Pacific Gas and Electric Company®

James R. Becker
Site Vice President

Diablo Canyon Power Plant
Mail Code 104/5/601
P. O. Box 56
Avila Beach, CA 93424

805.545.3462
Internal: 691.3462
Fax: 805.545.6445

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PG&E Letter DCL-10-039

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, 2008, through December 31, 2009

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.59, "Changes, tests, and experiments," Pacific Gas and Electric Company (PG&E) is enclosing the 10 CFR 50.59 Report for Diablo Canyon Power Plant (DCPP), Units 1 and 2, for the period January 1, 2008, through December 31, 2009. In accordance with 10 CFR 50.59(d)(2), the report provides a summary of all 10 CFR 50.59 evaluations performed during this period.

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

The Plant Staff Review Committee has reviewed the referenced LBIEs and has concurred with the determination regarding NRC approval as specified in the summaries.

Some LBIEs in the enclosure involve the replacement reactor vessel heads. PG&E has subsequently submitted License Amendment Request 09-06 "Critical Damping Value for Structural Dynamic Qualification of the Control Rod Drive Mechanism Pressure Housings," dated December 14, 2009, to support replacement of the reactor heads.

Pacific Gas and Electric Company makes no regulatory commitments (as defined by NEI 99-04) in this letter.

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NRR

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

Sincerely,

James R. Becker
Site Vice President

grg/50045548
Enclosure
cc/enc: Elmo E. Collins, NRC Region IV
Michael S. Peck, NRC Senior Resident Inspector
Alan B. Wang, Project Manager NRR
Diablo Distribution

Enclosure
PG&E Letter DCL-10-039

10 CFR 50.59 REPORT OF CHANGES, TESTS, AND EXPERIMENTS for the Period January 1, 2008, through December 31, 2009

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

Index of LBIEs

08-001	**Auxiliary Building Control Board Replacement (Phase 2) (Unit Common)**

Reference Document No.:	DCP J-49855, Rev. 0
Reference Document Title:	Auxiliary Building Control Board Replacement (Phase 2)

Activity Description:

The overall mission is to replace the obsolete Auxiliary Building Control Board control system with a new control system. The Auxiliary Building Control Board Replacement (Phase 2) design change will replace existing controls and integrates the functions of the Gaseous Radwaste (GRW), MEDT, Aux Building Sump and BA Concentrates Holding Tank 0-2 into a hybrid digital control system. The overall scope of Phase 2 Auxiliary control Board Replacement design change will establish the following:

- Data Acquisition and control System (DACS-Controllogix)
- Two (2) redundant controller/communication chassis will be installed at 73' panel PAXBIO.
- Three (3) remote I/O chassis (each having redundant power supplies). Two (2) remote I/O will be installed at 73' panel PAXBIO. One (1) remote I/O will be installed at panel POWE.
- One (1) Auxiliary Maintenance Terminal at 73' panel PAXBIO.
- Network Connections to DACS
  - Ethernet/fiber for local area network
  - ControlNet for DACS
  - 24 VDC power distribution between panels PAXBNET, PAXBIO, PAXBPNIO andPOWE.
- System upgrade to Controls/Indication
  - GRW-Fluidic Logic replacement/Gas Decay Tanks
  - GRW-Waste Gas Compressor
  - LRW-MEDT/Aux Building Sump (ABS) level controls
  - CVCS-Boric Acid Concentrates Holding tank (BACHT 0-2)
  - POWE annunciation
  - O2 analyzer 2 percent alarm and 4 percent trip
  - Selected Annunciation at other panels

Phase 2 design change scope will be completed as two separate design change packages (DCP J-49855 & J-49877). The scoping for this DCP J-49855 replaces and upgrades the Auxiliary Control Board system with a new digital control system. DCP J-49877 removes abandoned-in-place (AIP), unused components, and power receptacles and relocates waste gas compressor relays to support this digital upgrade DCP J-49855.

Summary of Evaluation:

The new system will be designed with redundant networking systems and independent power sources. The redundant power supplies are derived from independent (Unit 1&2) power sources to protect the system in the event of a loss power. Having the infrastructure designed to have redundant network systems, HMI control stations and power supplies improves reliability and minimizes down time. Also integrating some of the old instruments into the new hybrid control system will minimize maintenance.

The automatic and manual functions will be integrated into the hybrid digital control system. The new digital control system will be designed to enhance the ability for operations to control and monitor the process. The existing Aux Building Control Board annunciator alarm functions will be integrated into the hybrid digital display as alarms. Hand switches used for automatic/manual control of waste gas compressors, pumps and valves will be integrated into the control system. The new control system consolidates needed information onto display screens (HMIs) that provides much more effective view of system operation. The new system will provide a MIMIC as part of the HMI display to replace the old board MIMIC. The new system will save time for an operator to seek meter readings or other indications. The new control features will not adversely increase time required to perform the control actions from the HMI display. Cautions and warnings can be displayed to prevent potential errors. Although there is a fundamental change to the way information is presented and controls are interacted with the operator, the control capability requirements are not impacted.

Also GDT fill, vent and purge activity functions are maintained and integrated into the new control system. Failure mode for the GDT control valves to fail closed upon loss of power or air is maintained. The ability to control tank quantities including the GDT cross-over valve (FCV-417) is maintained. The key operated switch for FCV-417 was replaced with a cut out switch to maintain "defense in depth" measures. The ability for the flow control vent waste gas valve (FCV-410) providing the means for redundant isolation and manual operation to initiate valve opening will not be affected. The key operated switch for FCV-410 is removed since it requires manual operation to open the valve and establish venting. Also FCV-410 has a common "redundant" relationship to both the upstream dedicated GDT vent valves (FCV-404, 405 & 406) and the downstream vent header radiation monitor control valve RCV-17. This requires independent action to open the GDT vent valves to avoid inadvertent venting. Also added features such as administrative controls within the digital controls will require a minimum of a two step function to initiate venting. The RM-22 radiation monitor, signals to EARS, control circuit for RCV-17 and valve RCV-17 will be unaffected and maintained independent of the new control system except for the existing HI

radiation alarm signal fed to the new Aux Building Control Board. RCV-17 downstream of FCV-410 will close automatically in the event of high radiation. A new feature of the new digital system processes a HI radiation alarm and provides interlock to automatically close GDT vent valves FCV-404, 405, 406, and common vent header valve FCV-410.

A Functional Requirements Specification (FRS), 663195-31 has been written to specify specific programming and design functional requirements for the digital controller and HMIs. The FRS provides the basis for the plant software QA program and configuration management of the reactor makeup control system.

In general, the new system improves the control and provides human factor enhancements to better support operator tasks and reduce risk of errors.

This design change does not require prior NRC approval. There is no adverse impact to any radwaste design function described in the FSARU. All electrical and instrument control schemes being replaced or modified provide a non-safety related (Design Class II) function. All control schemes are supplied by non-vital power. The associated systems are non safety related and not relied upon for the safe shutdown of the plant. The design change replaces the existing obsolete Gaseous Radwaste Fluidic Logic control system and other components to increase system reliability and availability.

08-002 Unit 1 RI Rack Replacement and LTB Elimination

Reference Document No.:	DCP J-049907, Rev. 0
Reference Document Title:	U1 RI Rack Replacement

Activity Description:

Existing System:

The instrument rack (RI) in the cable spreading room includes Fisher power supplies, controllers, and manual/auto stations and Moore current switches. With the exception of a Class IB current isolator that interfaces with the EARS system, all components in the RI Rack are Class II. The functions of RI Rack components include providing field instrument loop power; providing alarm inputs to the main annunciator system; providing process controllers for chemical injection, feed water pump recirculation valves FCV-53 and 54, condensate recirculation valve FCV-30, stator coil and cooling water flow valve FCV-31, and hot well level control valves LCV-8 and 12; and providing analog outputs to control room indicators and/or the Plant Process Computer (PPC).
The Fisher AC2 equipment in the RI Rack has been obsolete for many years

and will become increasingly difficult to maintain with the passage of time. In addition, existing Westinghouse Hagan manual/auto stations on VB-3 that interface with the RI Rack are obsolete.

DCPP has a load transient bypass (LTB) system. The purpose of the system is to provide a means to prevent loss of suction to the MFWPs upon a full load rejection. Following the Steam Generator Replacement project, the DCPP design basis will no longer include a full load rejection. The maximum design basis load reduction will be 50 percent. Per Westinghouse analyses, the licensing bases for DCPP will be met with or without LTB. Keeping LTB exposes DCPP to the negative effects of spurious LTB actuation (which has occurred in the past) and requires continuing maintenance of the associated equipment.

New System:

To resolve obsolescence issues with the existing Fisher AC2 equipment, all instrumentation in the RI Rack will be removed and replaced with Triconex Tricon triple modular redundant (TMR) chassis, a fiber switch, a cabinet maintenance HMI, and new, redundant power supplies. The obsolete Hagan manual/auto stations on VB3 will be replaced with new manual/auto stations with the same fit and function to be provided by NUS.

The new RI Rack equipment and control room manual/auto stations will provide the same functionality as the existing devices with the addition of control functions to improve MFWP suction pressure during load transients (see above). Additionally a data link will be provided to the Plant Data Network (PDN) via a server that has been installed in the adjacent EARS cabinet under EDT 31194. Existing analog inputs to the PPC will be spared and the same data will be sent via the existing PDN to PPC data link.

To ensure that no Class I devices are in the Class II instrument rack, the Class I isolator in the RI Rack will be moved to Class I panel RCRM.

Load transient bypass will be eliminated by this project. Simulator runs (see DCP Attachment 11) have demonstrated that there will be adequate NPSH for the feed pumps under all load rejection and load reduction scenarios provided additional logic is implemented as follows. This DCP adds control logic and I/O to close LCV-12, open FCV-31, and start the standby condensate/condensate booster pump set upon a load drop anticipate or a loss of MFWP load ramp.

The PLC in the adjacent EARS cabinet RNM1 per EDT 31194 provides data acquisition for Radiation Monitoring System (RMS) inputs to the EARS system. In this project, the PLC and its associated maintenance terminal and power supply will be removed, the PLC's field termination panels (FTPs) in

cabinet RMN1 will be reconnected to the new Triconex chassis in the RI Rack, the Triconex will be connected to the PDN, and the EARS data acquisition functionality for the affected inputs will be incorporated in the Triconex equipment. (Note: Unit 2 RMS inputs will be connected to the Unit 1 RI rack Triconex chassis per this DCP. These will be reconnected to the Unit 2 RI rack in DCP J-50907.)

To provide power to the RI Rack during bus transfers following unit trips, the RI Rack will be connected to the Digital Feedwater Control System UPS.

Summary of Evaluation:

There is no adverse impact to the design functions of the RI rack. There is no adverse impact due to the deletion of load transient bypass logic. A thorough FMEA was performed in the DCP summary and evaluation to support this conclusion.

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

08-003 Unit 2 Reduced Tavg with RSGs for Cycle 15

Reference Document No.:	DCP M-050790, Rev. 1
Reference Document Title:	RSG Component Modification

Activity Description:

DCP M-050790 Rev. 0 is being revised to Revision 1 to address operation of Unit 2 for a range of Tavg from 577.6°F to 565.0°F (Tavg Range) for one cycle, Cycle 15, with the replacement steam generators (RSGs). Revision 0 addressed operation with RSGs at the design Tavg of 577.6°F, since the full set of analyses and evaluations to support lower Tavg values was not available. Unit 2 with the RSGs is expected to be operated with Tavg in the range of 568-569°F instead of the current value of approximately 572°F, a reduction in Tavg of up to 3°F, to limit steam outlet pressure to approximately 805 psia, the current design pressure. This reduction in Tavg is a result of: (1) a higher overall heat transfer coefficient for the RSGs compared to the OSGs, and (2) limitations of the turbine and other secondary side equipment. This LBIE addresses operation of Unit 2 for Cycle 15 with the RSGs and a reduction in Tavg operation from 577.6°F down to 565.0°F (reduced Tavg operation). Analyses and systems, structures, and components which LBIE Screen identified as may be adversely impacted or are adversely impacted are addressed in this LBIE.

This LBIE only considers the impact associated with the RSGs and the

reduced Tavg operation. This LBIE Screen credits the discussions and conclusions of LAR 07-01 (DCL-07-002 dated *1/11/07;* DCL- 07-075 dated *8/9/07;* DCL-07-089 dated *9/28/07),* associated NRC Safety Evaluation (dated January 8, 2008) and LBIE documentation for DCP M-050790 Rev. 0 (LBIE 2007-013 approved *10/12/07)* for all aspects of Unit 2 operations with the RSGs at a design Tavg of 577.6°F. This LBIE does not evaluate any aspects of reduced Tavg operation on the cycle specific fuel (core) design. The effects of Tavg on the fuel design, including the fuel mechanical design changes, is evaluated separately as part of the Unit 2 Cycle 15 reload evaluation and associated 10 CFR 50.59.

