance of SSCs credited in safety analyses for DCPP are not affected. Therefore, the previous evaluations of accidents and malfunctions in the DCPP licensing basis are not impacted.

The only licensing-basis document revision is updating the FSARU figures showing the electrical scheme and logic for the CFCUs. Of the CFCU and vital power systems' design and functional requirements and licensing commitments, none is impacted by the modifications. There is no resulting change to the way the CFCUs operate or how their operation will be periodically tested. The DCP effects are transparent to the operation of the CFCUs.

Conclusion:
The replacement relays and wiring modifications maintain the same qualification, function, and performance of the CFCUs. There is no compromise to the accident-mitigating capabilities of the CFCUs or the 480 V vital power system credited in the DCPP licensing basis. There is no change to the existing failure modes or consequences. The CFCUs continue to meet the timed-sequence requirements as defined in the bases of the ITS and ECGs, thus assuring ESF equipment responses that are consistent with accident analyses. No USQ results from implementation of the modifications.

01-025 DCPP Unit 2 Cycle 11 Reactor Core Fuel Load and COLR

Reference Document No.: DCP N-50537
Rev. No: 01-025
Reference Document Title: DCPP Unit 2 Cycle 11 Reactor Core Fuel Load and Core Operating Limits Report
Safety Evaluation Description:
This DCP incorporates the new fuel loading pattern for Cycle 11 into the plant design. This design also evaluates and accepts equivalency between the Westinghouse Rod Cluster Control Assembly (RCCA) and Framatome RCCA for use at DCPP. 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. The installation of the Framatome models will be based on the results of eddy-current testing performed during 2R10 on the original RCCAs.

Safety Evaluation Summary:
The RSE identified limits on Cycle 11 operation in concluding that there were no USQs resulting from the Unit 2, Cycle 11 core design.

The Framatome Safety evaluation demonstrates the RCCA design differences versus the DCPP resident Westinghouse model. The dimensional and material configuration of the Framatome model is described as equal to or superior to the Westinghouse model. The improved features include the use of ion-nitriding on the cladding surface, low contaminant 316L SS cladding material, improved rod-to-spider connections, and a single spider casting. The absorber is tapered at the tip to accommodate cladding swelling in the tip that is subject to neutron fluence due to its proximity to the active fuel. These upgrades have the purpose of improving wear resistance and ensuring at least 20 EFPYs of life.

The impact of the new chemistry Lithium concentration limits has been evaluated to maintain corrosion effects below design limits for fuel cladding, SG tube and other RCS components.

Conclusion:
The evaluation performed in the Westinghouse RSE for Cycle 11 and the additional evaluations performed as part of this design change ensure that the nuclear fuel for Cycle 11 is designed in accordance with the proper licensing and design-bases and that no impact on nuclear safety results from its implementation.

The evaluation performed in the Framatome safety analysis and the additional evaluations performed as part of this design change evaluation ensure that the manufacturing and performance features of the Framatome RCCAs rely on the proper design-bases such that no impact on nuclear safety results from their use in the reactor core. This activity does not result in a USQ.

01-026 CFCU Time-Delay Relay Replacement
Reference Document No.: DCP E-049547
Rev. No: 01-026
Reference Document Title: CFCU Time-Delay Relay Replacement
Safety Evaluation Description:
Time-delay relays are used in the control circuits for the CFCUs to provide appropriate start sequencing following various postulated plant events such as LOCA, MSLB, and LOOP. The previously-installed Agastat relays were the subject of a 10 CFR 21 report and became obsolete since the original vendor no longer supports this product line. Replacement Allen-Bradley relays were installed for these devices. At the same time, another time-delay relay (with a time-delay setting of 0 seconds) was removed. A new interposing relay (Potter & Brumfield) was installed to simplify circuitry and maintain the control circuit functions as before. All of these modifications are physically contained within the MCCs and auxiliary relay panels of the 480 V vital switchgear. The changes implemented by this DCP maintain the same time-delay settings and CFCU functions.

Safety Evaluation Summary:

The CFCUs are used for the mitigation of such accidents/events as LOCA, MSLB, and LOOP. All of the modifications made under this DCP maintain the present design, ratings, qualification, installation, performance, function, and failure modes of the current equipment. The inputs, assumptions, and performance of SSCs credited in safety analyses for DCPP are not affected. Therefore, the previous evaluations of accidents and malfunctions in the DCPP licensing basis are not impacted.

The only licensing-basis document revision is updating the FSARU figures showing the electrical scheme and logic for the CFCUs. Of the CFCU and vital power systems' design and functional requirements and licensing commitments, none is impacted by the modifications. There is no resulting change to the way the CFCUs operate or how their operation will be periodically tested. The DCP effects are transparent to the operation of the CFCUs.

Conclusion:
The replacement relays and wiring modifications maintain the same qualification, function, and performance of the CFCUs. There is no compromise to the accident-mitigating capabilities of the CFCUs or the 480 V vital power system credited in the DCPP licensing basis. There is no change to the existing failure modes or consequences. The CFCUs continue to meet the timed-sequence requirements as defined in the bases of the ITS and ECGs, thus assuring ESF equipment responses that are consistent with accident analyses. No USQ results from implementation of the modifications.

