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PG&E Letter DCL-08-031

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, 2006, through December 31, 2007

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, 2006, through December 31, 2007. 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 that the changes do not require prior NRC approval or require changes to the DCPP Technical Specifications.

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

Sincerely,

James R. Becker

IE47

NRR

why1/4279/R0286818
Enclosure

cc:	Elmo E. Collins, NRC Region IV
	Michael S. Peck, NRC Senior Resident Inspector
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10 CFR 50.59 REPORT OF CHANGES, TESTS, AND EXPERIMENTS
for the Period
January 1, 2006, through December 31, 2007

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

Index of LBIEs

06-002 Update DCM S-17 and FSARU for Tsunami Drawdown Analysis (Unit Common)

Reference Document No.:	DCP M-49791, Rev. 0
Reference Document Title:	Update FSAR & DCM S-17 for Tsunami Drawdown Analysis

Activity Description:

During a review of the Tsunami Drawdown Analysis per A0614550, it was determined that a clarification was required for both Design Criteria Memorandum (DCM) S-17, and the Final Safety Analysis Report Update (FSARU) descriptions of the impact of a tsunami drawdown on the Auxiliary Salt Water (ASW) pumps. Although the existing descriptions of the tsunami drawdown not impacting pump function are accurate, the analysis only describes the case where one ASW pump is supplying one component cooling water (CCW) heat exchanger. Calculation M-953 has shown that there would not be sufficient water level (submergence) to allow ASW pump operation at the design low tsunami level for cases where one ASW pump is supplying two CCW heat exchangers.

In the unlikely event that a tsunami occurs when one ASW pump is operating through two CCW heat exchangers, operators would have to isolate one of the heat exchangers to prevent or mitigate ASW pump cavitation. This potential situation needs to be addressed by the DCM and FSAR.

Summary of Evaluation:

The subject of this Licensing Basis Impact Evaluation (LBIE) is a document change that recognizes that the original DCM S-17 and FSAR description of ASW pump operation at tsunami drawdown conditions is valid for the normal 1-pump, 1-heat exchanger alignment, but that it is possible that an ASW pump could cavitate due to insufficient suction head (submergence) during tsunami drawdown if it is operating in a 1-pump, 2-heat exchanger alignment.

The concern would be that if the ASW system were operating in the 1-pump, 2-heat exchanger alignment (as required for high ocean temperature per Technical Specification [TS]) 3.7.9 and there was a design basis tsunami drawdown, operator action would be required to isolate one of the CCW heat exchangers. (It should be noted that the 1-pump, 2-heat exchanger alignment needs to be considered for this evaluation even if 2 ASW pumps are available, since a single failure criteria applies for this event.)

Operator action would be to isolate one CCW heat exchanger by closing its ASW inlet valve (SW-602/603) and its CCW outlet valve (CCW-430/431). These actions can be taken from the control room. The above steps/information should be included in Casualty Procedure CP M-5, "Tsunami Warning."

The operator action will restore the 1-pump, 1-heat exchanger alignment that was evaluated for the FSAR and DCM S-17 tsunami drawdown analysis. The use of operator action in this case is justified as the 2-heat exchanger alignment is not often used, and there would be multiple indications of a potential problem that could be addressed from the control room.

Crediting the operator re-aligning the system is judged to prevent the potential malfunction of the ASW system in the event of a tsunami drawdown for the nonstandard ASW system alignment. The addition of an operator action to isolate the second CCW heat exchanger does not introduce the possibility of a change in the consequences of an accident because the operator is not an initiator of any accident and no new failure modes are introduced. Operator actions previously evaluated for a complete loss of intake function envelope operator actions required here and those actions have been evaluated per the guidelines of Diablo Canyon Power Plant (DCPP) Administrative Procedure OP1.ID2 and ANSI/ANS 58.8.

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

06-010 Digital Feedwater Control System Replacement (Unit 1)

Reference Document No.:	DCP J-049736, Rev. 0
Reference Document Title:	Replace Digital Feedwater Control System

Activity Description:

This change is similar to Unit 2 LBIE No. 05-008, DCP J-050731, "Digital Feedwater Control System Replacement (Unit 2)." See Pacific Gas and Electric Company (PG&E) Letter DCL-06-050 dated April 21, 2006, for the activity description and evaluation summary.