Based on the specified plant operational parameters for the cycle (e.g., Tavg, power, pressure, etc.), the reload evaluation addresses the impact of the cycle specific fuel design on the design bases and postulated licensing basis accidents. The reload evaluation includes assessment of any accident analyses which are impacted or require change as a result of the cycle fuel design. The FSAR sections 5.2.1.5 and 15.1.2.2 are updated to state Unit 2 is expected to operate at an RCS vessel Tavg of approximately 568°F following SG replacement and that a design change has been implemented that provides analyses and evaluations to support operation over a Tavg range of 565.0 of to 577.6 of for Cycle 15. FSAR Table 5.1-1 is updated to state Unit 2 is expected to operate at an RCS vessel Tavg of approximately 568 of following SG replacement. FSAR Table 5.2-4 is updated to state for Unit 2 the Tavg/power coast down design transient conditions are enveloped by analysis and evaluations contained in a design change to support operation over a Tavg range of 565.0 of to 577.6 of for Cycle 15. These changes will remain for the interval between Unit 2 and Unit 1 Steam Generator replacements, and then the FSAR will be updated to reflect the overall Tavg analysis program being performed under DCP M-049947. There are no changes to ECGs or the Q-List.

Summary of Evaluation:

Plant operation with the DCPP Unit 2 RSGs and a Tavg as low as 565.0°F does not affect any existing accident initiators and the frequency of occurrence of an accident previously evaluated in the FSAR is not increased. Plant operation with the RSGs and a Tavg as low as 565.0°F does not create any additional malfunctions of SSCs important to safety and the SSCs will remain capable of performing their safety function based on individual evaluations, which concluded the current analyses are bounding, the impact is negligible, or sufficient margin exists to acceptance limits. Therefore, there is no increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR.

The dose consequences are not impacted by this change, since the

consequence analyses bound operation with Tavg at 565.0°F or are unaffected by a change in Tavg. No change to the RSG consequence analyses is required to support operating the plant with the RSGs and a Tavg as low as 565.0°F. The RSG SGTR margin to overfill analysis has been evaluated using existing margin. Since a net margin to overfill of 104 ft^3 exists, SG overfill will not occur and there is no increase in consequences for the SGTR margin to overfill analysis.

Operating the plant with the RSGs and a Tavg as low as 565.0°F does not cause a worse malfunction of an SSC important to safety, does not create any new accident scenarios, and does not introduce new single failures or failure modes. The SSCs important to safety will continue to perform their safety function as assumed in the safety analyses in the same manner as before. Therefore, the change does not create the possibility for an accident of a different type than any previously evaluated in the FSAR and does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR. Plant operation with the RSGs and a Tavg as low as 565.0°F does not impact the fuel cladding, reactor coolant pressure boundary, or containment. For the Small Break LOCA analysis for Diablo Canyon completed using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP, using a temperature range of 565.0°F-577.6°F for Unit 2 for the Tavg will have negligible effect on the SBLOCA transient.

Plant operation with the RSGs and a Tavg as low as 565°F does not result in any change to the methods of evaluation as described in the FSAR.

08-009 Diablo Canyon 500kV Special Protection Scheme

Reference Document No.:	AR A0570522 & A0570525
Reference Document Title:	

Activity Description:

The Transmission Service Provider (TSP) is adding a "Special Protection Scheme (SPS)" at the DCPP 500kV switchyard. An SPS is designed to detect abnormal bulk electric system (i.e., grid) conditions and take pre-planned, corrective action (other than isolation of faulted grid elements). The use of an SPS is an acceptable practice per NERC/WECC planning stability standards. This TSP change effects both Unit 1 and Unit 2. This review documents the review of the TSP project effects on Diablo Canyon Power Plants Unit 1 and Unit 2.

The TSP has determined that for certain grid events involving the DCPP 500kV outlet lines, both DCPP units would trip due to actuation of existing

generator out-of-step protection and/or Reactor Coolant Pump (RCP) undervoltage protection. The trips would be the direct result of intensive synchronous swings across the remaining 500kV line. The loss of multiple 500kV lines ***and*** two large generating units would result in a violation of NERC/WECC stability performance criteria. To preclude this, the SPS will trip the output breakers of one unit to reduce the severity of the synchronous swing on the remaining unit and the overall impact on the bulk electric system at the TSP boundaries.

The specific 500kV events of concern that could lead to the postulated intensive synchronous swings or to a loss of synchronism between the plant generators and the WECC system, are:

A. Double line outages (DLO) in a normal three line scheme,
B. Single line outages (SLO) in two line scheme, and;
C. Delayed SLO.

The bases for the SPS addition are:

1. A dual DCPP unit outage (DUO), in conjunction with the above initiating events, can bring stability of the WECC system close to critical conditions causing a severe test for the variety of control and protective devices in the entire system The prevention of DUO would minimize the possibility of cascading, e.g., collapse and separation of California-Oregon Intertie (COI). Cascading is not permitted by the NERC/WECC Planning Standard even for relatively rare disturbances combined with a failure or a partial operation of control and protective devices.
2. Grid transient voltage dips during the intensive synchronous swings or in out-of-step conditions (before out-of-step protection trips the units) may exceed values allowed by the NERCNVECC and PG&E Planning Standards.
3. A dual DCPP unit outage (DUO) would impose a definite strain on the remaining generating resources in the system to supply demand, especially if other resources are off-line.
4. Out-of-step conditions would impose severe stress on the equipment of both units. REFERENCE DOCUMENT NO.: None (Ref. A0570522 & A0570525)

The subject activity is a grid change at the point of DCPP connection. It is not a DCPP design change.

Summary of Evaluation:

The 50.59 evaluation concludes that the proposed change of adding a 500kV Special Protection Scheme (SPS) by the DCPP Transmission Service Provider (TSP) to the DCPP 500kV system does not require prior NRC approval. This TSP SPS does not affect the normal operation of the PG&E

transmission system or the normal operation of the plant to the extent that it would result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSARU which is the loss of external electrical load. Other criteria in the 50.59 evaluation are determined to be either not affected or not applicable (change in evaluation methodology).

The relays comprising the SPS function are entirely owned, operated, and maintained by the TSP. The SPS addition enhances grid stability for specific WECC Category C & D events (i.e., multi-contingency) local to DCPP. The only SPS interfaces with DCPP systems, structures, and components are:

System: 500kV Transmission System (FSARU Chapter 8)
Structures: None
Components: 500kV Circuit Breakers 532 & 632 Unit 1,542 & 642 Unit 2 (FSARU Chapter 8.2.1.2)

It has been determined that an inadvertent SPS actuation has a less than minimal probability of increasing the frequency of occurrence of the FSAR previously analyzed "Loss of External Electrical Load and/or Turbine Trip" Condition II Fault as described in FSAR Section 15.2.7. Condition II Faults are considered to be of moderate frequency (i.e., once per year). Condition II Faults (i.e., events) do not propagate to cause a more serious fault, i.e., a Condition III or IV fault. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system (RCS) overpressurization.

08-010 Exceedance of ECG 40.1 Completion Times

Reference Document No.:	ECG 40.1, Rev. 4
Reference Document Title:	Meteorological Instrument

Activity Description:

This change evaluates exceeding the Equipment Control Guideline (ECG) 40.1, "Meteorological Instrumentation," Required Action A.1 and Required Action C.2 Completion Times of 30 days for 73 days (a total inoperable time of 103 days) due to inoperable primary meteorological (met) tower and inoperable communication links between the primary met tower and the primary data network (PDN), safety parameter display system (SPDS), and the emergency assessment and response system (EARS). ECG 40.1 Action Note 1 states that prior to exceeding the Completion Time of any Required Action, a 10 CFR 50.59 evaluation must be approved by the PSRC justifying the acceptability of exceeding the Completion Time. When the ECG 40.1 Required Action A.1 and Required Action C.2 Completion Times are not met, ECG 0.3 is entered, which requires initiation of a quality problem AR, as appropriate, and obtaining station director approval per OP1.DC16.

On 5/21/08 the primary met tower was damaged by impact of a mobile crane with a guy wire and AR A0730749 was initiated. On 5/21/08 TS sheet T0063121 was created for entry into ECG 40.1, Action A.1 and C.2, the backup met tower was confirmed to be operable, and the portable battery-powered met station was deployed per procedure AWP EP-005. All ECG 40.1 required met tower functions and communications links are being provided by the back-up met tower and associated communication links as documented in AR A0731550, AE 4. The ECG 40.1 Required Action A.1 and Required Action C.2 Completion Times of 30 days will expire on 6/20/08. On 5/22/08 a quality problem was initiated and on 5/28/08 NCR N0002224, "Mobile Crane Operation Damaged the Primary Met Tower," was initiated. The ECG 40.1 Required Action A.1 and Required Action C.2 Completion Times of 30 days are expected to be exceeded for 73 days to support replacement of the primary meteorological tower. A total inoperable time of 103 days is expected to be required to procure, construct, install, calibrate, and test the new primary meteorological tower and communication links to the PDN, SPDS, and EARS. The next scheduled due date for the back-up met tower channel calibration is 7/16/08 and the ECG due date is 9/10/08 (R0311929, AR A0731550 AE 2). The back-up met tower channel calibration will be managed, including a possible one-time exceedance of the ECG 40.1 SR 40.1.2 frequency, in order to prevent or limit the loss of the back-up met tower due to the channel calibration.

Summary of Evaluation:

The change is to exceed the ECG 40.1 Required Action A.1 and Required Action C.2 30 day Completion Times for 73 days due to inoperable primary met tower and inoperable communication links between the primary met tower and the PDN, SPDS, and the EARS (a total inoperable time of 103 days).

The primary met tower and communication links between the primary met tower and the PDN, the SPDS, and the EARS are instrumentation used to respond to an accident and are not accident initiators. There is not an impact on the probabilistic risk assessment (PRA) core damage frequency (CDF) figure of merit. Therefore the change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSARU.

The change does not increase the probability of malfunction of the back-up met tower. Failures resulting in functions with no operable channel are not a new situation created by the change and this situation is addressed by ECG 40.1 Condition B. Based on the ECG 40.1 Required Action B.2 limitation of 7 days for a function with no channel operable, the required deployment of the portable battery-powered meteorological system during the periods no channel is operable, and the availability of the supplemental

meteorological measurement system, the change does not result in more than a minimal increase in the likelihood of occurrence of malfunction of meteorological instrumentation.

The change does not physically alter any plant structures, systems, or components, and is not an accident initiator; therefore, there is no effect on the probability of accidents previously evaluated. The back-up met tower and associated communications links, that perform in an identical manner to the primary system, are still required by ECG 40.1 and the portable battery-powered meteorological system are still available to perform the required function. The change does not affect the types or amounts of radionuclides released following an accident. It is judged there is not a significant impact on the PRA Large Early Release Frequency (LERF) figure of merit and the potential change in LERF estimate is well below the NRC's risk significance thresholds. Therefore, the change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSARU.

The change does not increase the probability of malfunction of the back-up met tower and does not affect the types or amounts of radionuclides released following an accident. The back-up met tower and associated communications links are still required by ECG 40.1 and the portable battery-powered meteorological system are still available to perform the required function. Therefore, the change does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSARU.

The change does not result in any new types of failure modes, failure mechanisms, or new accident precursors and does not involve any modification to the operational limits or physical design of the plant safety systems credited in the accident analyses. There are no new accident precursors generated due to the change. Therefore, the change does not create the possibility for an accident of a different type than any previous evaluated in the FSARU, does not create the possibility for a malfunction of an SSC important to safety with a different result than any previous evaluated in the FSARU, and does not create the possibility for an accident of a different type than any previous evaluated in the FSARU.

The change does not impact the fuel cladding, reactor coolant system pressure boundary, or containment. Therefore, the change does not result in a design basis limit for a fission product barrier as described in the FSARU being exceeded or altered.

The change does not impact the safety analyses because the safety analyses do not credit the meteorological instrumentation. The change involves a change to the period meteorological instrumentation is operable and does not

involve a change to a method of evaluation described in the FSARU. The ECG 40.1 required operability of the back-up met tower and associated communication links ensures that the meteorological instrumentation function is provided.

Therefore, the change does not result in a departure from a method of evaluation described in the FSARU used in establishing the design bases or in the safety analyses.

08-015 Mods to the Holtec HI-TRAC 125D Transfer Cask

Reference Document No.:	DCP N-49773, Rev. 0
Reference Document Title:	Dry Cask Storage System

Activity Description:

Holtec has modified its HI-TRAC 125D transfer cask for use at the Diablo Canyon Power Plant and ISFSI facility. The modifications involve shortening the cask to allow vertical handling throughout the loading or unloading and transport processes. The overall height of the HI-TRAC has been reduced by approximately 9 inches. The bottom flange and the flange gussets of the transfer cask have been modified in order to meet Diablo Canyon's site-specific requirements. The lift lugs and some the attachment points that could be used to restrain the transfer cask have been eliminated. The original DC ISFSI design used these attachment points for the bumpers during the horizontal movement of the transfer cask, which is no longer the process at DCPP. As a result, these attachment points are no longer required on the DC ISFSI transfer cask.

Summary of Evaluation:

The modifications to the HI-TRAC transfer cask do not affect criticality or confinement because the transfer cask does not perform a criticality or confinement function.

The modifications do affect shielding capability, thermal performance, and structural integrity. However, as discussed in the attached evaluation for these changes, the effects of these changes are within the design limits of the DCPP and DC ISFSI licensed HI-STORM 100 system and none of the effects are considered adverse and are acceptable under 10 CFR 50.59 and 10 CFR 72.48 criteria.