01-027 Pyrocrete Enclosure Thickness

Reference Document No.: FHARE 145
Rev. No: 1
Reference Document Title: Pyrocrete Enclosure Thickness
Safety Evaluation Description:
Along the East wall of the Unit 1 and 2, 12 kV Switchgear Rooms (Fire Areas 10 and 20, respectively), the 4 kV Bus D bus duct and the 4 kV Bus E bus duct runs adjacent to Enclosures 10-28-6 and 20-29-1, such that an adequate thickness and configuration to comply with the 2 hour tested configuration can not be obtained. According to Omega Point Laboratory's fire endurance test a Pyrocrete thickness of 2 inches, with an air gap of 2 inches between the interior edge of the Pyrocrete and the conduit, is required to provide the necessary 2 hour fire rating for these enclosures. However, because of the interference with the Unit 1, 4 kV Bus D and Unit 2, Bus E bus ducts, a small section in both of these areas (4 inches by 18 inches and 24 inches by 21 inches for Enclosures 10-28-6 and 20-29-1, respectively) is only capable of obtaining a 1-1/2 inch thickness of Pyrocrete which would be positioned against the 4 inch conduit inside the enclosure. While a test report for Pyrocrete 241 does show that the 1-1/2 inch thickness will provide a 2 hour rating, this configuration does not match the tested fire endurance test for the enclosures. Since these 4 inch by 18 inch and 24 inch by 21 inch sections do not comply with an approved tested configuration, these configurations were analyzed in FHARE 145, Rev. 1, to determine if they will withstand the hazard associated with the area and to determine if the configuration will affect the ability to safely shut down the plants in the event of a fire in the fire area.

Safety Evaluation Summary:
While Enclosures 10-28-6 and 20-29-1, will have small areas (4 inches by 18 inches and 24 inches by 21 inches, respectively) that do not comply with the tested configuration an adequate thickness of Pyrocrete will be installed to provide reasonable assurance that a fire in the area will not impact the safe shutdown circuits in the enclosures. 1-1/2 inches of Pyrocrete will provide a 2 hour fire rating and while this thickness was not used in the fire endurance test performed at Omega Point Laboratories it does provide reasonable assurance that the reduced Pyrocrete thickness will adequately protect the circuits from the hazards in the area. As noted in FHARE 145, Rev. 1, the fire loading for each of these areas is less than 45 minutes, with a majority of that fire loading located on the 76-ft elevation. This fact coupled with the fact that this reduction in Pyrocrete area is approximately 10-ft above the floor on the 85-ft elevation the possibility of a fire of significant nature propagating to this area of the enclosure is remote.

Conclusion:
Because of the low combustible loading in the area, the location of the reduced Pyrocrete cross-sectional areas, the presence of an automatic smoke detection system, these configurations are expected to withstand the hazards of the area. Redundant safe-shutdown circuits will remain free from fire damage and the ability of the plant to achieve and maintain safe-shutdown will not be affected. This activity does not result in a USQ.

01-028 COLR for DCPP Unit 2 Cycle 11

Reference Document No.: COLR 2-11
Rev. No: 0
Reference Document Title: COLR for DCPP Unit 2 Cycle 11
Safety Evaluation Description:
The COLR is updated every fuel cycle to reflect the new core design. The FQ margin penalties in excess of 2 percent per 31 EFPD change, as well as the load follow W(z) factors.

Safety Evaluation Summary:
The RSE identified limits on Cycle 11 operation in concluding that there were no USQs resulting from the Unit 2, Cycle 11 core design.

The Framatome Safety evaluation demonstrates the RCCA design differences versus the DCPP resident Westinghouse model. The dimensional and material configuration of the Framatome model is described as equal to or superior to the Westinghouse model. The improved features include the use of ion-nitriding on the cladding surface, low contaminant 316L SS cladding material, improved rod-to-spider connections, and a single spider casting.

The absorber is tapered at the tip to accommodate cladding swelling in the tip that is subject to neutron fluence due to its proximity to the active fuel. These upgrades have the purpose of improving wear resistance and ensuring at least 20 EFPYs of life.

The impact of the new chemistry Lithium concentration limits has been evaluated to maintain corrosion effects below design limits for fuel cladding, SG tube and other RCS components.

Conclusion:
The evaluation performed in the Westinghouse RSE for Cycle 11 and the additional evaluations performed as part of this design change ensure that the nuclear fuel for Cycle 11 is designed in accordance with the proper licensing and design bases and that no impact on nuclear safety results from its implementation.

The evaluation performed in the Framatome safety analysis and the additional evaluations performed as part of this design change evaluation ensure that the manufacturing and performance features of the Framatome RCCAs rely on the proper design bases such that no impact on nuclear safety results from their use in the reactor core. This activity does not result in a USQ.

01-029 ECG 45.3

Reference Document No.: ECG 45.3
Rev. No: 2
Reference Document Title: Electrical Power Systems – Containment Penetration Conductor Overcurrent Protective Devices

Safety Evaluation Description:
This change revises the bases section of ECG 45.3 to clarify which penetration protective devices are covered under ECG SR 45.3.5. This SR only applies to circuit breakers and relays that have manufacturer's recommended maintenance. This SR does not apply to thermal overload relays. Thermal overload relays are tested in accordance with SR 45.3.4, which satisfies the maintenance requirement of SR 45.3.5. This change also revises the reference section of ECG 45.3 by deleting the outdated attachment numbers of STP M-83A. The reference change is considered an editorial correction.