06-011	**Main Turbine Moisture Separator Reheater Temperature Control Replacement (Unit 1)**

Reference Document No.:	DCP J-49813, Rev. 0
Reference Document Title:	Replace MSR Temperature Controller Unit 1

Activity Description:

The high pressure heating steam to the moisture separator reheaters (MSRs) is controlled by a bypass valve and a main control valve to each MSR. The bypass valves are controlled by an electronic controller located in VB-3 in the main control room. The main valves are opened by hard wired time delay relays. The operator interface is via the reheat control valve controller panel and a main control position indicating panel located on the bench board section of VB-3. The controller and the operator interface are original plant equipment provided by Westinghouse. The equipment is obsolete and no longer supported by Westinghouse.

The design change makes the following changes:

The design change eliminates the existing controller and operator interface as well as the hard wired relays. The controls will be relocated to the Triconex main turbine control system (MTCS). The operator interface will be via the MTCS human machine interface (HMIs) located on CC-3.

Input/Output (I/O) will be connected to the MTCS consisting of each low pressure (LP) rotor's inlet steam temperature, bypass and main control valve demand signals, open and closed positions for each bypass and each main control valve, and an annunciator output. Control logic will be added to the MTCS. Screens for control of the MSRs will be added to the MTCS touch screens on CC-3.

The control logic will retain the functionality of the existing control system with enhancements. Additional alarming will be provided to alert operators to equipment malfunctions and improper mode selections. Permissives for mode selections will be added. Automatic selection of the 400 deg F mode and the reset mode will be provided which can be enabled or disabled by the operator. The operator will be able to override any main valve open if the associated bypass valve position switch malfunctions.

The control system and all associated devices are Class II. The existing control system, power supply, and operator interface is nonredundant. The migration to the MTCS will have the benefits of triple modular redundant I/O channels and logic processors, two power sources, and dual HMIs. The new

controls will improve the reliability of MSR temperature control and provide additional alarming to help operators to diagnose equipment problems earlier. The operator interface modifications and the addition of automatic and manual functions have the potential to introduce an adverse impact on FSARU described design functions.
Summary of Evaluation:
There is no adverse impact to the MSR design function that is described in the FSARU. This change replaces the hardwired operator interface on control board VB3 with new operator screens, on the existing MTCS HMIs on control console CC3. The new HMI screens will provide the same information to the operators as the existing system, but in a different format. The format change in itself is not potentially adverse. The new HMI is not required for steady state operation of the MSR temperature controls, which will continue in automatic operation even if both HMI units fail. The new control features will not have a more than minimally adverse effect on the time required to perform manual control actions from the touch screen display compared to using the existing hard wired operator interface. The new controls will also not have a more than minimally adverse effect on the existing operator interface for the MTCS. The new controls will reduce operator burden during turbine shutdowns by automating time dependent functions (i.e., selection of 400°F mode at 10 percent load and reset after a turbine trip). Therefore, the proposed change did not require prior NRC approval.

06-012 Replace the Positive Displacement Pump with a New Centrifugal Charging Pump (Unit 1)

Reference Document No.:	DCP M-49704 Revision 1
Reference Document Title:	Replace the PDP with a new Centrifugal Charging Pump

Activity Description:

The basic purpose and scope of this design change is to replace the existing Unit 1 Positive Displacement Charging Pump (PDP) with a multi-stage centrifugal pump, similar to those already in use for the two high head emergency core cooling system (ECCS) pumps (CCPs 1-1 & 1-2). The PDP was originally intended to be the normal means of supplying charging flow to the reactor coolant system (RCS). Following this modification, the new CCP 1-3 will perform this function. Operating experience at DCPP and elsewhere with the chemical and volume control system (CVCS) PDP has shown the PDP to require a relatively high maintenance effort. Moreover, its reliability and capacity have proven inadequate to meet all the operational needs of the plant.

In addition to replacing the existing PDP with a multi-stage centrifugal pump, this modification will install new suction discharge and minimum-flow recirculation piping to and from the pump, including appropriate valving, removable piping spools for pump maintenance, and an low temperature over pressure (LTOP) flow-limiting pressure reducing orifice.