08-016	Holtec Multiple Purpose Canister (MPC) Mods
Reference Document No.:	DCP N-49773, Rev. 0
Reference Document Title:	Dry Cask Storage System

Activity Description/Summary of Evaluation:

The Holtec MPC-32 has been modified for use at the Diablo Canyon Power Plant and ISFSI facility. The modifications involve shortening the MPC for use in the shortened HI-TRAC transfer cask to allow vertical handling throughout the loading and transport processes.

Summary of Evaluation:

The changes to the MPC affect criticality, confinement, shielding capability, thermal performance, and structural integrity and required further review in this LBIE. As a result of that review and as discussed in the attached evaluation for these changes, the effects of these changes are within the design limits of the DC ISFSI licensed HI-STORM 100 system and none of the effects are considered adverse and are acceptable under both 10 CFR 50.59 and 10 CFR 72.48 criteria.

08-017	Holtec Vertical Cask Transporter
Reference Document No.:	DCP N-49773, Rev. 0
Reference Document Title:	Dry Cask Storage System

Activity Description:

This design change implements the use of the Holtec Vertical Cask Transporter (VCT) which has been designed for use at the Diablo Canyon Power Plant and ISFSI facility. The VCT is designed to transport vertical a loaded or unloaded HI-TRAC transfer cask or HI-STORM overpack on site. The VCT will provide for all lifting and transport requirements associated with movement of a loaded transfer cask from just outside the fuel handling building (FHB) to the cask transfer facility (CTF) and a loaded overpack from the CTF to the ISFSI pad. The VCT also performs the lifting requirements for the transfer of a loaded Multiple Purpose Canister (MPC) between the transfer cask and the overpack at the CTF.

The VCT is designed to be capable to transport a loaded HI-TRAC up a 10 percent grade. In Holtec Calc HI-012768 R3 and PG&E Calc OQE-014,

the VCT is evaluated for seismic stability and sliding during a design basis earthquake while carrying a loaded HI-TRAC. Based on the modeling methodology used in these calculations the change in the center of gravity from carrying the cask vertically or horizontally does not affect the results of the calculations and the results are bounding for both configurations.

The VCT was always designed to be capable of vertical and horizontal carry. In addition, the VCT has always been capable of lifting the maximum load of a loaded HI-STORM overpack; however, previously it was not the primary lifting mechanism at the CTF as there was a CTF jackscrew system that performed this action. This jackscrew system has been eliminated in the current design.

The VCT has also been fitted with a cable winch used to pull the Low Profile Transporter (LPT) with a loaded transfer cask out of the FHB. The LPT is pulled along rails that allow it to exit the FHB and moved to a position within the VCT footprint where the VCT will capture the transfer cask and carry it to the CTF. The VCT has been designed to support this function during any associated design basis accident.

The VCT has a licensing basis limit of 50 gallons of fuel on board based on an engulfing fire for the HISTORM system. The licensing basis for the DC ISFSI provides that the 50 gallons must not be capable of entering the CTF during use of the VCT. The NRC safety evaluation report indicates that this may be performed by the use of a removable fuel tank that will be moved away from the CTF during VCT use. In the licensing submittals for the DC ISFSI, PG&E provided the use of the removable fuel tank as an example of possible solutions to this issue. It was not provided as the actual solution. The actual VCT design provides for an on board 45 gallon diesel fuel tank and a catch pan that is designed to catch and divert any leakage away from the CTF through a hose connection on the pan. This design meets the requirements of the license in that it will ensure that any fuel leakage can not reach the CTF.

The VCT operates between just outside of the Unit 2 fuel handling building up to and at the CTF; and between the CTF and the ISFSI pads. The VCT does not enter any 10 CFR 50 structures, however, its route does move within the 10 CFR 50 protected areas. As a result, the 10 CFR 50 licensed facilities have been reviewed for effects of the VCT.

Summary of Evaluation:

The design of the VCT does not affect criticality, confinement, shielding capability, or thermal performance of the HI-STORM 100 system. The transporter has been designed to applicable codes and standards to ensure

its structural integrity. The change in the primary function of the VCT to perform all the lifting at the CTF required further review in this LBIE. In addition, the use of the onboard catch pan for the diesel fuel tank also needed further review.

As a result of that review, the effects of these changes are within the design limits of the licensed DC ISFSI facility and the VCT remains capable of performing all design and licensing basis functions, and all operational requirements. None of the VCT changes are considered adverse and are acceptable under 10 CFR 72.48 and 10 CFR 50.59 criteria.

08-018 Low Profile Transporter (LPT) Modifications

Reference Document No.:	DCP N-49773, Rev. 0
Reference Document Title:	Dry Cask Storage System

Activity Description/Summary of Evaluation:

This design change implements the use of a new Low Profile Transporter (LPT) to move the transfer cask into and out of the fuel handling building/auxiliary building (FHB/AB) in a vertical orientation. The LPT will only be used within the Unit 2 FHB/AB because all movement of fuel to the ISFSI will commence on the Unit 2 side of the FHB/AB. There is no LPT for Unit 1 (Refer to the FHB Crane use 50.59). The LPT replaces the licensed transport frame, which was provided to down-end the transfer cask to a horizontal configuration and to move the transfer cask to a position outside of the FBH/AB to allow the cask transporter to move the transfer cask to the CTF. The use of the LPT eliminates the need for the horizontal orientation of the transfer cask in this process and eliminates the need for impact limiters.

Summary of Evaluation:

The use of the LPT does not affect criticality and confinement because the LPT does not perform a criticality or confinement function.

The change to a vertical orientation has been demonstrated to preclude tip-over and impacts with safety significant components based on the data presented in Holtec Report HI-2053390, Rev 4. Thermal performance of the system is improved in the vertical configuration due to the continued operation of the Thermal Siphon effects within the MPC.

08-022	DCM T-3 & FSAR Change Request for FHBSS Bolts

Reference Document No.:	DCM T-3 & FSAR Change, Rev. 5D
Reference Document Title:	Structural Design of the FHB

Activity Description/Summary of Evaluation:

Increase the allowable shear stress acceptance criteria as specified in Design Criteria Memorandum (DCM) T-3 (Structural Design of the Fuel Handling Building Superstructure (FHBSS) for high strength ASTM A325 bolt bearing type connections under Hosgri Earthquake (HE) loading condition. The revised allowable shear stresses will also be reflected in the FSARU sections 3.8.2.3.2.2 (Abnormal Condition), 3.8.2.5 (Structural Acceptance Criteria), 3.8.2.6.5 (Structural Bolts), and 3.8.9 (References).

The subject acceptance criteria are being increase to support the re-evaluation of the FHBSS A325 bolted bearing connections. The re-evaluation is being triggered by the building crane up-grade associated with handling of the Used Fuel Transfer Casks under the spent fuel dry storage project. For the Design Earthquake (DE-OBE) and Double Design Earthquake (DDE-SSE) loading conditions, the re-evaluation results are within existing acceptance criteria specified in DCM T-3. Only the allowable shear stress acceptance criteria for the HE loading condition is being considered due to the abnormal loading associated with this accident condition. Allowable bolt shear stress values are being proposed in accordance with Enova Position Paper (ref. 13).

Summary of Evaluation:

10 CFR 50.59 provides a process for determining if prior NRC approval is required before making changes to the facility as described in the UFSAR. The proposed activity being addressed under this evaluation primarily revises/replaces the existing methodology described in DCPP FSARU for the FHBSS high strength A325 bearing type bolted connection. In general, licensees can make changes to elements of a methodology without first obtaining a license amendment if the results are essentially the same as, or more conservative than, previous results. Similarly, licensees can also use different methods without first obtaining a license amendment if those methods have been approved by the NRC for the intended application.

The attached 10 CFR 50.59 evaluated the proposed activity concluded it does not result in a departure from a method of evaluation described in the FSARU used to establish the design bases or in the safety analysis and therefore does not require prior NRC approval.

08-023	Perform a 10 CFR 50.59 Evaluation per ECG 24.2
Reference Document No.:	ECG 24.2, Rev. 0
Reference Document Title:	Gaseous Radwaste

Activity Description/Summary of Evaluation:

ECG 24.2 Condition A requires the gaseous radwaste system maintain oxygen concentration below 2 percent. As described in A0735759, oxygen concentration cannot be reduced to below 2 percent within the required completion time of 48 hours. As required by the ECG, a 10 CFR 50.59 evaluation is performed to justify the acceptability of exceeding the 48 hour completion time of Condition A by an additional 48 hours.

Summary of Evaluation:

The proposed activity of accepting the extension of the ECG 24.2 condition A completion time of 48 hours has been reviewed. It is has concluded that there are no positive answers to the eight questions of the LBIE Evaluation. Therefore, no prior NRC approval is required.

08-024	Auxiliary Control Board Replacement - Phase 3A
Reference Document No.:	DCP J-49856, Rev. 0
Reference Document Title:	Aux Cont Board Replace-Ph 3A

Activity Description/Summary of Evaluation:

The following "Phase 3A" Auxiliary Building Control Board Replacement applies to both Units 1 & 2. The LBIE applies only to Phase 3A design change. The Auxiliary Building Control Board for Diablo Canyon Power Plant provides an Operator Station to control the receiving, storage, treatment and discharge of liquid and gaseous Radwaste products generated by both units. It also controls the Boric Acid recovery system. Many Systems are Common to both Units 1 & 2.

The systems are constantly in use to reduce and concentrate radioactive waste for off-site disposal or onsite storage. The existing panel configurations and indications are poorly located and board modifications over the years have been installed with a minimal concern for human factors. Controls and related Indications are not always adjacent to one another, and in some cases the indications only exist remotely in the field. There are several components that were installed for systems that were never made

functional and other components no longer used.

Many of the panel instruments and controllers are air operated and no longer available. In the case of the Fluidic Logic controls, the original vendor no longer manufactures replacement parts and attempts to find a replacement vendor have been unsuccessful to this point. As a pneumatic dependent system, tubing and control elements are susceptible to leaks, reducing reliability. Also, due to the failure to properly label components and maintain drawings they are very difficult to provide maintenance or troubleshooting.

The overall mission is to replace the obsolete Auxiliary Building Control Board control system with a new digital control system. The new system will be designed with redundant networking systems and independent power sources. The new digital control system will be designed to enhance the ability for operations to control and monitor the process. The new control system will consolidate needed information onto a display that provides much more effective view of system operation. The new system will save time for an operator to seek meter readings or other indications. The new control features will not adversely increase time required to perform the control actions. Although there is a fundamental change to the way information will be presented and how controls are interacted with the operator, the control capability requirements are not impacted. Adequate training of the new control system will be provided to the operators.

The Auxiliary Control Board "Obsolescence Management" overview summary in PG&E (Digsys) website describes the conceptual design changes for proposed phases 1-8 to replace and upgrade the system. The tracking AR for all phases is AR A0581789.

1. Phase 1 - Install main digital network Infrastructure (DCP J-49810)
2. Phase 2 - Gaseous Radwaste (GRW) panel modification and digital upgrade (DCP J-49855 & J-49877)
3. Phase 3 - Liquid Radwaste (LRW) panel modification and digital upgrade (DCP J-49856 & J-49961)
4. Phase 4 - CVCS panel POCV1 panel modification and digital upgrade.
5. Phase 5 - CVCS panel POCV2 panel modification and digital upgrade.
6. Phase 6 - Boric Acid Panel (POB1) panel modification and digital upgrade.
7. Phase 7 - Boric Acid Panel (POB2) panel modification and digital upgrade.
8. Phase 8 - Remote I/O Panels (Auxiliary Bldg 64'/100'/115'/140' elevations).

Phase 1 and 2 design changes above were approved. Phases 4 through 8

are future proposed design changes in which DCP numbers will be assigned pending determination of future budgetary schedules. The Phase 3 Auxiliary Control Board design change scoping will be issued as two (2) separate design change packages:

- Phase 3A (DCP J-49856): This design change replaces and upgrades the existing Auxiliary Control Board Liquid Radwaste panel control schemes with a new digital control system. DCP J-49856 is tracked per AT-DCP AR A0683899.
- Phase 3B (DCP J-49961): The design will remove the old MIMIC console associated with panels POWE and POEC and upgrade associated board and Auxiliary Control Room furniture. Phase 3B will be implemented subsequent to the Phase 3A completion.

This design change DCP J-49856 implements Phase 3A of the Auxiliary Control Board "Obsolescence Management upgrade.

The Phase 3A design change provides digital upgrade and system integration of the Auxiliary Control Board Liquid Radwaste (LRW) panel POEC and associated remote panel instruments to improve equipment reliability and availability. The modification and integration of system controls, indication and alarms will include the following:

1. Liquid Radwaste (LRW)-System 19
    a. Chemical Drains (CD)
    b. Laundry/Hot Showers (LHS)
    c. Floor Drains (FD)
    d. Equipment Drains (ED)
    e. Processed Waste Receivers (PWR)
    f. Waste Filters (WF)
    g. Demineralizer Regenerative Receivers (DRR)
    h. Containment Structure Sumps
    i. Reactor Cavity Sumps

2. N2 and H2 (System 26) supply pressure Instrumentation

3. Chemical Volume Control system (CVCS) - System 8
    a. Boric Acid Concentrates Holding tank BACHT 0-1.
    b. LHUT Instrumentation and Tank Inlet/Outlet Valves

Components within the POEC and remote panels that become obsolete due to software functionality will be removed and/or replaced. The fluidic Logic control system is completely replaced. Phase 3 relies upon the completion of Phase 1 (Network infrastructure) and Phase 2 (Gaseous Radwaste Panel POWE and associated remote panels).