Safety Evaluation Summary:
This change is a clarification of an old TS SR that was based on an 18 month refueling cycle. The frequency of 60 months was based on three - 18 month refueling cycles, with 6 months of margin. Since we have changed to a 24 month refueling cycle, we currently function test the thermal overload

relays on a 72 month frequency, not to exceed 96 months, in accordance with ECG 45.3.4.

The intent of the SR was to verify that maintenance was performed on circuit breakers every 60 months to ensure the operating mechanism was able to perform its function of tripping open to protect the penetration. The original wording of the SR used the term "circuit breaker." When the TS SR was converted to an ECG, the term "circuit breaker" was replaced with "penetration protective device" to include the overcurrent relays on the RCP breakers. Circuit breakers and the overcurrent relays for the RCPs have manufacturer's recommended maintenance, which is required to be performed to ensure the devices can function to provide overcurrent protection.

Thermal overload relays have no vendor recommended maintenance. A function test of the thermal overload relay is performed to verify operability. The function test is performed to satisfy ECG SR 45.3.4.

The testing to satisfy ECG SR 45.3.4 is the same as the maintenance to satisfy SR 45.3.5. Therefore, to avoid duplication and reduce administrative burden, the thermal overload relays should be excluded from the requirement of ECG SR 45.3.5. The current frequency at which the relays are required to be tested, every 96 months in accordance with ECG SR 45.3.4, will remain the same. The safety evaluation to test the thermal overload relays on a 96 month frequency was approved when PG&E extended the surveillance frequency to account for a 24 month fuel cycle. This change results in going from a 60 month maintenance frequency to a 96 month testing frequency. This increase in frequency is acceptable based on operating, surveillance, and maintenance histories.

ECG SR 45.3.5 is verified by the performance of STP M-83B. As a result of this change, Engineering will no longer have to do a history search and paper review of past-performed dates for thermal overload relays for STP M-83B.

Conclusion:
This change to the bases section of ECG 45.3 clarifies which penetration protection devices are required to have maintenance performed within 60 months in accordance with ECG SR 45.3.5. This change restores the ECG to the original TS design bases, and does not result in a USQ.

01-030 Revision 1 to ECG 2.1

Reference Document No.: ECG 2.1
Rev. No: 1
Reference Document Title: Primary-to-Secondary Leakage Monitoring

Safety Evaluation Description:
This change modifies the applicability statement for ECG 2.1 to indicate that the ECG is applicable in Modes 1-3 only when a main steam isolation valve (MSIV) is open with vacuum in the condenser. Specifically, the ECG 2.1 applicability statement is changed from:

MODES 1 and 2.

Mode 3 with an MSIV open and vacuum in condenser.

To the following:

MODES 1, 2, and 3 with an MSIV open and vacuum in condenser

The Bases have been updated accordingly.

Safety Evaluation Summary:
This is an administrative change to correct the ECG 2.1 applicability statement to make it read as it was originally intended. ECG 2.1 was created by moving the contents of ECG 39.2, "Air Ejector Gaseous Effluent Monitors," to ECG 2.1 with no change, and by adding additional requirements for controlling the steam jet air ejector wide-range flow rate channel, FIT – 81. When the ECG 39.2 applicability statement was relocated to ECG 2.1, it was inadvertently modified, and the modification was not caught in subsequent reviews. The change converts the ECG 2.1 applicability back to the way it was originally stated in ECG 39.2.

Conclusion:
This change is administrative only, and does not result in a USQ.

01-031 Removal of Radwaste Process Control Program Changes from PSRC Review
Reference Document No.: Multiple
Rev. No: Multiple
Reference Document Title: See Description Below
Safety Evaluation Description:
Remove PSRC review of changes to the radwaste Process Control Program (PCP) from OM4.ID2, RP2, RP2.DC2, RP2.DC3, section 13.5.2.1 of the FSARU, and T36104 of the Procedure Commitment Database.

Safety Evaluation Summary:
The ITS relocated the radwaste PCP to the FSARU and plant procedures. The Offsite Dose Calculation Manual (ODCM) was retained in the ITS, but review of changes to ODCM by the PSRC was eliminated. The review level for changes to the radwaste PCP needs be no more restrictive than the ODCM, which is reviewed and approved by the Station Director. In addition, the FSARU (Chapter 17) and plant procedures still require PSRC to review evaluations of proposed changes to the ODCM and the PCP that are performed under 10 CFR 50.59.

Conclusion:
Elimination of PSRC review of changes to the radwaste PCP will not decrease the margin of safety and does not result in a USQ.

01-032 Unit 1 VCT Argon Injection Jumper

Reference Document No.: A0527475
Rev. No: 01-032
Reference Document Title: VCT Argon Injection Jumper
Safety Evaluation Description:
Proposed Change
This jumper/activity injects argon gas (argon-40) into the VCT to increase the argon-41 concentration in the reactor coolant. A compressed gas bottle, pressure regulator, flow indicating controller, and check valve will be connected to the Class II zinc injection line located behind the primary sample sink. This line discharges to the VCT liquid space. The maximum argon gas injection rate is 10 standard cubic centimeters per minute (scc/min) and is controlled manually by the flow indicating controller. The jumper pressure regulator is set above VCT pressure to provide a motive force with a maximum of 60 psig, which is below the maximum VCT pressure of 75 psig. RCS argon-41 concentration for this jumper/activity is limited to 0.2 uci/cc or less and is monitored by sampling of the RCS. A chemistry test procedure provides instructions for controlling injection flow rate and sampling. Argon is not currently injected into the VCT or reactor coolant.