Charging flow rate is currently controlled when using the PDP by a pump speed controller, with input and feedback from Pressurizer Level Controller LC-459D. The new CCP 1-3 is a direct drive machine with no speed controller. The speed control scheme is eliminated and HC-459A is being removed from the Main Control Room Console 1CC2. The new CCP 1-3 discharge flow is directed to valve FCV-128, so normal charging flow will be controlled using FCV-128, as is the case when one of the existing centrifugal pumps is used for normal charging. This modification includes design features intended to ensure that FCV-128 is operated within its existing design limits.

Summary of Evaluation:

This modification does not require prior NRC approval. However, this activity screened in, requiring a 10 CFR 50.59 evaluation, because of the following three adverse effects:

1. The new CCP 1-3 will have a higher flow capacity than the PDP it replaces. Specifically, the maximum flow of the CCP is 200 gallons per minute (gpm), compared to the maximum flow of the PDP of 98 gpm. For most functions, the increased flow capacity of CCP 1-3 is beneficial. However, for certain design basis accidents and other events, increased charging flow is potentially adverse. The higher flow rates of which the new pump is capable may increase the demands on manual operator action or increase likelihood of consequences in which higher flows are adverse. Evaluations demonstrate that plant remains within its design and licensing basis following the modification.

2. The new CCP 1-3 will be driven by a 600-horsepower (hp) induction motor, which is an increase of 400-hp over the existing PDP motor. The increase in connected load to the 4kv vital bus G was evaluated and determined to be acceptable.

3. Additional operator action in a location outside the control room is required to configure the CVCS for LTOP operation. This constitutes an adverse effect on the control of the LTOP design function. The additional operator action outside the control room was evaluated and determined to be acceptable.

The following effects have been evaluated and have been determined not to adversely affect any design functions:

- The potential for pressurized thermal shock events will be mitigated by specifying valve alignment and pump tagouts when RCS temperature and pressure are reduced below the LTOP enable setpoint.

- Noncondensable gas coming out of solution has been a problem for CVCS PDPs, during hydraulic transients in which the volume control tank (VCT) is suddenly depressurized. A multi-stage restricting orifice (RO) similar to the one to be installed with this modification was specifically designed to address dissolved gas issues and avoid flow accelerated corrosion. Operating experience with this RO design has been favorable. The new orifice provided with CCP 1-3 is tested by the pump supplier to assure no unacceptable gas evolution.

- Currently, both centrifugal charging pump min-flow recirculation paths join together to a common header prior to entering the seal water heat exchanger. The new CCP 1-3 mini-flow will also join into this common header. There is no operating condition that will result in all three charging pumps running in mini-flow recirculation, while dead-headed. Thus, there is no adverse effect.

- The heat loads on the room heating ventilation and air conditioning (HVAC) during normal operation will increase as a result of the higher hp drawn by the CCP, and due to the fact that the new pump and motor bearings are air-cooled. Accommodating this increased load will be accomplished within the design capability of the system. Thus, there is no adverse effect.

In summary, no accident precursors result from this change, operator actions regarding LTOP can be anticipated since associated cooldown rates provide time, there is no significant increase in the probability of vital bus overload or LTOP enable delays; and there is no significant increase in equipment malfunction or resulting consequences. Therefore, the proposed changes did not require prior NRC approval.

06-013 Replace Reactor Makeup Control System (Unit 1)

Reference Document No.:	DCP J-49779 Rev. 0
Reference Document Title:	Replace Reactor Makeup Control System

Activity Description:

This design change replaces the existing reactor makeup control system (RMCS). The RMCS provides makeup control functions for maintaining fluid inventory in the VCT and adjusting reactor coolant boron concentration for reactivity control.

The existing boric acid control valve (FCV-110A) and primary water control valve (FCV-111A) will be replaced with new Control Component Inc. control valves. The new control valves are designed to improve low flow control and accuracy while maintaining the existing upper flow range (C_v) requirements for normal batching and emergency boration.

