UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD.
ARLINGTON, TX 76011-4511

June 7, 2016

Mr. Edward D. Halpin, Senior Vice President,
Generation and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNITS 1 AND 2 - NRC EXAMINATION REPORT 05000275/2016301; 05000323/2016301

Dear Mr. Halpin:

On April 23, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an initial operator license examination at the Diablo Canyon Power Plant. The enclosed report documents the examination results and licensing decisions. The preliminary examination results were discussed on April 22, 2016, with Ms. P. Gerfen, Station Director, and other members of your staff. A telephonic meeting was conducted on May 9, 2016, with Mr. R. Fortier, Examination Developer, who was provided with the NRC licensing decisions. A telephonic exit meeting was conducted on May 11, 2016, with Ms. P. Gerfen, Station Director.

The examination included the evaluation of 6 applicants for reactor operator licenses, 5 applicants for instant senior reactor operator licenses, and 2 applicants for upgrade senior reactor operator licenses. The license examiners determined that 12 of the 13 applicants satisfied the requirements of 10 CFR Part 55 and the appropriate licenses have been issued. There was one post-examination comment submitted by your staff. Enclosure 1 contains details of this report and Enclosure 2 summarizes post-examination comment resolution.

Additionally, the NRC identified one finding involving the Diablo Canyon Emergency Operating Procedure, EOP E-2, “Faulted Steam Generator Isolation.” The finding was evaluated under the risk significance determination process as having very low safety significance (Green). Because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy. If you contest these violations or the significance of the non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 1600 E. Lamar Blvd., Arlington, TX 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Senior Resident Inspector at the Diablo Canyon Power Plant. In addition, if you disagree with the cross-cutting aspect assigned

to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Senior Resident Inspector at the Diablo Canyon Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice and Procedure," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Vincent G. Gaddy, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-275 and 50-323
License Nos. DPR-80 and DPR-82

Enclosures:
1. Examination Report 05000275/2016301; 05000323/2016301 w/Attachment: Supplemental Information
2. NRC Review of Diablo Canyon Power Plant Written Post-Examination Comments

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to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Senior Resident Inspector at the Diablo Canyon Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice and Procedure," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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ADAMS ACCESSION NUMBER: ML16159A330

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OFFICE	OE:OB	OE:OB	OE:OB	SOE:OB	SOE:OB	SRA:PSB2	C/PBA
NAME	CCowdrey	MBloodgood	MKennard	TFarina	BLarson	RDeese	JGroom
SIGNATURE	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/
DATE	5/31/16	6/7/16	5/31/16	5/31/16	5/31/16	6/1/16	6/1/16
OFFICE	C:OB						
NAME	VGaddy						
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DATE	6/7/16						

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Letter to Edward D. Halpin from Vincent G. Gaddy, dated June 7, 2016

SUBJECT: DIABLO CANYON POWER PLANT, UNITS 1 AND 2 - NRC EXAMINATION REPORT 05000275/2016301; 05000323/2016301

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	05000275; 05000323
Licenses:	DPR-80; DPR-82
Reports:	05000275/2016301; 05000323/2016301
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, CA
Dates:	March 21 through May 11, 2016
Inspectors:	C. Cowdrey, Chief Examiner, Operations Engineer B. Larson, Senior Operations Engineer T. Farina, Senior Operations Engineer M. Bloodgood, Operations Engineer M. Kennard, Operations Engineer
Senior Reactor Analyst:	R. Deese, Senior Reactor Analyst
Approved By:	Vincent G. Gaddy Chief, Operations Branch Division of Reactor Safety

Enclosure 1

SUMMARY

ER 05000275/2016301; 05000323/2016301; 03/21/2016 – 05/11/2016; Diablo Canyon Power Plant, Units 1 and 2; Initial Operator Licensing Examination Report.

The NRC examiners evaluated the competency of six applicants for reactor operator licenses, five applicants for instant senior reactor operator licenses, and two applicants for upgrade senior reactor operator licenses at Diablo Canyon Power Plant, Units 1 and 2. One applicant for an upgrade senior reactor operator license withdrew from the examination after completing the written examination and two job performance measures (JPMs) as part of the operating test,

The licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 10. The written examination was administered by the licensee on April 15, 2016. The NRC examiners administered the operating tests on April 18 – April 23, 2016.