Reason for Proposed Change
The purpose of the jumper/activity is to: (1) obtain a correlation between argon injection rate and reactor coolant argon-41 concentration; and (2) determine the suitability of the injection method. Note: This jumper/activity and evaluation does not take credit for satisfying any SG leak detection requirements or commitments. This information will be used to maintain argon-41 concentration for identifying primary-to-secondary SG tube leakage. Increased Argon-41 concentration should allow detection of a 30 gallon per day SG tube leak via the Condenser Steam Jet Air Ejector Radiation Monitor RE-15 if this method proves practical. Industry guidelines (i.e., NEI 97-06,

EPRI Primary-to-Secondary Leak Guidelines TR-104788, Rev.2) require detection to 30 gpd but does not specify a method of achieving it.

Safety Evaluation Summary:

DCPP is committed to implement NEI 97-06, "Steam Generator Program Guidelines," which requires the capability of continuous on-line monitoring that can detect a 30 gpd SG primary-to-secondary tube leak. This commitment requires an operable radiation monitor that can produce an alarm in the main control room at a leak rate of 30 gpd based on RCS activity levels. Addition of argon gas to the RCS is used to increase argon-41 in the RCS. Increased argon-41 in the RCS provides the required primary-to-secondary leak radiation monitor sensitivity.

There is no impact to plant operations or to operating procedures. The VCT will continue to be aligned and operated as currently done. There is no change or impact to VCT level, temperature, or pressure control or indication functions. There are no changes to the components that control or indicate VCT level, temperature, or pressure.

Prudent engineering features have been incorporated into the jumper to assure proper operation. The jumper is connected to Class II tubing outside safety-related pressure boundaries. An adjustable pressure regulator with pressure indication is provided to allow observation and adjustment of injection pressure.

Injection of argon gas into the VCT liquid space allows the gas to dissolve into the process liquid or degas into the VCT gas space. The liquid at the surface is in equilibrium with the gas space; however, the liquid capacity for dissolved gas below the surface increases with depth (increased pressure) allowing absorption of argon. Argon that does not dissolve enters the gas space and mixes with hydrogen. At equilibrium, argon is less than 1 percent of the VCT gas space concentration. The VCT will continue to control dissolved gases and not introduce voids into the ECCS systems.

Conclusion:
The licensing-basis review for this jumper finds it is within current licensing requirements. There is no impact to VCT operation, including charging injection supply or RCS letdown. The jumper/activity has no impact on VCT tank rupture accident frequency or consequences. This activity does not result in a USQ.

01-033 Removal of a Regulatory Commitment to Maintain and Analyze the Turbine Valve Failure History Database

Reference Document No.: Westinghouse MUHP7002/8002
Rev. No: 0
Reference Document Title: Final Update and Evaluation of BB-95/96 Turbine Valve Failure Database

Safety Evaluation Description:
As part of LAR 88-02, which proposed a change to the frequency of the main turbine valves stroke testing (STP M-21C) from weekly to quarterly, PG&E committed to participate in developing and maintaining the turbine valves failure database, to include it in the FSARU, to update it at least once every three years, and to reevaluate it if a significant upward trend was identified.

In the Safety Evaluation of LA 42/41, the NRC accepted such commitment in their evaluation. PG&E subsequently included the commitment in the FSARU.

DCPP recently completed the 10 year project (Westinghouse Turbine Valve Test Frequency Evaluation Subgroup, WOG TVTF) of developing and analyzing the valve failure database and issued the final report (WOG-TVTF-00-014). The report concluded that based on the low historical turbine valve failure rate, the current quarterly valve testing was well substantiated and did not require continued formal tracking (and evaluation) of turbine valve failures.

Therefore, it is recommended that the commitment made in LAR 88-02 be removed.

Safety Evaluation Summary:
The answers to all 50.59 questions are negative based on the facts that;

- The valve failure database did not show any time-dependent valve failure mechanism.
- The valve failure rates are not expected to significantly change from the current level as the BB-95/96 fleet ages,
- The turbine missile generation is not considered a credible event at DCPP.
- The proposed activity will not change the bases of such conclusion,
- This is a document only change; there is no physical change to the valves that might affect their functions.

Conclusion:
The analysis of the BB-95/96 turbine valve failure database spanning over 10 years demonstrated that the turbine valve failure rate, which was a key

parameter used in the turbine missile probabilistic analysis, had remained low and it is not expected to change (i.e., increase) as the BB-95/96 turbine fleet ages. Formal continued tracking and analysis of the industry-wide valve failures is no longer necessary and the removal of such commitment should not result in a USQ.

****

NOTE: The revised 10 CFR 50.59 rule was implemented at DCPP on July 17, 2001. LBIEs starting with No. 01-037 were performed under the new rule, except No. 01-039 which was initiated under the old rule, and was completed under the provisions of the old rule. For this report, the significant changes in the evaluation summaries from this point forward (except No. 01-039) are:

1. The headings "Safety Evaluation Description," "Safety Evaluation Summary," and "Conclusion" are replaced with "Activity Description," and "Summary of Evaluation."
2. The term "Unreviewed Safety Question," or "USQ" is replaced with "requires prior NRC approval," or equivalent terminology.