The design change also replaces an obsolete primary water flowmeter. The new flowmeter is designed to improve the accuracy and rangeability of the flow loop. The new primary water flowmeter can be used as performance monitoring equipment (PME). The existing boric acid flowmeter (FIT-110) and new primary water flowmeter (FIT-111) provide analog input to the new digital controller. The analog output signal from these flowmeters to their respective analog flow indicators and recorder is maintained.

This design change does not prevent manual operation of the reactor makeup control valves, the boric acid pumps, or the primary water pumps from their respective handswitches.

Summary of Evaluation:
Since the critical function performed by the RMCS is important for reactivity control, this design change is treated as a digital upgrade and guidance from EPRI TR-102348, Revision 1, is followed. There is no adverse impact to any RMCS design function described in the FSARU. The new digital control system will enhance the ability for operations to initiate a batch makeup and to monitor the process. The new control system has improved automatic features to reduce batch errors and provide better low flow control. The new control system consolidates needed information onto a single display that provides much more effective view of system operation. The system provides an automatic mathematical calculation for flow controller blending. The new system will save time for an operator to seek meter readings or indications. Cautions and warnings can be displayed to prevent errors. The new system reduces the amount of manual valve manipulations to prevent out of sequences control valves. The new control system maintains the boric acid and primary water flow deviation setpoint alarms. The new control features will not adversely increase time required to perform the control action from the touch screen display. 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. Adequate training of the new control system will be provided to the operators. In general, the new system improves batch makeup control and provides human factor enhancement to better support operator tasks and reduce risk of errors. Therefore, the proposed changes did not require prior NRC approval.

06-014	One Time Increase in ECG 8.1 Completion Time (Unit Common)

Reference Document No.:	Equipment Control Guideline (ECG) 8.1, Rev. 5
Reference Document Title:	Positive Displacement Pump (PDP)

Activity Description:

Proposed Revision 5 to ECG 8.1 includes a one time extension, from 14 days to 28 days, to the completion time for an inoperable PDP. This change screened in as an adverse change, requiring a 10 CFR 50.59 evaluation.

FSARU Section 9.5H contains a summary of the ECG 8.1 bases stating, "This ECG ensures the Units 1 and 2 Positive Displacment Pumps (PDP) are available to pump at least 55 gpm at 5800 feet (2550 psid) of pump head to the RCS during plant conditions if the CCPs were to become inoperable due to fire in the CCP area."

Summary of Evaluation:

The proposed change did not require prior NRC approval. The only 10 CFR 50.59 criterion that is impacted by the proposed change is Criterion No. 2, "Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC [system, structure, and component] important to safety previously evaluated in the FSARU?"

The answer is "No". The PDP and CCPs are SSCs important to safety. The Unit 1 and 2 PDPs are required to be available to provide charging flow to the RCS if the CCPs become inoperable due to fire in the CCP area. When a PDP is inoperable, ECG 8.1 minimizes the risk of a fire in the CCP area by establishing a continuous fire watch in the CCP area or establishing a one-hour fire watch and verifying detection and suppression equipment is operable. A probabilistic risk assessment has been performed that shows a one-time increase in completion time from 14 days to 28 days is risk insignificant. The increase in the likelihood of a malfunction of the CCPs due to the unavailability of the PDP is negligible, and does not rise to the "minimal" level. The remaining seven 10 CFR 50.59 criteria are not impacted by the change.

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

07-001 Replace Reactor Makeup Control System (Unit 2)

Reference Document No.:	DCP J-50779, Rev. 1
Reference Document Title:	Replace Reactor Makeup Control System

Activity Description/Summary of Evaluation:

This change is similar to Unit 1 LBIE No. 06-013, "Replace Reactor Makeup Control System (Unit 1)."

07-002 Implement Main Electrical Generator Seal Oil Temperature Controls (Unit 2)

Reference Document No.:	DCP J-050835, Rev. 0
Reference Document Title:	Unit 2 Seal Oil Temperature Controls

Activity Description:

The temperature of the air-side and hydrogen-side generator seal oil are controlled by manual throttling valves on the service cooling water (SCW) discharges of each cooler. Operators use local temperature indicating switch readouts to maintain temperatures at the set point.

This project will automate the temperature control of the air and hydrogen side seal oil so that operator intervention is not normally required. All new equipment is Class II.