The scope of the work will include removal of various components in panel POEC including control switches, process controllers, indicators, PK-64 annunciator and the fluidic logic air controls associated with the LRW system. The design upgrades associated LRW pump controls and valve status indications. The design installs a new programmable logic control (PLC) remote I/O chassis at Auxiliary Control Board panel POEC and at the Demin Regenerant Receiver Tank panel PM-207. The design modifies and adds new electrical raceway and wiring to and from the associated remote panels.

The pneumatic I/O lines associated with POEC indications and controls will be removed between POEC and PAXBPNIO. At PAXBPNIO I/O modifiers will be installed to provide signals to and from the Remote I/O Chassis in Panel PAXBIO.

The existing local level transmitters (LT-130 to 133) on the Chemical Drain and Laundry Hot Shower tanks are obsolete and will be replaced to improve availability and reliability.

Liquid Holdup tank (LHUT) Low-Low pressure switches PS-162A/B will be replaced with a pressure range of 30" wc (vacuum) to +30" wc to provide better accuracy and consistency with the LHUT Low pressure switch range for PS-161A/B. These pressure switches provide trip/alarm for the LHUT recirculation pump and gas stripper feed pump to prevent drawing further vacuum prior to reaching the LHUT vacuum relief settings for relief valves RV-140, -141 and -142. Another LHUT vacuum relief valve PCV-140 is not in use and its respective line has been manually isolated by Operations per OVID drawing 106708-6/107708-6. PCV-140 is not in use since each LHUT is protected by a dedicated relief valve (RV-140, 141, and 142). To prevent air in-leakage to the LHUT due to inadvertent actuation or seat leakage of PCV-140, the manual isolation valve downstream of PCV-140 has been normally closed. The DCM S-8 and Piping schematics will be updated to reflect that PCV-140 is not in use and the respective manual isolation valve downstream of PCV-140 will be depicted as a normally closed DR valve (Ref: AR A0726046).

Summary of Evaluation:

This design change does not require prior NRC approval. There is no adverse impact to any radwaste design function described in the FSARU. All electrical and instrument control schemes being replaced or modified provide a non-safety related (Design Class II) function. All control schemes are supplied by non-vital power. The associated systems are non safety related and not relied upon for the safe shutdown of the plant. The design change replaces the existing obsolete Liquid Radwaste Fluidic Logic control system

and other components to increase system reliability and availability.

The new system will be designed with redundant networking systems and independent power sources. The redundant power supplies are derived from independent (Unit 1&2) power sources to protect the system in the event of a loss power. Having the infrastructure designed to have redundant network systems, HMI control stations and power supplies improves reliability and minimizes down time. Also integrating some of the old instruments into the new hybrid control system will minimize maintenance.

The automatic and manual functions of the Liquid Radwaste Fluidic Logic control system, Reactor cavity sump level, Containment Structure sump level, Liquid holdup tank instrument/valve, *N2/H2* pressure and BA Concentrates Holding tank 0-1 temperature controls/alarms will be integrated into the hybrid digital control system. The new digital control system will enhance the ability for operations to monitor and control the process. The existing Aux Building Control Board annunciator alarm functions will be integrated into the hybrid digital display as alarms. The new control system consolidates needed information onto display screens (HMIs) that provides much more effective view of system operation. The new system will provide a MIMIC as part of the HMI display to replace the old board MIMIC. The new system can save time for an operator to seek meter readings or indications. Cautions and warnings can be displayed to prevent potential errors. The new control features will not adversely increase time required to perform the control actions from the HMI display.

Auxiliary Control Board hand-switches used for automatic control will be integrated into the control system. This includes automatic control features for the pumps and valves. The existing "local hand-switch stations" that provide pneumatic control of various liquid radwaste control valves will not be replaced. These hand-switches are located at various local radwaste panels and allow the control valve to be selected to the open-auto-close position. Therefore, this design change maintains the ability to perform manual operation of the control valve from the local field hand station.

The key operated switch for FCV-647 (Liquid Radwaste Overboard Valve) will be removed and manual operation function to open the valve and establish flow is maintained through the new digital control system. New features such as administrative controls are added within the digital controls and will require a minimum of a two step function to initiate flow. FCV-647 is located upstream of RCV-18 (Liquid Radwaste overboard valve) and FCV-477 (Liquid Radwaste Dump to the Equipment Drain receiver). In the event of a HI radiation signal, RCV-18 closes and FCV-477 opens to divert flow to the Equipment Drain receiver. The RM -18 radiation monitor, signals to EARS and control circuit for RCV-18 and FCV-477 are maintained independent of the new control system for defense in depth measures. The

valve position indication for FCV-477 & RCV-18 and the High Radiation Level PK64 annunciator alarm displayed on the Auxiliary Board Control room will be integrated into the new system.

The design change does not adversely affect existing alarms and control capability functions for specific setpoints associated with the Westinghouse PLS document 663229-47. These setpoint functions pertain to LHUT level/pressure and Concentrates Holding tank 0-1 temperature and do not require Westinghouse coordination. The Westinghouse PLS document 663229-47 will be updated to reflect that these particular setpoints are part of the DCPP Configuration Management Program, which includes procedures, drawings, Design Criteria Memorandums (DCMs) and/or calculations.

A Functional Requirements Specification (FRS), 663195-32 has been written to specify specific programming and design functional requirements for the digital controller and HMIs. The FRS provides the basis for the plant software QA program and configuration management of the control system. Also data acquisition and control system (DACS) configuration drawing 6023221-1 and communication configuration drawing 6023221-2 have been provided to ensure further control. The overall philosophy for the development of display graphics are consistent with the DCPP Human Systems Interface (HSI) Development Guidelines and found per I&C Obsolescence Management Website (http://wwwnpg/osps/group/sbpllindexfiles/frame.htm).

In general, the new system improves the control and provides human factor enhancements to better support operator tasks and reduce risk of errors.

08-026 Reactor Head Replacement - Unit 2 IHA

Reference Document No.:	DCP M-050863, Rev. 0
Reference Document Title:	Integrated Head Assembly

Activity Description/Summary of Evaluation:

NRC Order EA-03-009, "Issuance of First Revised NRC Order establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," established the frequency of inspections and inspection techniques for reactor vessel closure heads (RVCH). Under-the-head nondestructive testing is required every outage for plants in the high potential cracking category, as defined by EA-03-009. The Diablo Canyon Unit 2 RVCH is in this category because of its length of service time. PG&E has elected to replace the RVCH in order to eliminate the possibility of pressurized water stress corrosion cracking (PWSCC) of the Alloy 600 material.

As part of the RVCH replacement activities, Diablo Canyon Power Plant (DCPP) will replace the existing Unit 2 control rod drive mechanism (CRDM) seismic platform above the existing RVCH with an integrated head assembly (IHA) in order to incorporate enhancements into the replacement structure's design that will help to vastly improve future outage performances and personnel safety for PG&E.

This particular activity entails the replacement of the CRDM seismic platform, located above the RVCH, with an IHA, which includes the following activities:

1. Removal of the existing CRDM seismic support structure.
2. Replacement of the existing CRDM reactor missile shield.
3. Removal of the existing CRDM cooling shroud and ductwork.
4. Removal of the existing RVCH lift rig tripod.
5. Installation of an IHA which incorporates the CRDM cooling ductwork and shroud, the RVCH lifting rig, the CRDM missile shield, and the CRDM upper seismic support and restraints (which includes two (2) new double lugs at the refueling cavity wall to accommodate two (2) new additional seismic tie-rods).
6. Replacement of the existing reactor vessel level indication system (RVLIS) and reactor vessel head vent (RVHV) piping, valves, orifices, and supports.
7. Reuse of the existing core exit thermocouple (CET) cables.
8. Replacement and relocation of existing CRDM cooling fans, as well as reduction in number of fans from four to three.
9. Replacement of the existing CRDM, digital rod position indication (DRPI), loose parts monitoring system (LPMS), LPMS Pinger, Stud Hoist VSD (variable speed drive) Cabinet power and CRDM cooling fan related cables, connectors, and miscellaneous hardware with new components.
10. Replacement of two (2) existing LPMS transducers with transducers of a newer design.
11. Replacement of the existing RVCH insulation with new metal reflective insulation.
12. Installation of new permanent integral radiation shielding to eliminate the need for temporary lead radiation shielding.
13. Addition of new vibration monitoring accelerometers for the upper and lower bearings on each new CRDM cooling fan motor, and associated cables from the accelerometers to the poolside disconnect box.

DCP M-050863 assesses compliance with specified design requirements, discusses evaluations, analyses, and justifies that the IHA is a suitable replacement for the existing component for plant operation. Additionally, DCP M-050863 includes a description of any functional impacts to interfacing

and affected systems, structures, and components due to installation of the IHA and the required changes to design and license basis documents.

Summary of Evaluation:

The existing CRDM seismic structure and various interfacing components and systems will be replaced with an IHA that is equivalent in function to the existing seismic structure design, but incorporates many enhancements into the component and systems design. The 10 CFR 50.59 review considers the mechanical and structural effects of IHA, enhancements associated with the IHA, and the effect of changes in the weight of the IHA on the reactor coolant loop.

The installation of the IHA has resulted in the re-establishment of the seismic and LOCA response spectra due to the IHA weight increase such that the reactor coolant loop, vessel supports, internals, fuel, and attached auxiliary systems can be reanalyzed and/or evaluated. Based on these analyses it has been concluded that the evaluation methodology used for the reactor coolant loop and interfacing components and systems does not involve a change that adversely revises or replaces an FSARU described method of evaluation that is used in establishing the design bases or that is used in the safety analyses.

Due to the installation of the IHA, the DCPP NUREG-0612 submittal and corresponding licensing basis were reviewed to ensure that lifting and movement of the heavier IHA would be in compliance with the DCPP licensing basis. Based on this review it was concluded that installation, lifting, and movement of the IHA inside containment is within the existing DCPP licensing basis.

However, the IHA does require a 10 CFR 50.59 evaluation based on the use of an alternative methodology, Regulatory Guide 1.61 damping values, as described in the FSARU for evaluating and qualification of the IHA. As found in the Evaluation, the use of higher damping values for qualification of the IHA is not a departure from a method of evaluation as described in the FSARU because this alternate methodology "is considered approved by the NRC for its intended application." Therefore, the alternative methodology change does not constitute a departure from the methodology defined by 10 CFR 50.59 and be utilized without prior NRC approval. Additionally, the evaluation methodologies utilized for the remaining components and systems which interface with the IHA do not involve revising or replacing any FSARU described evaluation methodology that is used in establishing the design bases or used in the safety analyses.

Installation of the replacement components does not adversely affect any

FSARU described design function and does not involve a change to any procedure that adversely affects how FSARU described SSC design functions are performed or controlled.

The review also concluded that installation of the replacement components will not involve a test or experiment not described in the FSARU or controlled in a manner that is outside the reference bounds of the design for the SCC, or is inconsistent with analyses or descriptions in the FSARU.

As evaluated, all proposed activities within the scope of this review are within the DCPP Licensing Basis and Technical Specifications and can be implemented without prior NRC approval.

08-028 U2 RI Rack Replacement & LTB Elimination

Reference Document No.:	DCP J-50907, Rev. 0
Reference Document Title:	U2 RI Rack Replacement

Activity Description/Summary of Evaluation:

Existing system:

The instrument rack (RI) in the cable spreading room includes Fisher power supplies, controllers, and manual/auto stations and Moore current switches. With the exception of a Class IB current isolator that interfaces with the EARS system, all components in the RI Rack are Class II. The functions of RI Rack components include providing field instrument loop power; providing alarm inputs to the main annunciator system; providing process controllers for chemical injection, feed water pump recirculation valves FCV-53 and 54, condensate recirculation valve FCV-30, stator coil and cooling water flow valve FCV-31, and hot well level control valves LCV-8 and 12; and providing analog outputs to control room indicators and/or the Plant Process Computer (PPC).

The Fisher AC2 equipment in the RI Rack has been obsolete for many years and will become increasingly difficult to maintain with the passage of time. In addition, existing Westinghouse Hagan manual/auto stations on VB-3 that interface with the RI Rack are obsolete.

DCPP has a load transient bypass (LTB) system. The purpose of the system is to provide a means to prevent loss of suction to the MFWPs upon a full load rejection. Following the Steam Generator Replacement project, the DCPP design basis no longer includes a full load rejection. The maximum design basis load reduction is 50 percent. Per Westinghouse analyses, the licensing bases for DCPP are met with or without LTB. Keeping LTB

exposes DCPP to the negative effects of spurious LTB actuation (which has occurred in the past).