****

01-037 ECG 11.1 Revision to Eliminate PASS

Reference Document No.: ECG 11.1
Rev. No: 8
Reference Document Title: Post Accident Sampling System (PASS)
Activity Description:
LAs 149 (Unit 1) and 149 (Unit 2) dated July 13, 2001, approved elimination of PASS, provided three commitments were established. One of these commitments is to develop contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump fluid, and containment atmosphere (Ref. AR A0523097, Evaluation No. 7). To support meeting this commitment, selected portions of PASS will be retained for use in obtaining these samples and will be controlled by ECG 11.1. ECG 11.1 has been modified to reflect the significant reduction in licensing requirements for PASS.

Summary of Evaluation:
The proposed revision to ECG 11.1 fundamentally alters the existing means of performing or controlling design functions of PASS, and the design functions themselves have been changed significantly by LAs 149/149. In accordance with NEI 96-07, Revision 1, Section 4.2.1.2, changes that fundamentally alter the existing means of performing or controlling design functions should be conservatively treated as adverse and screened in. However, the proposed revision implements LAs 149/149 which allow elimination of PASS. PG&E has elected to maintain and utilize portions of PASS to support fulfilling its commitment to have a contingency plan for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump fluid, and containment atmosphere. All eight 50.59 evaluation questions answered "No." Therefore, the proposed change does not require prior NRC approval.

01-038 ECG 40.1 Revision to Incorporate Backup Meteorological Tower Instrumentation

Reference Document No.: ECG 40.1
Rev. No: 1
Reference Document Title: Meteorological Instrumentation
Activity Description:
This procedure is being revised to incorporate administrative controls for the backup meteorological tower instrumentation. The backup system performs in an identical manner to the primary system by providing the necessary meteorological data for offsite dose calculations used in emergency plan procedures. Although the backup system is not currently controlled by ECG 40.1, it is relied upon to provide meteorological data in the event the primary system is inoperable. This change provides additional assurance that the backup meteorological instrumentation is operable in such an event.

The special report currently required in the event of total loss of primary instrumentation has been deleted because any event that results in major loss of emergency assessment capability, specifically loss of primary and backup meteorological indication of wind speed, wind direction, or air temperature delta-T, will be reportable as a non-emergency event in accordance with 10 CFR 50.72(b)(3)(xiii). This requirement has been incorporated into ECG 40.1 as a note in the Required Actions and refers to procedure XI1.ID2 for further guidance.

Summary of Evaluation:
The proposed revision to ECG 40.1 fundamentally alters the existing means of performing or controlling design functions of the meteorological instrumentation. In accordance with NEI 96-07, Revision 1, Section 4.2.1.2, changes that fundamentally alter the existing means of performing or controlling design functions should be conservatively treated as adverse and screened in. However, the proposed revision provides additional assurance that the backup meteorological instrumentation will be operable in the event of loss of primary instrumentation. Furthermore, the reporting requirements of 10 CFR 50.72(b)(3)(xiii) will ensure that NRC is notified of any event that results in a total loss of primary and secondary meteorological instrumentation indication of wind speed, wind direction, or air temperature delta-T.

All eight 50.59 evaluation questions answered “No.” Therefore, the proposed change does not need prior NRC approval.

01-039 Upgrade of PORV Automatic Actuation Circuitry

(Note: This change was evaluated under the old 10 CFR 50.59 rule.)

Reference Document No.: DCP J-49569

Rev. No: 0

Reference Document Title: Upgrade of PORV Automatic Actuation Circuitry

Safety Evaluation 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 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 Class II portions of the instrument loops (actuating the relays directly from Eagle21). Then the automatic actuation of the PORV can be credited for ensuring that the pressurizer safety valves do not provide pressure relief for subcooled water during an inadvertent SI at power.

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

The PT-403A and PT-405A (alternate LTOP) transmitter signals will be processed through Eagle21. The alternate LTOP transmitter channels from Eagle21 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 the PT-403 and PT-405 (via PR-403 and PI-405) will continue to be performed by PT-403 and PT-405.

Safety Evaluation Summary:

This change will have the following impact on plant operations:

1. By actuating the PORVs directly from Eagle21, the master controller(PC-455K) will no longer control PCV-455C. In addition, the P/455A selector switch will no longer select the pressure transmitter for PCV-456 actuation. In order to preserve the pressurizer pressure control function (Ref. FSARU Figure 7.7-4), PCV-474 actuation will be transferred to the master controller.
2. The AOP/EOPs will have to be revised to lead operators to verifying operation of the PORVs.
3. The use of PT-403A and PT-405A will change the instrumentation used for LTOP and verification of conditions for opening 8701/8702. The PT-

403A and PT-405A instrument loop will be displayed on VB2 (PI-403A and PI-405A).

The changes that will affect the licensing requirements are the upgrade of the PORV automatic actuation circuitry and the swapping of the control function to PCV-474 (FSARU Figure 7.7-4). This change is adding 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 limiting conditions for operation and the SRs. Since the new protective function has not been previously reviewed by the NRC, it is being considered a USQ. LAR 01-08 is being submitted as a USQ for NRC review.