Automatic air to close/fail open control valves with positioners will be installed in the SCW discharges of the air-side and hydrogen-side seal oil coolers. Manual bypass valves with the same flow characteristics as the control valves will be installed in parallel to the control valve. Separate, digital, single loop controllers for the air-side and hydrogen-side and a common temperature recorder will be installed in a panel at the seal oil skid. In each seal oil cooler discharge line, the existing single element thermocouple used for local indication/high temperature alarming and the single element thermocouple used for input to the plant process computer (PPC) will be replaced with new, dual element thermocouples. The existing indicating temperature switches will be removed. Thermocouple elements for the air-side and hydrogen-side will be connected to the PPC, the recorder, and the controllers. Each controller will have dual inputs. The recorder will provide an alarm to the annunciator for high or low temperature on either the air or hydrogen side, high differential temperature between the air and hydrogen side, or loss of control power. The controllers will provide an alarm to the annunciator for thermocouple failure, reject to manual, or loss of power. Each controller will output a control signal to its respective control valve. The controllers will provide automatic control of each side's seal oil temperature to an adjustable set point. The controllers can also be operated in manual.

Summary of Evaluation:

The evaluation concluded that there is no adverse impact to the seal oil design function described in the FSARU including any potential new failure mode that might increase the frequency and consequences of any accident previously evaluated in the FSARU. The evaluation also concluded that the new seal oil control would not create a possibility of an accident of different type nor a malfunction of an SSC (main generator) important to safety with a different result than previously evaluated in the FSARU. The main generator is not part of the fission product barrier and the new seal oil control would not alter or exceed any fission product barrier design basis limit as described in the FSARU. This proposed activity is a physical change to a plant SSC (Main Generator) and does not impact any evaluation methodology described in the FSARU.

Because the control equipment to be installed is digital, additional concerns per the guidance in EPRI TR-102348, "Guideline on Licensing Digital Upgrades," were appropriately addressed.

The evaluation concluded that this modification did not require prior NRC approval.

07-003 Replace the Positive Displacement Pump with a New Centrifugal Charging Pump (Unit 2)

Reference Document No.:	DCP M-50704, Rev. 1
Reference Document Title:	Replace the Positive Displacement Pump with a Centrifugal Charging Pump

Activity Description/Summary of Evaluation:

This change is similar to Unit 1 LBIE No. 06-012, "Replace the Positive Displacement Pump with a Centrifugal Charging Pump (Unit 1)."

07-004	**Main Turbine Moisture Separator Reheater Temperature Control Replacement (Unit 2)**

Reference Document No.:	DCP J-50813, Rev. 0
Reference Document Title:	Replace MSR Temperature Controller Unit 2

Activity Description/Summary of Evaluation:

This change is similar to LBIE No. 06-011, "Main Turbine Moisture Separator Reheater Temperature Control Replacement (Unit 1)."

07-013	**Replacement Steam Generator Component Modification (Unit 2)**

Reference Document No.:	DCP M-050790, Rev. 0
Reference Document Title:	RSG Component Modification

Activity Description:

DCPP will replace the original Westinghouse Model 51 steam generators (OSG) with Westinghouse Delta 54 steam generators (RSGs). A comprehensive steam generator replacement program (SGRP) has been conducted to assess the impacts of steam generator replacement. SGRP evaluates RSG operations at the current power level and design conditions. DCP M-050790 assesses compliance with specified design requirements; discusses evaluations, analyses and results; and determines that the RSG is suitable as a replacement for the OSG for plant operation at the current power level and design conditions.

DCP M-050790 includes a description of the functional impacts to interfacing and affected systems, structures, and components due to RSG design and physical/functional differences; a description of the effects on FSARU design transients; and the required changes to design and license basis documents.

Summary of Evaluation:

The RSGs required changes to the TSs. Therefore, a License Amendment Request (LAR) was submitted January 11, 2007 (PG&E Letter DCL-07-002, LAR 07-01), and the NRC issued License Amendments No. 198 and 199 on January 8, 2008. The RSGs did not require additional NRC approvals.