New System:

To resolve obsolescence issues with the existing Fisher AC2 equipment, all instrumentation in the RI Rack will be removed and replaced with Triconex Tricon triple modular redundant (TMR) chassis, a fiber switch, a cabinet maintenance HMI, and new, redundant power supplies. The obsolete Hagan manual/auto stations on VB3 for hotwell level control (LCV-8 and 18) and for stator coil and cooling water (FCV-31) will be replaced with new manual/auto stations with the same fit and function to be provided by NUS. Chemical injection manual/auto stations will be replaced with manual only stations to be provided by NUS.

The new RI Rack equipment and control room manual/auto stations will provide the same functionality as the existing devices with the addition of control functions to improve MFWP suction pressure during load transients (see above). Replacement manual only stations for chemical injection provide sufficient functionality for Operations since the existing manual/auto stations are only operated in manual and the pumps that they control are only used when large chemical additions are required (e.g., in startups). Additionally a data link will be provided to the Plant Data Network (PDN). Existing analog inputs to the PPC will be spared and the same data will be sent via the existing PDN to PPC data link.

To ensure that no Class I devices are in the Class II instrument rack, the Class I isolator in the RI Rack will be moved to Class I panel RCRM.

The PLC in the adjacent EARS cabinet RNMI that was installed per EDT 31194 provides data acquisition for Radiation Monitoring System (RMS) inputs to the EARS system. In the Unit 1 RI Rack DCP J-049907, the PLC and its associated maintenance terminal, power supply, and switch will be removed, the PLC's field termination panels (FIPs) in cabinet RMNI will be reconnected to the new Triconex chassis in the Unit 1 RI Rack, the Unit 1 Triconex will be connected to the PDN, and the EARS data acquisition functionality for both Unit 1 and Unit 2 RMS inputs will be incorporated in the Unit 1 Triconex equipment. This Unit 2 DCP J-50907 will relocate the Unit 2 RMS inputs to the Unit 2 RI Rack Triconex I/O and will implement a data link to EARS via the PDN for the Unit 2 RI rack. Therefore, after this change (DCP-50907), each unit's radiation monitoring system inputs and data link to the PDN will be separated and dedicated to their own units. The PDN and EARS remain common.

To provide power to the RI Rack during bus transfers following unit trips, the RI Rack will be connected to the Digital Feedwater Control System UPS.

Load transient bypass will be eliminated by this project. Simulator runs (see DCP Attachment 11) predict that there will be adequate NPSH for the feed pumps under all load rejection and load reduction scenarios provided additional logic is implemented as follows. This DCP adds control logic and I/O to close LCV-12, open FCV-31, and start the standby condensate/condensate booster pump set upon a load drop anticipate or a loss of MFWP load ramp.

Summary of Evaluation:

This modification does not require prior NRC approval.

There is no adverse impact to the design functions of the RI rack. There is no adverse impact due to the deletion of load transient bypass logic. A thorough FMEA was performed in the OCP summary and evaluation to support this conclusion.

08-029 Replace AFHBVS Control System (Unit 2)

Reference Document No.:	DCP J-50927, Rev. 0
Reference Document Title:	Unit 2 POV Panel Replacement

Activity Description/Summary of Evaluation:

Install a new Auxiliary and Fuel Handling Building Ventilation System (AFHBVS) control system. The new AFHBVS control system will be designed to replicate the operation of the existing control system as closely as possible.

The replacement AFHBVS control system will be based on a safety-related programmable logic controller (PLC) developed and supplied by Triconex, a division of Invensys Systems. The product line of the safety-related controller is referred to as TRICON. The TRICON is a state-of-the art logic controller that provides fault tolerance by means of Triple-Modular Redundant (TMR) architecture. The TRICON has been approved by the USNRC for safety-related applications and was issued a Safety Evaluation Report December 2001.

As with the existing system the new AFHBVS control system will be comprised of two subsystems; the Auxiliary Building Ventilation System (ABVS) and the Fuel Handling Building Ventilation System (FHBVS). The new AFHBVS control system equipment will be mounted in panels POV1 and POV2, which are located in the main control room. The new TMR

controller will use three identical channels each of which will independently execute the control program in parallel with the other two channels. Specialized hardware/software voting mechanisms qualify and verify all digital inputs and outputs from the field. Each channel is isolated from the others; no single point of failure in any channel can pass to another. If a hardware failure occurs on one channel, the other channels override it. A faulted card can be removed and replaced while the controller is online without interrupting the system operation and safety-related functions.

The two POV panels along with the relay control panels RCV1 and RCV2 located in the cable spreading room make up two redundant trains of the AFHBVS control system. POV1 and RCV1 make up one train. POV2 and RCV2 make up the redundant train. The redundant AFHBVS trains each provide a redundant train of ABVS and a redundant train of FHBVS.

As with the existing system the controller in POV1 will control fans E-1, E-5, S-1 and S-33 (Unit 1 S-31) and the controller in POV2 will control fans E-2, E-6, S-2 and S-34 (Unit 1 S-32). Signals from both POV1 and POV2 are required to start Fan E-4. Each controller monitors the fans and dampers controlled by the other controller to provide back-up fan actuation and facilitate required automatic ventilation mode changes.

The new AFHBVS control system utilizes redundant sensor and output devices to prevent a single failure of a field device from causing a loss of safety related functions. The digital technology provides improved system reliability, availability, fault isolation capability, and operator interface.

The new AFHBVS control system will increase ease of operation by adding touch screen displays at each of the POV panels and VB4.

The new system increases the reliability and dependability of the AFHBVS, provides diagnostic tools and increases ease of maintenance and trouble shooting.

Summary of Evaluation:

The existing system logic diagrams, design criteria memorandums and surveillance test procedures were used to replicate the new AFHBVS control system operating logic. There are no changes to the design bases functions of the AFHBVS as described in FSARU sections 9.4.2.1 and 9.4.4.1. Minor AFHBVS control system description changes have been made to FSARU sections 3.10.2.15, 9.4.2.2, and 9.4.4.2. The FSARU change is attached to DCP J-05927. There is no change in the operating modes of the AFHBVS. There is no change in the status or position of any AFHBVS fans or dampers in either normal, accident or post accident modes. This design change does

not affect fan air flow volume, damper positions, filter banks or room temperatures. There is no change to any of the design bases accidents analysis described in UFSAR Chapter 15.4 and 15.5. There is no change to the any licensing bases documents including plant technical specifications.

The new AFHBVS control system utilizes software to perform logic functions previously performed by discrete components. The system software will be developed in accordance with CF2.ID9, including independent third party verification by Triconex who is the qualified supplier of the system. The software system will be thoroughly verified and tested to ensure no common mode failures have been created.

Upon loss of 125 VDC control power to one train of the AFHBVS control system a new automatic "S" TEST signal will be entered on both trains of the AFHBVS system to align the command signals from the unaffected panel with the failsafe signals from the affected panel. Since this was previously a required manual operator action on the existing system automating this feature is considered an operator convenience and not a new design function.

The mimic bus LED indicators on the POV panel doors will be replaced with design Class I LED indicators on the edge of the I/O cards. Additionally design Class II operator interface monitors will be available on POV1, POV2 and VB4 to provide system status displays. Although this is a fundamental change in the manner in which the fan status and damper position is displayed it is not considered adverse in that the Class I indicators on the I/O are easily accessible and are equivalent to the mimic bus LED indicators. The additional Class II indication of the operator interface monitors have been designed for human factors to provide easy to read fan status, damper position, alarm status and diagnostics.

09-002 Use of Dual-Analysis Approach for LONF/LOAC

Reference Document No.:	M-050790 & M-049790, Rev. 1/0
Reference Document Title:	RSG Component Modification

Activity Description/Summary of Evaluation:

This LBIE documentation supplements LBIE 2007-13 and addresses only the use of an alternative methodology for the loss of normal feedwater and loss of offsite AC power to the station auxiliaries (LONF/LOAC) safety analyses performed to support DCP M-050790 Rev. 0 & 1 and M-049790 Rev 0. Since the entire scope of this supplemental LBIE concerns the use of an alternative methodology, only Question 8 is addressed per NEI 96-07 Section 4.3.8.

The design function of the replacement steam generators (RSGs) is intended to be equivalent or better than the original steam generators (OSGs). One RSG design objective was to increase the operating range for steam generator water level. This was achieved by lowering the lower Narrow Range Water Level (NRWL) sensing taps, which increased NRWL span. A second RSG design objective was to gain margin to the TS LCO for the motor-driven AFW pump (MDAFWP) flow rate. This was achieved by demonstrating that applicable safety analysis acceptance criteria could be met using a lower assumed AFW flow rate per pump, but assuming two pumps providing flow. These changes affected several FSARU Chapter 15 safety analyses and other non-FSARU analyses, and in particular:

- FSARU Section 15.2.8/*15.2.9,* LONF/LOAC
- Auxiliary Feedwater (AFW) Reliability Analysis (Reference 8)

The goal of the LONF/LOAC analysis is to show that DCPP can withstand a loss of normal feedwater transient without filling the pressurizer, assuming worst case initial conditions and a limiting single failure (loss of the turbine-driven auxiliary feedwater pump, or TDAFWP). The goal of the AFW Reliability Analysis is to show that DCPP can withstand a loss of normal feedwater transient, without filling the pressurizer, assuming better estimate plant conditions with only a single MDAFWP available.

Prior to the SGRP, acceptable results for the LONF/LOAC and the AFW Reliability analyses had been demonstrated using a single analysis. This analysis was based on a combination of the FSARU worst case conditions and the beyond-design-basis assumption of only one MDAFWP being available.

Because PG&E chose to reanalyze the LONF/LOAC using an assumed lower AFW flow rate and revised RSG water level setpoints, Westinghouse was no longer able to show acceptable results (i.e., no pressurizer overfill) for the LONF/LOAC using the beyond-design-basis assumption of a single available MDAFWP. Similarly, Westinghouse was no longer able to show acceptable results for the AFW Reliability Analysis using the design basis LONF/LOAC input parameters (e.g. worst case initial conditions).

To show acceptable results, it was necessary for Westinghouse to employ a change in the analysis method for these two analyses. Specifically, the LONF/LOAC and AFW Reliability analyses were performed separately, each using slightly different input assumptions. This "dual-analysis" method has been reviewed and approved by the NRC for other Westinghouse 4-loop plants.

Summary of Evaluation:

Major conclusions established in this LBIE documentation are:

- Design objectives for operation with the Replacement Steam Generators (RSGs) required a change in the methodology used for the LONF/LOAC safety analyses (FSARU Section 15.2.8 / *15.2.9).*
- The required change in methodology, a "dual analysis approach" for LONF/LOAC and Auxiliary Feedwater reliability, was approved for the Callaway Plant in an NRC Safety Evaluation that is referenced in this LBIE.
- The NEI 96-07 guidance for application of 10 CFR 50.59, endorsed by the NRC in RG 1.187, specifies the following considerations / requirements need to be addressed / satisfied to demonstrate that a methodology change does not require NRC approval (NEI 96-07 Section 4.3.8.1):

  1. Callaway and DCPP share the similarities in design and configuration pertinent to the application of the methodology.
  2. The methodology is technically applicable to DCPP.
  3. Comparison of the inputs, assumptions, and methodology used by both plants demonstrates that the approved methodology is applied conservatively and *en toto* by DCPP (NEI 96-07 Section 4.3.8.2).
  4. All applicable terms, limitations, and conditions for its approved use are satisfied.
  5. Westinghouse qualification to apply the methodology is affirmed.

- The assumption in the LONF/LOAC analysis of two motor-driven AFW pumps supplying flow and the assumption in the AFW reliability analysis of better-estimate conditions are elements of the NRC approved "dual analysis" methodology.
- Differences in the value of input parameters for DCPP and Callaway are acceptable because the methodology permits establishing those values on the basis of plant-specific considerations (NEI 96-07 Section 3.8). Gaining margin as a result of using a replacement methodology that is NRC-approved, technically applicable, and meets all terms and conditions is an acceptable outcome (NEI 96-07 Section 4.3.8.3 Example 2).
- The proposed activity involves a change in methodology only and therefore only a response to LBIE Section 1 question 8 [10 CFR 50.59(c)(2)(viii)] is applicable per NEI 96-07 guidance (NEI 96-07 Section 4.3.8).
- The proposed activity does not result in a departure from a method of evaluation described in the FSARU per NEI 96-07 guidance.

This is the first implementation for DCPP of the dual-analysis approach to

separately address accident analysis and AFWS reliability concerns for the loss of normal feedwater (with and without offsite power available, or LONF/LOAC) event. Previously, a single bounding analysis was performed combining conservative accident analysis assumptions and the reduced AFW flow consistent with a single MDAFWP. This resulted in an analysis that was overly conservative.

The analysis performed for the replacement steam generator (RSG) for the LONF/LOAC transient uses the NRC approved RETRAN-02W computer code (WCAP-14882-P-A, "RETRAN-02 Modeling & Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999). The assumptions used in the analysis are designed to maximize the time to reactor trip and to minimize the energy removal capability of the AFWS.