Since LAR 01-08 will credit the automatic actuation of the PORVs to mitigate the consequences of the inadvertent SI at power accident, this LBIE will only evaluate 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 control for PCV-455C/456, and the swapping of the control valve for pressurizer pressure control.

Conclusion:
The LBIE has concluded 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 the swapping of the pressurizer pressure control valves do not constitute a USQ. This portion of this design change will not create/delete any new safety functions, alter any licensed safety functions, or alter the licensed method of performing a safety function.

The use of Westinghouse Letter PGE-99-534 and the PORV automatic-actuation circuitry to address the mitigation of the Inadvertent SI at power accident constitutes a USQ. It will require NRC approval prior to crediting the automatic actuation of the PORVs to mitigate the consequences (Ref. LAR 01-08).

01-040 Reactor Coolant System Vacuum Refill

Reference Document No.: DCP N-049532
Rev. No: 0
Reference Document Title: Reactor Coolant System Vacuum Refill
Activity Description:
This LBIE addresses permanent modification to Unit 1, temporary use of the RCSVR equipment, and the acceptability of using the RCSVR during Mode 5 in Unit 1.

1. Permanent modifications to the Plant:

   A new 1 inch RCSVR connection with dual isolation valves is added on the 6 inch PORV inlet header with provisions to connect a 1-1/2 inch vacuum hose. The dual isolation valves provide part of the RCPB during Modes 1 through 4. In Mode 5, these valves provide the connection to the RCSVR components.

2. For temporary use of the RCSVR components:

   Temporary hoses and a skid-mounted vacuum pump will be used to connect the RCSVR components to the RCS. The new valves added will be used with a 1-1/2 inch vacuum hose to connect the vacuum pump to the Pressurizer volume and a 2 inch vacuum hose will connect the remaining RCS via an existing 2 inch connection on the RCS Loop 1 Cold Leg, which includes two (2) existing RCPB valves.

3. Acceptability of using RCSVR during Mode 5:

   The RCSVR process will be used during restart after a refueling outage or other plant shutdown periods when the RCS is drained to mid-loop, allowing the SG tubes to be filled with air. The RCSVR process will be initiated at mid-loop during the refill process and continue until the RCS is filled.

Summary of Evaluation:
DCP N-049532 allows the modification of the Unit 1 RCPB to add a 1 inch connection and use an existing 2 inch RCPB connection to use a RCSVR process during Mode 5 in Unit 1.

The use of the RCSVR process does not impact the operating license, licensing basis, or the design basis of the unit. A new, alternate method of refilling the RCS during an outage is provided. This method uses a vacuum to evacuate the air/gasses from the RCS during the fill process. It can result in the earlier establishment of natural circulation cooling capability for the RCS during plant restart and earlier formation of a steam bubble in the Pressurizer beginning with the Pressurizer at a vacuum, avoiding water-solid operation of the RCS. This process is added as an alternate method in the FSARU description of plant restart.

The revised ISI boundary drawing and the piping schematic sheets are part of the FSARU figures. The design installation and operation of the RCPB connections are in accordance with the applicable design classifications and meet the description of the system in the SAR.

In summary, the use of the RCSVR will not impair the safety function or performance of the reactor vessel, reactor internals, CRDM system, level monitoring systems, RCP seals, pressurizer, SGs, tanks, pumps, heat exchangers, filters, demineralizers, valves, or RCS piping. Based on the above information, the use of the RCSVR will not adversely affect the safe operation of the plant. The operation of the RCSVR does not represent a change requiring prior NRC approval nor does it require a change to plant TSs.

01-041 Change Containment Closure Commitment at Midloop

Reference Document No.: T32858
Rev. No: 0
Reference Document Title: Containment Boundary – Release Pathways
Activity Description:
This modification to the commitment will allow the equipment hatch, personnel airlock, emergency airlock, and containment penetrations that open directly to the outside atmosphere to be open during midloop operations as long as they are capable of being isolated within the calculated time-to-boil of the RCS after initiation of a loss of RHR event.

PG&E's current commitment to establish containment closure whenever RCS inventory is less 111 feet is modified to require the capability of containment closure within the calculated time-to-boil whenever the RCS inventory is less than 111 feet.

Allowing this flexibility will increase the efficiency of outage-related activities, reduce the potential dose to the personnel and, eliminate personnel safety issues.

Summary of Evaluation:

The initial commitment was in direct response to GL 88-17 and provided expeditious actions to ensure that a loss of RHR capability would not result in release of radioactive material outside containment. In GL 88-17, the NRC provided guidance that containment closure was required within 30 minutes of the initiation of a loss of RHR at midloop operation. PG&E's response went beyond that requirement and mandated that the containment be closed when the RCS inventory was less than 111 feet. This is considered an over commitment.

In GL 88-17, the NRC allows for program enhancements based on better understanding or improved procedures. These are allowed if the enhancements provide a significant safety improvement or enhancement of plant operations with no decrease in safety. The NRC specifically provides that program enhancements could lessen the initial impact of the expeditious actions such as the speed with which containment closure must be achieved.