The examiners determined that 12 of the 13 applicants satisfied the requirements of 10 CFR Part 55 and the appropriate licenses have been issued.

A. <u>NRC-Identified Finding</u>

Cornerstone: Barrier Integrity

<u>Green</u>. The examiners identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, Procedure EOP E-2, "Faulted Steam Generator Isolation," does not contain sufficient procedural direction for isolating auxiliary feedwater flow to a faulted steam generator in the event that auxiliary feedwater control valves cannot be closed from the control room. Procedure EOP E-2, Appendix HH, "Isolated Faulted Steam Generator," Step 1.d, and its associated column, Response Not Obtained, does not ensure that a faulted steam generator would remain isolated under all conditions. The Response Not Obtained column permits operators to either locally close auxiliary feedwater control valves <u>OR</u> secure the auxiliary feedwater pump feeding the faulted steam generator. However, due to the absence of pull-to-lock or hard stop switches for the auxiliary feedwater pumps, the possibility exists for an automatic restart of an auxiliary feedwater pump and a re-initiation of feedwater to a faulted steam generator.

The failure to ensure that Procedure EOP E-2 contained sufficient direction to isolate a faulted steam generator when auxiliary feedwater flow control valves cannot be closed from the control room was a performance deficiency. This performance deficiency was of more than minor safety significance because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone (reactor coolant system and containment) and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the re-initiation of feedwater to an isolated, faulted steam generator has the potential to adversely affect the reactor coolant system barrier by causing an additional unintended cooldown of the reactor coolant system, increased potential for pressurized thermal shock, and thermal stress to the steam

generator u-tubes. Additionally, the containment barrier would be affected by the re-initiation of feedwater to a steam line break within containment. Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, the team determined that the finding required a detailed risk evaluation due to the potential to affect the reactor coolant system boundary. A senior reactor analyst performed a bounding detailed risk evaluation and estimated the maximum increase in core damage frequency to be 5.9E-8/year, and therefore the finding was determined to be of very low safety significance (Green). This increase in core damage frequency was mitigated by the low probability of multiple equipment failures in the auxiliary feedwater system when combined with the low initiating event frequency of a faulted steam generator. Because the violation was of very low safety significance (Green) and the issue was entered into the licensee’s corrective action program as Notification 50847218, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000275/2016301; 05000323/2016301-01, “Insufficient Procedural Direction Contained Within E-2, Faulted Steam Generator Isolation.” This finding has a cross-cutting aspect in the area of human performance associated with resources because the organization did not ensure procedures are available and adequate to support nuclear safety (H.1). (Section 4OA5)

B. <u>Licensee-Identified Violations</u>

None

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

The NRC examiners reviewed all license applications submitted to ensure each applicant satisfied relevant license eligibility requirements. Examiners also audited three of the license applications in detail to confirm that they accurately reflected the subject applicant's qualifications. This audit focused on the applicant's experience and on-the-job training, including control manipulations that provided significant reactivity changes.

b. Findings

No findings were identified.

.2 Examination Development

a. Scope

NRC examiners reviewed integrated examination outlines and draft examinations submitted by the licensee against the requirements of NUREG-1021. The NRC examination team conducted an on-site validation of the operating tests.

b. Findings

One finding was identified.

Insufficient Procedural Direction Contained Within Procedure EOP E-2, "Faulted Steam Generator Isolation"

Introduction. The examiners identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, Procedure EOP E-2 does not contain sufficient procedural direction for isolating auxiliary feedwater flow to a faulted steam generator in the event that auxiliary feedwater control valves cannot be closed from the control room.

Description. Step 1 of Procedure EOP E-2, Appendix HH, provides direction for operators to isolate a faulted steam generator. Step 1.d requires that turbine-driven and motor-driven auxiliary feedwater control valves be closed. In the Response Not Obtained (RNO) step for Step 1.d, the procedure directs operators to locally close the auxiliary feedwater control valves OR secure the auxiliary feedwater pump feeding any faulted steam generators. Diablo Canyon Power Plant is unique in that the auxiliary