In conjunction with the LONF/LOAC accident analysis, a separate "better estimate" analysis is performed to address the reliability of the Auxiliary Feedwater System (AFWS). The reliability analysis is performed in a manner similar to that described for the accident analysis (e.g., same computer code), but assumes only a single MDAFWP is available to feed two of the four SGs. The analysis cases assume better estimate conditions for several key input parameters. Specifically, initial conditions (NSSS power, RCS pressure and temperature, pressurizer level), and reactor trip and equipment setpoints are assumed to be at their nominal values. A better estimate decay heat model, consistent with American Nuclear Society (ANS) 1971 full decay heat with no uncertainties, is used.

Utilizing the dual-analysis approach, with both analyses assuming the failure of the turbine-driven auxiliary feedwater pump (TDAFWP) as the limiting single failure, allows the plant to address both topics separately while continuing to show that the acceptance criterion for this event (preventing pressurizer filling) is met for both scenarios. By demonstrating that acceptable results are achieved in the better estimate analysis crediting a single MDAFWP, the accident analysis can be performed assuming the operation of both MDAFWPs. The dual analysis approach has been previously used by Westinghouse and in other LONF analyses and accepted by the NRC, most recently for the Callaway RSG program (Callaway Plant, Unit 1 - Issuance Of Amendment Regarding The Steam Generator Replacement Project (TAC NO. MC4437), September 29, 2005).

NEI 96-07 Rev. 1, Section 4.3.8.2 states that a new method of evaluation may be evaluated under 10 CFR 50.59 and utilized without submittal for NRC approval if the method has been approved by the NRC for the intended application. This involves showing in the 50.59 Evaluation that the method is approved for the type of analysis being conducted and applicable terms, conditions, and limitations for its approved use are satisfied.

Comparison of DCPP and Callaway - Design and Configuration

DCPP and Callaway are both 4-loop Westinghouse NSSS plants of similar design and power level. The AFW system arrangement for both plants is typical of Westinghouse NSSS plants, with a TDAFWP that feeds all four steam generators, and two MDAFWPs, each of which feeds two SGs.

Comparison of DCPP and Callaway - LONF/LOAC & AFW Reliability Analyses

The results of the DCPP and Callaway LONF/LOAC analyses are similar, in that the pressurizer does not become water-solid and the primary and secondary side pressures remain below 110 percent of the design values. The dual analysis approach used for the Callaway and DCPP analyses were both performed by Westinghouse using the same RETRAN-02W code and Westinghouse methodology. The Westinghouse LONF/LOAC methodology for the FSARU Chapter 15 analysis involves the application of conservative assumptions to maximize the challenge to the acceptance criterion. As stated in the Callaway SER (Reference 3), the assumptions applied in the analysis are intended to delay the time of reactor trip and to minimize the energy removal capability of the AFW system. For parameters with calculated uncertainties, the uncertainties are applied in the conservative direction to establish threshold values for use in the analysis. Therefore, the approach and the methodology are the same.

In comparing the inputs and assumptions used by Callaway and DCPP, many of the parameter values and assumptions are identical. For others, the DCPP value is more conservative. The remaining parameter values are values derived directly from the physical characteristics of SSC or processes in the plant, including flow rates, temperatures, pressures, dimensions or measurements (e.g., volume, weight, size, etc.), and system response times. Such parameters are defined in NEI 96-07 guidance as inputs to the methodology. In other words, the inputs that differ are not elements and are not considered part of the methodology based on NEI 96-07 guidance.

The assessment of technical applicability of the LONF/LOAC methodology to DCPP per NEI 96-07 guidelines concluded the following:

1. The application of the methodology is consistent with DCPP's licensing basis.
2. The methodology does not supersede a methodology addressed by other regulations such as 10 CFR 50.46, 10 CFR 50.55a or the plant technical specifications.
3. The methodology is consistent with relevant industry standards.
4. The computer code used (RETRAN-02W) has been installed in

accordance with applicable software quality assurance requirements, is approved for Westinghouse PWR Non-LOCA safety analysis, has been applied consistent with capabilities and limitations, and industry experience with the computer code has been appropriately considered.

5. DCPP is designed and operated in the same manner as Callaway.
6. There are no differences in plant configuration or licensing bases that could invalidate the application.
7. Westinghouse is qualified to apply the methodology.
8. The methodology has been applied *en toto.*

Therefore, the dual-analysis LONF/LOAC approach used for Callaway and approved by the NRC is technically applicable for use at DCPP and was performed by Westinghouse in accordance with all applicable terms, limitations, and conditions on its use prescribed by the NRC. Based on the foregoing, the proposed activity does not result in a departure from a method of evaluation described in the FSARU used in establishing the design bases or in the safety analyses.

09-003 Use of Alternative Methodologies for RSGs

Reference Document No.:	M-050790 & M-049790, Rev. 0/1 and 0
Reference Document Title:	RSG Component Modification

Activity Description/Summary of Evaluation:

This LBIE documentation supplements LBIE 2007-13 (Reference 1) and addresses only the use of the following alternative methods of evaluation in analyses performed to support steam generator replacement (SGRP, References 2, 3, & 4). Since the entire scope of this supplemental LBIE concerns the use of alternative methods of evaluation, only Question 8 is addressed per NEI 96-07 Section 4.3.8 (Reference 5).

(1) RETRAN-02W Code

Prior to SGRP, the analyses of record (AOR) for several non-LOCA accident analyses were performed by Westinghouse (WEC) using the LOFTRAN transient response code. For the Replacement Steam Generator (RSG) analyses, RETRAN-02W was used by Westinghouse instead of LOFTRAN. The RETRAN-02W code was approved for use by the NRC in Reference 6. RETRAN-02W was used the first time at DCPP for the following FSARU Chapter 15 analyses performed for the RSG Project:

- Loss of Normal Feedwater (LONF) (FSARU 15.2.8)
- Loss of Offsite Power to Station Auxiliaries (LOAC) (FSARU 15.2.9)

- Excessive Heat Removal due to Feedwater Malfunction (FWM) (FSARU 15.2.10)
- Sudden Feedwater Temperature Reduction (FSARU 15.2.11)
- Main Steam Line Break (MSLB) & Feedwater Line Break (FLB) (FSARU 15.4.2)
- Steam Generator Tube Rupture (SGTR) (FSARU 15.4.3)
- MSLB Mass & Energy (FSARU Appendix 6.2C.4)

(2) GOTHIC Code

Prior to SGRP, the analyses of record (AOR) for containment response for the LOCA and MSLB events used the COCO computer code (FSARU Appendix 6.2C.4). For the RSG analyses (FSARU Appendix 6.2D.4), the GOTHIC code version 7.2 was used by Westinghouse instead of COCO. GOTHIC Code version 7.2 was approved for use for containment integrity analyses by the NRC in a Safety Evaluation related to Amendment No. 97 to Renewed Facility Operating License No. DPR-18, R. E. Ginna Nuclear Plant, Inc. (Reference 7). The GOTHIC code (Generation of Thermal-Hydraulic Information for Containment) is used for performing both inside and outside of containment pressure and temperature design basis analyses. The GOTHIC code has been used for performing containment integrity analyses by various licensees.

(3) Iodine spiking factor of 335 for SGTR

Prior to steam generator replacement, the AOR for the SGTR event used an iodine spiking factor of 335 for the exclusion area boundary (EAB) and a factor of 500 for the low population zone (LPZ) and control room dose analyses. For the RSG analyses, a factor of 335 was uniformly used for the EAB, LPZ and control room dose analyses. For the AOR, the NRC approved the use of an iodine spiking factor of 335 for DCPP SGTR analysis without specifying a particular dose calculation, i.e., EAB, LPZ, or control room (Reference 8). In the related License Amendment Request (Reference 9), however, the analysis that utilized the change from 500 to 335 used the 335 value only for the EAB case, the limiting case in terms of margin to the limit.

(4) Modeling of MSSVs

Prior to SGRP, the AOR for the LONF/LOAC and FLB accident analyses modeled the main steam safety valves (MSSVs) with a 3 percent opening tolerance (drift) and 3 percent accumulation (pressure increase required for full opening). In the new analyses performed for the RSGs, the assumption for accumulation models 5 psi instead of 3 percent of the setpoint pressure, since 3 percent accumulation is considered overly conservative.

Summary of Evaluation:

Major conclusions established in this LBIE documentation are:

For (1) RETRAN-02W Code and (2) GOTHIC Code

- Based on guidance for application of 10 CFR 50.59 (NEI 96-07, Section 4.3.8.2), endorsed by the NRC in RG 1.187, the method has been approved for use by the NRC and can be utilized by DCPP without site-specific approval because:

  1. The method is technically applicable to DCPP.
  2. The method is applied en toto.
  3. All applicable terms, limitations, and conditions for the approved use are satisfied.
  4. The method is applied by a qualified organization.

For (3) Iodine spiking factor of 335 for SGTR

- The change is to an element of the SGTR dose calculation method that has been previously approved for use at DCPP and is appropriate for application to the SGTR off site and control room dose analyses (NEI 96-07 Section 4.3.8.1).

For (4) Modeling of MSSVs

- There is some overlap between defining this change as strictly an element of a method vs. an input to the analysis. To ensure that all applicable 50.59 considerations are captured in this LBIE, this item is addressed in the Screen as an input in question 2.a (effect on design function) and as an element of a methodology in question 2.c (change to an evaluation methodology).
- Considered as a change to an element of methodology, the change screens in because it is adverse but the results are essentially the same (NEI 96-07 Section 4.3.8.1)
- Considered as a change to an input to the analysis, the effect is not adverse and this aspect screens out. Therefore, for item (4) only question 8 [10 CFR 50.59(c)(2)(viii)] is addressed in LBIE Section 1.

For alternative methods (1), (2) & (3) listed above

- The proposed activity involves a change in a method of evaluation only and therefore only a response to LBIE Section 1 question 8 [10 CFR 50.59(c)(2)(viii)] is applicable per NEI 96-07 guidance (NEI 96-07 Section 4.3.8).

For all four alternative methods listed above

- The proposed activity does not result in a departure from a method of evaluation described in the FSARU per NEI 96-07 guidance.

Discussion of Alternative Methods

(1) RETRAN-02W

The RETRAN-02W Safety Evaluation (SE) identifies three conditions of acceptance for plant-specific applications, which are summarized as follows along with the justifications for DCPP:

1. NRC approval of WCAP-14882-P-A was limited to the list of transients included in the SE. The NRC list encompasses all of the analyses performed for the RSG Project.
2. NRC approval of WCAP-14882-P-A was limited for use on Westinghouse designed 4-, 3-, and 2-loop plants of the type that are currently operating. The Diablo Canyon Power Plant (DCPP) consists of two 4-loop Westinghouse-designed units that were "currently operating" at the time the Safety Evaluation was written (February 11, 1999).
3. NRC approval of WCAP-14882-P-A was conditioned on the use of conservative input and specified that acceptable methodology for developing plant-specific input is discussed in WCAP-14882-P-A and in WCAP-9272-P-A. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A, the safety analysis input values used in the DCPP analyses were selected to conservatively bound the values expected in subsequent operating cycles.

It is also noted that the NRC has approved the use of RETRAN-02 by DCPP for the Loss of External Electrical Load and/or Turbine Trip (LOL/TT) for the overpressure concern and for the Spurious Safety Injection (SSI) pressurizer overfill analysis (FSARU 15.2.7 and 15.2.15).

DCPP contracted with the plant's NSSS vendor to perform the listed analyses. Westinghouse is qualified to utilize the RETRAN-02W computer code and performed the listed analyses according to the NRC-approved methodology. Since all applicable aspects of the approved methodology were applied for the DCPP analyses, the methodology has been applied *en toto.*

(2) GOTHIC Code

The DCPP GOTHIC containment evaluation model was developed based on GOTHIC code version 7.2. GOTHIC code version 7.2 is based on earlier GOTHIC code versions approved for the Kewaunee Nuclear Power Plant Containment Integrity Analysis with the changes identified in Appendix A of the GOTHIC Containment Analysis Package User Manual. Kewaunee, Ginna, and DCPP have dry containment designs with similar passive heat conductors and active heat removal systems as accepted by the NRC for the application of GOTHIC. The latest code version is used to take advantage of the diffusion layer model (DLM) heat transfer option. This heat transfer option was approved by the NRC for use in Kewaunee containment analyses with the condition that the effect of mist be excluded from what was earlier termed as the mist diffusion layer model (MDLM). As in the Ginna containment analysis, the GOTHIC containment modeling for DCPP has followed the conditions of acceptance placed on Kewaunee. None of the user-controlled enhancements added to version 7.2 were implemented in the DCPP containment model.

DCPP contracted with the plant's NSSS vendor to perform the listed analyses. Westinghouse is a member of the GOTHIC user group and is qualified to utilize the GOTHIC computer code. Numerical Applications, Inc (NAI) developed GOTHIC and maintains the code for EPRI. NAI issues software trouble/errors reports quarterly and Westinghouse addresses each issue listed per the applicable Westinghouse quality assurance procedures for software control. The same methodology was used for modeling the containment response to design basis LOCA and steamline transients as was done with Ginna with GOTHIC version 7.2. Since all applicable aspects of the approved method of evaluation were applied for the DCPP analyses, the method has been applied *en toto*.