The commitment changes allow for the containment penetrations, including the equipment hatch, personnel air lock, emergency air lock, and penetrations with direct access from the atmosphere inside containment to the outside containment atmosphere, to be open at midloop under administrative controls as long as there is reasonable assurance that the containment can be closed within the calculated time-to-boil from a loss of RHR. In addition, if the time-to-boil is calculated to be less than 20 minutes, then the containment will not be allowed open during midloop operation. These proposed commitment changes and the resulting administrative controls meet all of the requirements of GL 88-17 and provide reasonable assurance that the containment will be closed within the time to boil at midloop operation. This change does not require prior NRC approval.

01-042 Revision to Commitment T31460

Reference Document No.: Commitment T31460
Rev. No: 0
Reference Document Title: Commitment T31460
Activity Description:
Change the Statement of Commitment in the PCD as follows:

Calculation STA-133 provides criteria for allowable seepage into the RCS from the charging injection header flow path such that there will be no thermal stratification in the injection lines and the cyclic fatigue discussed in IE Bulletin 88-08 would be precluded. Surveillance Test Procedure STP I-1D will monitor charging injection header pressure at PI-947 and PI-155 monthly and contain guidance to initiate venting of the charging injection header if the rate of seepage contained in STA-133 is exceeded.

This change allows Engineering Services to revise applicable STPs to:

1) Allow high-head safety injection (HHSI) header pressure at PI-947 between isolation valves 8801A/B and 8803A/B, and PI-155 in the header bypass line to exceed the RCS pressure if the header pressurization rate is

$\leq$ 20 psi / hr.

2) Extend the surveillance test interval from weekly to monthly.

In addition to changes to the Unit 1 STP I-1 series of STPs, a requirement to check the rate of pressure increase in the HHSI header following stroking of either Unit 1 SI-8803 or SI-8803B during operating modes will be added to other applicable STPs.

Summary of Evaluation:
NRC IE Bulletin 88-08, "Thermal Stresses In Piping Connected To Reactor Coolant Systems," identified cracking in unisolable sections of ECCS piping connected to the RCS loops at several plants. The cause was determined to be high-cycle thermal fatigue caused by relatively cold water leaking through a closed valve at a pressure sufficient to inject into the RCS and a leak rate sufficient to create temperature stratification in affected piping. The NRC Bulletin recommends three methods that will provide continuing assurance that unisolable sections of piping connected to the RCS will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of a unit. As stated in the Bulletin, "This assurance may be provided by (1) redesigning and modifying these sections of piping to withstand combined stresses caused by various loads including temporal and special distributions of temperature resulting from leakage across valve seats, (2) instrumenting this piping to detect adverse temperature distributions, or (3) providing a means for ensuring that pressure upstream from block valves which might leak is monitored and does not exceed RCS pressure."

In response to IE Bulletin 88-08, surveillance procedure STP I-1C, "Routine Weekly Checks," was modified to verify that the HHSI system header pressure is less than the RCS pressure to eliminate the possibility of undetected leakage into the RCS.

However, a number of instances of seepage past the valve discs of Unit 1 valves SI-8803 A/B have resulted in the need for frequent venting of the charging injection header in Unit 1 over the past several operating cycles to maintain HHSI header pressure less than RCS pressure. Venting has become a burden for plant operators. It has also resulted in increased radiation doses to operators.

Calculation STA-133 determined that a small amount of RCS in-leakage (i.e., < 1.6 cc/min see page) past the SI-8803 valves would not result in the cyclic thermal fatigue phenomenon detailed in NRC Bulletin 88-08. The calculated allowable in-leakage rate ensures sufficient time (approximately 1.5 hours) for heat transfer to take place, gradually heating the leakage fluid as it approaches the RCS loop. Thus, this in-leakage will not result in any measurable radial temperature differences (i.e., temperature will continue to be homogeneous radially in the pipe). Without any substantial radial temperature differences, there will be no stratification-induced thermal stresses to the injection lines. This in-leakage rate corresponds to a pressurization rate of $\leq$ 20 psi/hr in the HHSI header, once HHSI pressure reaches or exceeds RCS pressure.

Regarding the time interval change between the surveillance tests from weekly to monthly, the additional guidelines provided in the NRC's acceptance letter (see 6.1.C above) recommended a 6-month interval if the option to allow in-leakage is selected. Thermal cycling is a long-term phenomenon that would need to exist for an extended period of time (e.g., years) before piping integrity would be impacted. Therefore, the 6-month interval recommended by the NRC is conservative, because thermal fatigue resulting from the stratification and thermal cycling is an accumulative effect over a long period of time on the pipe. However, instead of the 6-month interval, it is recommended to extend the surveillance interval from the current weekly to monthly; this frequency is still more conservative than the NRC-recommended guideline.

The proposed changes do not require prior NRC approval.

01-043 Scaffold Materials Storage Rack Installation

Reference Document No.: TP TA-0101

Rev. No: 0

Reference Document Title: Scaffold Materials Storage Rack Installation.

Activity Description:

The containment scaffold storage rack project, to be implemented in accordance with DCN DC1-SC-049556, requires that numerous embed sleeves be installed into the containment floor at the 91 foot elevation. Embed sleeves are essentially small sections of pipe into which the scaffold storage racks will be mounted. Installation involves chipping out small portions of concrete on the containment floor and grouting the sleeves into the chipped-out locations. It is desired to install the embed sleeves during Mode 1. The estimation duration of the embed installation is 120 hours (3 weeks x 40 hours/week).