feedwater pumps do not have pull-to-lock or hard-stop switches. Therefore, a pump that is stopped to secure feed to a faulted generator has the potential to restart on a number of automatic signals. These signals include an anticipated transient without scram mitigating system actuation circuitry signal, low-low steam generator level, 12kV bus undervoltage, transfer to diesel generator, and a safety injection signal. Should one of these signals be received, a pump that was secured in an effort to stop feeding a faulted steam generator would restart and reinitiate auxiliary feedwater to the faulted generator. The Westinghouse emergency operating procedure background document for Procedure EOP E-2 describes the purpose of isolating a faulted steam generator. Specifically, the step is intended to minimize the cooldown of the reactor coolant system and limit the mass and energy released into containment for a steamline break inside containment. Therefore, the re-initiation of auxiliary feedwater to a faulted, and potentially dry steam generator, has the potential to reinitiate another cooldown cycle that presents an additional pressurized thermal shock concern. Further, a pump restart would add additional mass and energy into containment should the steam generator be faulted inside containment. Finally, the addition of cold auxiliary feedwater to a dry steam generator has the potential to cause thermal shock to the steam generator u-tubes and increase the risk of a concurrent faulted and ruptured steam generator. The Westinghouse background document for Procedure EOP FR-H.1, "Response to Loss of Secondary Heat Sink," states that care should be taken when re-initiating feedwater to a dry steam generator. If while isolating a faulted steam generator, operators decide to stop an auxiliary feedwater pump and not close the flow control valves manually, as is permitted by the Step 1.d RNO of Appendix HH, full auxiliary feedwater pump flow could potentially be directed into a dry steam generator upon auxiliary feedwater pump restart.

Procedure EOP E-3 for a steam generator tube rupture recognizes the potential for auxiliary feedwater pumps to restart after isolating a ruptured generator. Step 4.b RNO provides a note stating that an "interlock device may be used." These interlock devices, called "hutch interlocks" at Diablo Canyon are physical devices that are placed on a control room switch to keep it in the OFF position. Additionally, Procedure EOP E-3 states that <u>IF</u> valves cannot be closed, <u>THEN</u> stop the auxiliary feedwater pump that is feeding the ruptured generator <u>UNTIL</u> manual valves can be shut. Therefore, Procedure EOP E-3 requires that operators eventually close the flow control valves, not simply stop the auxiliary feedwater pump. The guidance in Procedure EOP E-2 does not match the direction contained in EOP E-3 for isolating feedwater to an affected steam generator.

<u>Analysis</u>. The failure to ensure that Procedure EOP E-2 contained sufficient direction to isolate a faulted steam generator was a performance deficiency. This performance deficiency was of more than minor safety significance because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone (reactor coolant system and containment) and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the re-initiation of feedwater to an isolated, faulted steam generator has the potential to adversely affect the reactor coolant system barrier by causing an additional unintended cooldown of the reactor coolant system, increased potential for pressurized thermal shock, and thermal stress to

the steam generator u-tubes. Additionally, the containment barrier would be affected by the re-initiation of feedwater to a steam line break within containment.

Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the team determined that the finding required a detailed risk evaluation due to the potential to affect the reactor coolant system boundary. A senior reactor analyst performed the detailed risk evaluation by estimating the maximum core damage frequency which could result from the performance deficiency. For this estimate, the analyst determined the frequency of the conditions which would place the plant in an operational state where the auxiliary feedwater pumps would be secured and have the potential to restart and introduce water into a steam generator.

First, the analyst assumed the frequency of the initiating event to be that of a steam line break upstream of the main steam isolation valves. Feed breaks were not considered because auxiliary feed would be feeding the break location since the main and auxiliary feedwater systems share a common line. The analyst assumed this frequency to be 3.67E-4/year which was derived from Revision 8.23 of the Diablo Canyon standardized plant analysis risk (SPAR) model.

For the conditions of the performance deficiency to exist, after the steam generator faulted, the auxiliary feedwater control valves to the faulted steam generator would have to fail to close. The analyst assumed only one of these valves would have to fail to close and used the generic failure probability from the 2010 data update for NUREG/CR-6928 for a hydraulically operated valve failing to close. This probability was 1.20E-3.