(3) Iodine Spiking Factor of 335 for SGTR

There is no technical reason why 335 would apply to the EAB distance and not to the LPZ and CR distances from the core, since the iodine spiking factor is related to the core iodine source term and is applicable to all distances from the core. This is supported by RG 1.183 (Reference 10), which specifies a spiking model that assumes a factor of 335 for offsite dose consequences (EAB and LPZ) and states the following for control room dose determinations:

"The radioactive material releases and radiation levels used in the control room envelope dose analysis should be determined using the same source term, in-plant transport, and release assumptions used for determining the EAB and the LPZ dose values, unless these assumptions would result in nonconservative results for the control room envelope."

The 335 Iodine spiking factor has the same relative impact on the control room dose analysis as on the EAB and LPZ calculations. Therefore, it does not introduce an additional or unique non-conservatism in the control room dose results compared to the EAB and LPZ results and is appropriate for the control room analysis per the RG 1.183 guidance.

DCPP contracted with the plant's NSSS vendor to perform the indicated analyses. Westinghouse is qualified to utilize an iodine spiking factor of 335 in conjunction with DCPP's license basis methodology (FSARU 15.5.20) and performed the indicated analyses utilizing all applicable aspects as described in the FSARU. Therefore, the method has been applied *en toto.*

Assessment of Technical Applicability per NEI 96-07 guidelines NEI 96-07, Rev. 1, Section 4.3.8.2 states that a new method of evaluation may be evaluated under 10 CFR 50.59 and utilized without submittal for NRC approval if the method has been approved by the NRC for the intended application. This involves showing in the 50.59 Evaluation that the method is approved for the type of analysis being conducted and applicable terms, conditions, and limitations for its approved use are satisfied. NEI 96-07 lists considerations for determining whether the new methodology may be considered "approved for the intended application." Information in the preceding paragraphs supports the responses to the following considerations.

1. The application of the methods of evaluation are consistent with DCPP's licensing basis.
2. The methods of evaluation do not supersede a methodology addressed by other regulations such as 10 CFR 50.46, 10 CFR 50.55a or the plant technical specifications.
3. The methods of evaluation are consistent with relevant industry standards.
4. The computer codes used have been installed in accordance with applicable software quality assurance requirements, are approved for Westinghouse PWR safety analyses, have been applied consistent with capabilities and limitations, and industry experience with the computer codes has been appropriately considered.
5. DCPP is designed and operated in the same manner as plants where the methods have been approved for use.
6. There are no differences in plant configuration or licensing bases that could invalidate the application.

<u>(4) Modeling of MSSVs</u>

The ASME Code, Article NC-7512.1, specifies that the steam generator safety valves must be fully accumulated within 3 percent above the set

pressure (but does not dictate how to model the movement to full lift, except the valves must operate without chattering). Thus, safety analyses historically assumed that such spring-loaded valves open linearly over a pressure range of 3 percent of the opening pressure. However, the ASME Code requirement is not consistent with actual valve behavior. Extensive industry testing demonstrates that once a valve reaches the point of first stem movement, the valve immediately "pops" wide open relieving steam at, or near, full capacity with no appreciable accumulation. The use of a 5 psi valve accumulation is now part of the standard Westinghouse methodology and has been used in safety analyses for numerous plants. Note that the previous assumption of a 3 percent accumulation, in addition to the 3 percent valve tolerance, is overly conservative in that it results in multiple MSSVs opening at a pressure above the 110 percent design pressure limit. Thus, those valves could not be considered operable (capable of performing their intended safety function).

To conservatively model actual valve behavior, the rapid valve "pop" is simulated in the WEC model as a ramp function from closed to fully open over a 5 psi pressure range. The DCPP MSSVs are type 3700 valves manufactured by Dresser Industries to ASME code requirements for nuclear power applications. WEC confirms that this type valve will behave according to the 5 psi model if properly maintained per the manufacturer's maintenance manual and backpressure remains below 10 percent (Reference 11). At DCPP, the valves open to atmosphere so that backpressure is not an issue. Since maintenance is performed per DCPP procedures that follow the vendor manual guidance, valve behavior is appropriately simulated by the WEC model.

The MSSV modeling is considered an element of the methodology because a mathematical model is utilized to simulate the response of the valve when the setpoint is exceeded. A linear ramp is not exactly representative but is considered conservative and avoids computer code instabilities (i.e., specific limitations of a computer program). This is in accord with the definition and discussion of elements of methodology in NEI 96-07 Section 3.10. However, since modeling of the MSSVs is based on performance characteristics of the valve, this can also be perceived as an input to the analysis. Therefore, the LBIE Screen considers the change under both question 2.a (adverse effect on a design function) and question 2.c (adversely replaces or revises an evaluation methodology).

The results for the analyses performed for the RSG Project using the MSSV modeling described above meet all applicable acceptance criteria: (1) for LONF/LOAC, fuel damage is precluded, RCS and MSS pressures are maintained below 110 percent of design pressures, and the pressurizer does not become water-solid (i.e., a more serious plant condition is not generated), and (2) for FLB, RCS and MSS pressures are maintained below

110 percent of design pressures, long term core cooling is not precluded, and offsite dose consequences are within guidelines. In addition, sensitivity runs were performed to benchmark the 5 psi accumulation model with the 3 percent accumulation model used in the AOR (Reference 11). The results for the peak pressurizer water volume, used to ensure that the primary acceptance criterion of concern is met for the LONF/LOAC transients (i.e., the pressurizer does not fill with water) differ by less than 0.5 percent between the two models. The results for the peak RCS hot leg temperature, used to ensure that the primary acceptance criterion of concern is met for the FLB transient (i.e., no hot leg saturation) differ by 0.3 percent.

The benchmarking was done for the same set of plant conditions to ensure that the results are comparable. The comparison of the new and old models also considered time behavior of results and found it to be similar. The differences are deemed to be within the margins for error for the analyses and the results are therefore essentially the same per NEI 96-07, Section 4.3.8.1.

09-004 Tavg & Tfeed Ranges Program

Reference Document No.:	DCP 1000000120, Rev. 0
Reference Document Title:	Tavg & Tfeed Ranges Prog.

Activity Description/Summary of Evaluation:

Use of 1979 ANS *(2σ)* decay heat curve for SGTR Margin-to-Overfill

The SGTR reanalysis performed for the Tavg & Tfeed Ranges Program at DCPP accounted for Westinghouse Nuclear Safety Advisory Letter, NSAL-07-11 (Reference 1). The NSAL identified that, for certain plants, a less conservative decay heat assumption is more limiting for margin to overfill. The decay heat model is considered an element of the method of evaluation. The method identified in the FSARU is described in WCAP-10698-P-A (Reference 2), a generically approved WEC topical. WCAP-10698-P-A prescribes the use of 120 percent of the 1971 ANS decay heat curve. For the plants described in Reference 1, the 1979 ANS decay heat curve *(-2σ)* is used instead because the less conservative heat curve produces more conservative margin to overfill results. Westinghouse determined that the NSAL applies to DCPP and therefore used the ANS 1979 -2σ model.

Summary of Evaluation:

WCAP-10698-P-A provided the NRC-approved methodology used in the

analysis of record (AOR) for SGTR Margin-to-Overfill (MTO). The SGTR event was re-analyzed for the DCPP Tavg & Tfeed Ranges Program and the decay heat model was evaluated in accordance with the recommendations of NSAL-07-11. The reanalysis determined that, for DCPP as with certain other plants, a lower decay heat model is conservative with respect to margin to SG overfill. Therefore, this element of the approved methodology was changed to the ANS *1979-2σ* model in order to analyze the DCPP SGTR MTO conservatively.

The purpose of the SGTR MTO analysis is not to demonstrate a specific value of margin, but simply to show that SG overfill will not occur because margin exists. The revised SGTR MTO analysis, using a lower decay heat model, demonstrated margin to SG overfill but with a reduced margin compared to the AOR (i.e., a more conservative result).

In accordance with NEI 96-07 Section 4.3.8.1, changing an element of the methodology does not represent a departure from the method of evaluation if the results of the change are more conservative.

09-011 GDC 17 Compliance for 500-kV Backfeed

Reference Document No.:	DDP 1000000263, Rev. 0
Reference Document Title:	GDC 17 Compliance for 500-kV Backfeed

Activity Description/Summary of Evaluation:

Historical Background

In PG&E Letter DCL-99-014 to the NRC dated February 5, 1999, Diablo Canyon provided a copy of Licensing Basis Impact Evaluation (LBIE) Screen 98-114. This LBIE Screen assessed a revision of the FSAR Sections 3 and 8 related to Electric Power. The topic was the 30 minutes allocated for backfeed of power from the 500kV System. The LBIE referenced several Westinghouse letters and WCAP 10541 Supplement 1 to establish that Reactor Coolant Pump (RCP) seal leakage would not exceed the Technical Specification limit of 5 gpm per pump for the 30 minutes needed to implement the 500kV backfeed. The LBIE concluded that two DCPP FSAR Update (FSARU) Chapter 15 accident analyses were bounding for the General Design Criteria (GDC) 17 compliance conditions. These FSARU Chapter 15 accident analyses were cited to verify the fuel design limits and reactor coolant system (RCS) pressure boundary requirements per GDC 17 were met during the time required to establish the 500kV backfeed. Therefore, the 500kV system was found to meet the GDC 17 requirements for a delayed offsite power source to shut down the plant. The LBIE conclusion did not need to credit any operator actions.

NRC Information Notice IN 2005-14 was issued June 5, 2005, which requested licensees to evaluate their Appendix R RCP seal coping strategy for increased RCP seal leakage due to a loss of all seal cooling event consistent with the Westinghouse guidelines in Technical Bulletin TB-04-22 Rev. 1. In response to IN 2005-14, a revised Appendix R RCP seal coping strategy for increased RCP seal leakage, 21 gpm per RCP after 8.33 minutes, was implemented for a complete loss of RCP seal cooling during a fire induced control room evacuation as documented in SAPN 50033478. It should be noted that the 500kV backfeed was not evaluated at this time with respect to IN 2005-14.

The Station Blackout (SBO) analysis was not impacted by IN 2005-14 since an Alternate AC (AAC) source is available to preclude a complete loss of RCP seal cooling event.

PROPOSED ACTIVITY:

This proposed activity involves crediting the Appendix R RCP seal coping strategy for establishing continued compliance with GDC 17 for the 500kV backfeed with respect to IN 2005-14. This proposed activity also involves crediting the RETRAN-02 thermal hydraulic evaluation performed in STA-274 to verify that the Appendix R RCP seal coping strategy does not adversely impact the fuel design limits and reactor coolant system (RCS) pressure boundary requirements for establishing continued compliance with GDC 17 for the 500kV backfeed.

These two activities are being implemented in the document-change-only Design Change DDP 1000000263. This design change modifies the DCPP FSARU and Design Criteria Memoranda (DCM) to establish continued DCPP compliance with 1971 GDC 17 as it relates to the delayed access 500kV offsite power source design function (500kV backfeed event).

WHAT IS/ARE BEING CHANGED AND REASON FOR CHANGES:

1. Revise the DCPP design basis to credit the Appendix R RCP seal coping strategy for increased RCP seal leakage due to the loss of all seal cooling event that occurs during the 500kV backfeed event as required by the conditions of GDC 17.

2. Credit the Appendix R operator actions to isolate RCP seal injection from the Centrifugal Charging Pump (CCP) and Component Cooling Water (CCW) to the thermal barrier heat exchanger (TBHX) prior to the restoration of RCS makeup flow as currently established in EOP ECA-0.0 and EOP AP-8A.

3. Credit the procedure enhancements to EOP ECA-0.0 and operator training as documented in SAPN 50198089 to ensure that the 500kV backfeed is completed within 30 minutes and continues to meet the requirements per GDC 17.

4. Revise the DCPP design basis to clarify that the 30 minute time interval required for the 500kV backfeed to meet GDC 17 requirements, includes the restoration of RCS makeup within the 30 minute Interval.

5. Revise the DCPP design basis to reference the RETRAN thermal hydraulic analysis in STA-274 which evaluated the Appendix R RCP seal coping strategy for an increase in RCP seal leakage to 21 gpm per seal and concluded that the GDC 17 requirements (do not exceed any RCS pressure boundary or fuel design limits) are still met for the 500kV backfeed duration. The additional STA-274 evaluations and conclusions for the 500kV backfeed event are summarized below:

    a. Evaluated the loss of RCS inventory and RCS pressure decrease due to the increased Appendix R RCP seal leakage.
    b. Evaluated the available time margin before a RCS low pressure SI Signal is generated and before the pressurizer is completely drained of liquid.
    c. Evaluated the maximum volume of RCP seal leakage that occurs during the event.