The anticipated duration and the elective nature of this maintenance is what makes this activity differ from present practices. Typically, containment entries are made for short periods of time and involve inspection activities or restorative maintenance.

Performing the embed sleeve installation during Mode 1 will be beneficial for two reasons: (1) ALARA - the radiation dose at the containment 91 foot elevation is lower in Mode 1 than it is in outage modes, and (2) Outage duration - various scheduling and work conflicts will arise if the embed sleeve installation is implemented during a refueling outage since the containment 91 foot elevation floor space is one of the main laydown areas during an outage. This could lead to outage delays.

Summary of Evaluation:
The principle impacts to the licensing basis are related to the RHR sump screen's ability to support the ECCS in mitigating a LOCA and its consequences (FSARU Chapter 15). Additionally, the FSARU contains a specific evaluation on insulation and other debris affecting the RHR sump availability after a LOCA (FSARU Chapter 6). The key argument in this LBIE, with respect to the continued operability of the RHR sump screen, is that the increase in risk associated with impairing the RHR sump is minimal, considering the duration of the proposed work at power and the probability of a LOCA occurring during that time. Also, based on a review of numerous documents issued by the NRC related to debris in containment blocking emergency core cooling system sump screens or suppression pool strainers, transient material and individuals in containment that are not left unattended are not required to be considered debris that could block the sump screens.

The proposed work activity does not impair the ability of important-to-safety SSCs to perform their safety functions and does not change the way the plant is operated. The installation of embed sleeves into the 91 foot containment floor while at power does not require prior NRC approval.

01-044 Evaluation of Deferral of Reactor Trip Bypass Breaker PM's

Reference Document No.: A0044658-1/A0047030/31-1

Rev. No: N/A

Reference Document Title: Evaluate U1 52BYA PM for Postponement to 1R11.

Evaluate U2 52BA and BYB PM for Postponement to 2Rr11.

Activity Description:
The preventive maintenance (PM) on the reactor trip bypass breakers was originally scheduled during an outage. In the effort to reduce outage scope, the PMs were moved to the daily schedule and several on-line PMs were performed.

However, a review of the PMT requirements by the system engineer during this operating cycle determined that a trip risk existed while performing PMT on the bypass breakers. It was recommended that the bypass breaker PMs be returned to implementation during an outage. Deferring the PM until the next outage will result in exceeding the manufacturer's recommendation by approximately ten months. PG&E committed to performing maintenance on the reactor trip breakers in accordance with the manufacturer's recommended frequency in letter DCL 88-132. This commitment is listed in the PCD as T31087.

Summary of Evaluation:
Engineering evaluated the consequences of deferring the PMs on the reactor trip bypass breakers until the next refueling outages (ten months was rounded up to one year). This evaluation resulted in the determination that there is not more than a minimal increase in the frequency or consequences of an accident or malfunction due to a deferral of the PMs for one year.

The postponement of the PM on the bypass breakers should not adversely affect the ability of the breaker to perform its safety-related function, which is to trip on an engineered safety features actuation system signal. Therefore, this change does not require prior NRC approval.

01-045 Unit 2 EDUPS2 Battery Reconfiguration / Appendix R Compensatory Measures

Reference Document No.: Jumper 01-22
Rev. No: 0
Reference Document Title: Unit 2 EDUPS2 Battery Reconfiguration
Activity Description:
The Appendix R pipe rack light fixtures are supplied power from pipe rack lighting uninterruptible power supply (UPS), EJUPS2, which is supported by an 8 hour battery bank, EDUPS2. Following a successful 8 hour discharge test in accordance with STP M-17B6, the EJUPS2 input breaker tripped. During troubleshooting of the UPS, it was found that 10 of the 30 installed batteries had failed and were preventing proper recharging of the remaining 20 batteries. Jumper 01-22 reconfigures EDUPS2 battery bank from 3 strings of 10 batteries to 2 strings of 10 batteries, removing the 10 failed batteries from the circuit. This LBIE evaluates the compensatory measures provided by the jumpered configuration of EDUPS2, and placement of portable lanterns, supporting Appendix R emergency lighting for EOP operations in the pipe rack area for the 8 hours as described in FSARU 9.5.3. This non-conforming condition will be corrected in accordance with GL 91-18 as soon as replacement batteries are available.

Summary of Evaluation:
This activity does not require prior NRC approval because:

- FSARU Section 9.5.3 description of 8 hour operation of Appendix R light fixtures is not based on a specific analyzed accident but is a general requirement. The compensatory measures provided by the jumpering of EDUPS2 batteries and EJUPS2 along with in-place portable lanterns will provide emergency lighting to support EOP operations in the pipe rack area for the required 8 hours.

- T35110 commitment to repair, or replace, and retest Appendix R lights made in 1982 was based on battery-operated lights and was extended to include the EJUPS2 and EDUPS2 equipment when it was installed in 1994. Spare batteries for EDUPS2 cannot be stored in the warehouse and must be obtained fresh from the supplier. All possible actions to identify the failure and to repair the equipment within 7 days have been made. As a temporary deviation, PG&E is not seeking a change T35110.

- There are no TSs or ECGs impacted by this activity.

The compensatory measures in place will still provide emergency lighting for the EOP actions in the pipe rack area and do not require prior NRC approval.