The overall dimensions of the RSGs are approximately the same as the OSG, and the major nozzles are located in the same locations. The 10 CFR 50.59 review considers the mechanical effects of the RSGs, changes in the RSG design features, and the effect of changes in the weight of the RSGs and water volume on the piping and support structural analysis. The design of the RSGs includes requirements and features to address design issues resulting from operating experiences. This 10 CFR 50.59 review is based on the determination by the DCP that the RSGs will satisfy design requirements.

The RSGs require a 10 CFR 50.59 evaluation based on the use of alternative codes or methodologies and changes to elements of certain methodologies as described in the FSARU. The version of the alternative codes, GOTHIC and RETRAN-02W, are NRC-approved for generic use and are applicable to DCPP. The changes to the FSARU-described codes have been NRC-approved for other sites and are shown to be applicable to DCPP. Therefore, the alternative codes and methodology changes do not constitute a departure in methodology as defined by 10 CFR 50.59.

The evaluation addresses an increase in consequences (Control Room [CR] dose) for the steam generator tube rupture (SGTR) accident due to a correction to the analysis of record (unrelated to the RSGs). The increase in dose is less than 10 percent of the difference between the values reported in the FSARU and the regulatory guideline value. In addition the increase does not exceed the current Standard Review Plan guideline value. Therefore, the increases in SGTR CR dose do not result in more than a minimal increase with respect to the OSG SGTR CR doses and remain within the allowable guideline values. Doses associated with the RSGs are the same as the corrected values.

The design basis limits for fission product barriers as described in the FSARU are not exceeded or altered. Additionally, since the RSGs are similar to the OSGs and satisfy FSARU described design functions, there are no adverse effects on the frequency or type of accidents and malfunctions considered in the FSARU.

The Emergency Plan (EP) is changed because the SG Water Level Wide Range values for indicating loss of heat sink are different due to design/analysis differences for the RSG. However, the methods and bases for determining these values are the same as for the OSG and are conservative. In addition, values in implementing procedures for indicating SG conditions during accident conditions are revised to account for RSG design differences using the same methods and bases. There is no decrease in effectiveness of the EP, and the change will be reported to the NRC in accordance with 10 CFR 50.54(q).

Therefore, the change did not require any additional NRC approvals.

07-016 Revision to FSAR 9.1.4.2.3, "Refueling Procedure" (Unit Common)

Reference Document No.:	FSAR Section 9.1.4.2.3
Reference Document Title:	FSAR Update

Activity Description:

FSARU Section 9.1.4.2.3, "Refueling Procedure," is revised to remove the requirement to have the containment equipment hatch closed, one air lock closed, and each penetration providing direct access from the containment atmosphere being closed or capable of being closed by an operable automatic containment purge and exhaust valve prior to moving the reactor head. The FSARU Section 9.1.4.2.3 containment penetration requirements prior to moving the upper internals over fuel are also revised to require capability to close the equipment hatch, airlock, and penetrations providing direct access from the containment atmosphere to the outside atmosphere instead of the equipment hatch being closed and held in place by a minimum of 4 bolts, a minimum of one door in each air lock being closed, and the penetrations being closed.

Summary of Evaluation:

The change described above does not result in positive responses to any of the 10 CFR 50.59 questions. Factors considered in the evaluation included:

- The containment equipment hatch, air locks, and penetrations are components which mitigate the consequences of an accident and they do not initiate accidents.
- The containment equipment hatch and air locks are assumed to be open during fuel handling operations and they are not credited to prevent leakage of radioactivity outside containment for the FSAR Section 15.4.5 fuel handling accident.
- The failure of fuel due to the drop of the reactor head or upper internals is not considered to be realistic and is not analyzed in conjunction with the fuel handling accident.
- There are no credible accidents analyzed in the FSAR which can pressurize containment during movement of fuel in containment.

Therefore, the change did not require prior NRC approval.

07-017	**Implement Main Electrical Generator Seal Oil Temperature Controls (Unit 1)**

Reference Document No.:	DCP J-049835, Rev. 0
Reference Document Title:	U1 Seal Oil Temp Controls

Activity Description/Summary of Evaluation:

This change is similar to LBIE 07-002, "Implement Main Electrical Generator Seal Oil Temperature Controls (Unit 2)."