6. Revise the DCPP design basis documents to establish the continued compliance with GDC 17 with respect to the Appendix R RCP seal co in strategy for increased RCP seal leakage due to the loss of all seal cooling event as summarized below:

    d. Revise FSARU Section 5.5 to include an additional description of the Appendix R RCP seal coping strategy for increased RCP seal leakage due to the loss of all seal cooling event per IN 2005-14.
    e. Revise FSARU Section 8.2 to include a description of the revised GDC 17 compliance for the 500kV backfeed associated with the Appendix R RCP seal response evaluation results of ST A-274.
    f. Revise Design Criteria Memorandum S-7, Reactor Coolant System, to include an additional description of the Appendix R RCP seal coping strategy for increased RCP seal leakage due to the loss of all seal cooling event per IN 2005-14.
    g. Revise DCM S-8, Chemical and Volume Control System to include an additional description of the Appendix R RCP seal coping strategy for increased RCP seal leakage due to the loss

of all seal cooling event per IN 2005-14.

h. Revise DCM S-61B to include a description of the revised GDC 17 compliance for the 500kV backfeed associated with the Appendix R RCP seal response evaluation results of STA-274.

i. Revise DCM T-13, Appendix R Fire Protection, to reflect the Appendix R RCP seal coping strategy and increased RCP seal leakage for the loss of all seal cooling event per IN 2005-14 is consistent with the 500kV backfeed event per GDC 17 as described in DCM S-61B.

j. Revise DCM T-42, Station Blackout to clarify that the Appendix R RCP seal coping strategy per IN 2005-14 does not impact the SBO analysis since an Alternate AC (AAC) source is available to preclude a complete loss of seal cooling event.

Summary of Evaluation:

This proposed activity requires prior NRC approval to implement based on YES responses to 10 CFR 50.59 questions 1 and 6 as summarized below.

1. - Crediting the Appendix R RCP seal coping strategy for increased RCS leakage due to a loss of seal cooling during the 500kV backfeed event is now conservatively being considered a FSARU 15.2.13 accidental RCS depressurization event for which it had not previously been classified. Since a detailed evaluation of the 500kV event probability has not been performed, it is conservative per 10 CFR 50.59 to assume that the overall probability of this FSARU event occurring during the life of the plant has increased. Therefore, this proposed activity represents more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSARU.

6. - The Appendix R RCP seal coping strategy includes operator actions to ensure RCP seal integrity during a loss of seal cooling event. The operator actions are required to prevent a new potential for a gross RCP seal failure related to the loss of RCP seal cooling during the 500kV backfeed event. Therefore, this proposed activity creates the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the FSARU since it introduces an operator action and potential operator error for the 500kV backfeed event, where no previous operator action was required.

09-012 Holtec Vertical Cask Transporter

Reference Document No.:	DCP N-49773, Rev. 0

Reference Document Title:	Holtec Vertical Cask Transporter

Activity Description/Summary of Evaluation:

This LBIE replaces LBIE 2008-017 and implements the use of the Holtec Vertical Cask Transporter (VCT), which has been designed for use at the Diablo Canyon Power Plant and ISFSI facility. Changes from LBIE 2008-017 make the method of complying with the 22-hour limit in the CTF more generic to avoid invalidating this LBIE should the method of compliance be changed in the future. The contingency provided in LBIE 2008-017 is currently in the FSAR; this LBIE removes the specific requirements and states that the contingency plan will be provided in the FSAR. Also, reference to LPT "rails" has been changed to the current design of LPT "tracks."

The VCT is designed to transport vertical a loaded or unloaded HI-TRAC transfer cask or HI-STORM overpack on site. The VCT will provide for all lifting and transport requirements associated with movement of a loaded transfer cask from just outside the fuel handling building (FHB) to the cask transfer facility (CTF) and a loaded overpack from the CTF to the ISFSI pad. The VCT also performs the lifting requirements for the transfer of a loaded Multiple Purpose Canister (MPC) between the transfer cask and the overpack at the CTF.

The VCT is designed to be capable to transport a loaded HI-TRAC up a 10 percent grade. In Holtec Calc HI-012768 R3 and PG&E Calc OQE-014, the VCT is evaluated for seismic stability and sliding during a design basis earthquake while carrying a loaded HI-TRAC. Based on the modeling methodology used in these calculations the change in the center of gravity from carrying the cask vertically or horizontally does not affect the results of the calculations and the results are bounding for both configurations.

The VCT was always designed to be capable of vertical and horizontal carry. In addition, the VCT has always been capable of lifting the maximum load of a loaded HI-STORM overpack; however, previously it was not the primary lifting mechanism at the CTF as there was a CTF jackscrew system that performed this action. This jackscrew system has been eliminated in the current design.

The VCT has also been fitted with a cable winch used to pull the Low Profile Transporter (LPT) with a loaded transfer cask out of the FHB. The LPT is pulled along tracks that allow it to exit the FHB and moved to a position within the VCT footprint where the VCT will capture the transfer cask and carry it to the CTF. The VCT has been designed to support this function during any associated design basis accident. The VCT has a licensing basis limit of 50 gallons of fuel on board based on an engulfing fire for the

HISTORM system. The licensing must basis for the DC ISFSI provides that the 50 gallons not be capable of entering the CTF during use of the VCT. The NRC safety evaluation report indicates that this may be performed by the use of a removable fuel tank that will be moved away from the CTF during VCT use. In the licensing submittals for the DC ISFSI, PG&E provided the use of the removable fuel tank as an example of possible solutions for this issue. It was not provided as the actual solution. The actual VCT design provides for an on-board 45 gallon diesel fuel tank and a catch pan that is designed to catch and divert any leakage away from the CTF through a hose connection on the pan. This design meets the requirements of the licensee in that it will ensure that any fuel leakage cannot reach the CTF.

The VCT operates between just outside of the Unit 2 fuel handling building up to and at the CTF; and between the CTF and the ISFSI pads. The VCT does not enter any 10 CFR 50 structures, however, its route does move within the 10 CFR protected areas. As a result, the 10 CFR 50 licensed facilities have been reviewed for effects of the VCT.

Summary of Evaluation:

The design of the VCT does not affect criticality, confinement, shielding capability, or thermal performance of the HI-STORM 100 system. The transporter has been designed to applicable codes and standards to ensure its structural integrity. The change in the primary function of the VCT to perform all the lifting at the CTF required further review in this LBIE. In addition, the use of the onboard catch pan for the diesel fuel tank also needed further review.

As a result of that review, the effects of these changes are within the design limits of the licensed DC ISFSI facility and the VCT remains capable of performing all design and licensing basis functions, and all operational requirements. None of the VCT changes are considered adverse and are acceptable under 10 CFR 72.48 and 10 CFR 50.49 criteria.

09-020 Replace AFHBVS Control System (Unit 1)

Reference Document No.:	DCP J-100000106, Rev. 0
Reference Document Title:	Unit 1 POV Panel Replacement

Activity Description/Summary of Evaluation:

Install a new Auxiliary and Fuel Handling Building Ventilation System (AFHBVS) control system. The new AFHBVS control system will be

designed to replicate the operation of the existing control system as closely as possible.

The replacement AFHBVS control system will be based on a safety-related programmable logic controller (PLC) developed and supplied by Triconex, a division of Invensys systems. The product line of the safety-related controller is referred to as TRICON. The TRICON is a state-of-the art logic controller that provides fault tolerance by means of Triple-Modular Redundant (TMR) architecture. The TRICON has been approved by the USNRC for safety-related applications and was issued a Safety Evaluation Report December 2001.

As with the existing system the new AFHBVS control system will be comprised of two subsystems; the Auxiliary Building Ventilation System (ABVS) and the Fuel Handling Building Ventilation System (FRBVS). The new AFHBVS control system equipment will be mounted in panels POV1 and POV2, which are located in the main control room. The new TMR controller wilt use three identical processors each of which will independently execute the control program in parallel with the other two processors. Specialized hardware/software voting mechanisms qualify and verify all digital inputs and outputs from the field. Each processor is isolated from the others; no single-point of failure in any processor can pass to another. If a hardware failure occurs on one processor, the other processors override it. A faulted card can be removed and replaced while the controller is online without interrupting the system operation and safety-related functions.

The two POV panels along with the relay control panels RCV1 and RCV2 located in the cable spreading room make up two redundant trains of the AFHBVS control system. POV1 and RCV1 make up one train. POV2 and RCV2 make up the redundant train. The redundant AFHBVS trains each provide a redundant train of ABVS and a redundant train of FHBVS.

As with the existing system the controller in POV1 wilt control fans E-1, E-5, S-1 and S-31 and the controller in POV2 will control fans E-2, E-6, S-2 and S-32. Signals from both POV1 and POV2 are required to start Fan E-4. Each controller monitors the fans and dampers controlled by the other controller to provide back-up fan actuation and facilitate required automatic ventilation mode changes.

The new AFHBVS control system utilizes redundant sensor and output devices to prevent a single failure of a field device from causing a loss of safety related functions. The digital technology provides improved system reliability, availability, fault Isolation capability, and operator interface.

The new AFHBVS control system will increase ease of operation by adding touch screen displays at each of the POV panels and VB4.

The new system increases the reliability and dependability of the AFHBVS, provides diagnostic tools and increases ease of maintenance and trouble shooting.

Summary of Evaluation:

The existing system logic diagrams, design criteria memorandums and surveillance test procedures were used to replicate the new AFHBVS control system operating logic. There are no changes to the design bases functions of the AFHBVS as described in FSARU sections 9.4.2.1 and 9.4.4.1. Minor AFHBVS control system description changes have been made to FSARU Sections 3.10.2.15, 9.4.2.2, and 9.4.4.2. The FSARU change is attached to DCP J-1000000106. There is no change in the operating modes of the AFHBVS. There is no change in the status or position of any AFHBVS fans or dampers in either normal, accident or post accident modes. This design change does not affect fan air flow volume, damper positions, filter banks or room temperatures. There is no change to any of the design bases accidents analysis described in UFSAR Chapter 15.4 and 15.5. There is no change to the any licensing bases documents including plant technical specifications.

The new AFHBVS control system utilizes software to perform logic functions previously performed by discrete components. The system software will be developed in accordance with CF2.ID9, including independent third party verification by Triconex who is the qualified supplier of the system. The software system will be thoroughly verified and tested to ensue no common mode failures have been created.

Upon loss of 125 VDC control power to one train of the AFHBVS control system a new automatic "S" TEST signal will be entered on both trains of the AFHBVS system to align the command signals from the unaffected panel with the fail-safe signals from the affected panel. Since this was previously a required manual operator action on the existing system automating this feature is considered an operator convenience and not a new design function.

The mimic bus LED indicators on the POV panel doors will be replaced with design Class I LED indicators on the edge of the I/O cards. Additionally design Class II operator interface monitors will be available on POV1, POV2 and VB4 to provide system status displays. Although this is a fundamental change in the manner in which the fan status and damper position is displayed it is not considered adverse in that the Class I indicators on the I/O are easily accessible and are equivalent to the mimic bus LED indicators. The additional Class II indication of the operator Interface monitors have

been designed for human factors to provide easy to read fan status, damper position, alarm status and diagnostics.

09-021 Methodology Changes for RRVCH Design Change

Reference Document No.:	M-050863 DDP1000000075
Reference Document Title:	Integrated Head Assembly

Activity Description/Summary of Evaluation:

Due to design revisions and deficiencies identified in LBIE 2008-026 (Reference 1), this supplement provides revised and additional bases per NEI guidance (Section 4.3.8) to demonstrate that a "departure from a method of evaluation" is not involved. This LBIE supplements / supersedes LBIE 2008-026 (Reference 1) in the following areas concerning methodology changes (for all other aspects related to the Integrated Head Assembly (IHA), refer to LBIE 2008-026):

(1) Use of 4 percent vs. 3 percent damping for the DDE (same as SSE) event for CRDM seismic evaluation.
(2) Use of response spectrum analysis method for a portion of CRDM seismic analysis.

Replacement of the Reactor Vessel Closure Head (RVCH) includes new CRDM housings designed by AREVA. The housings are shown to be equivalent structurally to the existing Westinghouse components, but the DDE seismic analysis to confirm their structural adequacy utilizes a 4 percent damping value vs the existing design basis value of 3 percent. The analysis of the replacement CRDMs also uses the response spectra method as well as the time history method for seismic analysis (vs. only time history method for existing CRDMs). The use of an alternative method and the use of 4 percent damping are evaluated per NEI 96-07 Section 4.3.8 (Reference 2) to determine if a "departure from a method of evaluation" is involved.

Summary of Evaluation:

A structural analysis has been performed for the replacement CRDMs to demonstrate pressure boundary integrity per ASME requirements. The results of the analysis are also evaluated for effects on operability of the CRDM during rod drop. All analysis and evaluation results are acceptable. The 4 percent damping value used for analysis of the replacement CRDMs is specified as appropriate for seismic analysis of piping systems in RG 1.61 Revision 1 (Reference 3). Use of the piping system category damping value for CRDM seismic analysis is deemed conservative (based on testing,

Reference 16) due to the higher damping response for the CRDM compared to typical piping systems.

The response spectra method is considered generally more conservative than the time history method; use of either method is acceptable for dynamic analyses per RG 1.61 Rev. 1; and there are no restrictions or limitations specified in the guidance with respect to using both.

The proposed activity involves changes in methods of evaluation only and therefore only a response to LBIE Section 1 question 8 [10 CFR 50.59(c)(2)(viii)] is applicable per NEI 96-07 Section 4.3.8. Since the methods of evaluation used for the analysis of the replacement CRDMs are approved for use by the NRC in applicable guidance, the proposed activity does not result in a departure from a method of evaluation described in the FSARU.