When the auxiliary feedwater control valve to the faulted steam generator failed to close, the RNO column gave the operators two options – to locally close the control valve(s) to the faulted steam generator or to secure the auxiliary feedwater pump feeding the faulted steam generator. In assuming only one pump was feeding the faulted steam generator, operators would only have to close one valve locally in the plant. In performing the human reliability analysis for this basic event in SPAR-H, the analyst assumed that diagnosis and action were needed, that both were affected by high stress, and that operators had barely enough time. These assumptions resulted in the valve of the human reliability analysis basic event for locally closing one hydraulically operated valve to be 2.2E-1. On assuming that two or three auxiliary feedwater pumps were feeding the faulted steam generator, operators would have to close two or three valves locally. The analyst assumed complete dependence on closure of the second and third valves because the same crew would be performing it, at the same time, in the same location, and with no additional cues. This drove the human reliability basic event valve to 2.2E-1 for multiple local valve closures, or the same as for only one valve.

The second option in the RNO column was for operators to secure the auxiliary feedwater pump feeding the faulted steam generator. The analyst assumed that diagnosis and action were needed, that both were affected by high stress, and that operators had barely enough time. These assumptions resulted in the valve of the

human reliability analysis basic event for securing an auxiliary feedwater pump to the faulted steam generator being 2.2E-1. The analyst assumed high dependence on securing an auxiliary feedwater pump because the same crew would be performing it, at the same time, in a different location, and with no additional cues. This drove the human reliability basic event to 6.1E-1.

The analyst assumed that operators would try to perform one of the actions in the RNO column and upon failure would try the other action. As a result, both failure to shut valve(s) and secure an auxiliary feedwater pump would have to occur for the procedural conditions for the performance deficiency to occur. Therefore to obtain the frequency for the plant state for which the performance deficiency would occur, the analyst multiplied the following together:

- The faulted steam generator initiating event frequency (3.67E-4/year)
- The probability of the auxiliary feedwater control valves to close (1.2E-3)
- The probability of operators failing to close the auxiliary feedwater control valves locally, regardless of the number of valves needed to close (2.2E-1)
- The probability of operators failing to secure the auxiliary feedwater pump feeding the faulted steam generator (6.1E-1)

This produced a frequency for the plant state which would produce the performance deficiency of 5.9E-8/year. The maximum increase in core damage frequency would be 5.9E-8/year, regardless of what mitigating equipment was unavailable and therefore the finding was determined to be of very low safety significance (Green).

This finding has a cross-cutting aspect in the area of human performance associated with resources because the organization did not ensure procedures are available and adequate to support nuclear safety [H.1].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” states, in part, that “Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.”

Contrary to the above, prior to April 23, 2016, the licensee did not ensure that Procedure EOP E-2, “Faulted Steam Generator isolation,” contained appropriate quantitative or qualitative acceptance criteria for determining that important activities had been satisfactorily accomplished. Specifically, the licensee did not ensure that Procedure E-2 contained the necessary direction for isolating a faulted steam generator should the auxiliary feedwater flow control valves not close from the control room. To correct this issue, the licensee wrote Notification 50847218.

Because the violation was of very low safety significance (Green) and the issue was entered into the licensee’s corrective action program as Notification 50847218, this

violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000275/2016301; 05000323/2016301-01, “Insufficient Procedural Direction Contained Within EOP E-2, Faulted Steam Generator Isolation.”

c. Other Observations

The NRC examiners provided outline, draft examination, and post-validation comments to the licensee. The licensee satisfactorily completed comment resolution prior to examination administration.

The NRC examiners determined the reactor operator written examination initially submitted by the licensee was not within the range of acceptability expected for a proposed examination. The NUREG-1021 standard for an acceptable submittal requires that 20 percent or fewer of the written examination questions must be classified as unsatisfactory based on criteria in Section ES-401; this criterion applies for the reactor operator portion, the senior reactor operator portion, or both. The statistics for the written examination were as follows:

Reactor Operator (RO) written examination (75 total questions)
1. Sixteen questions were unsatisfactory (21 percent)
2. Thirty-six questions required editorial changes (48 percent)

Senior Reactor Operator (SRO) written examination (25 total questions)
1. Four questions were unsatisfactory (16 percent)
2. Fifteen questions required editorial changes (60 percent)

Total written examinations (100 total questions)
1. Twenty questions were unsatisfactory (20 percent)
2. Fifty-one questions required editorial changes (51 percent)

Because the 20 percent threshold was exceeded for the reactor operator section of the written examination, it was classified as an unsatisfactory submittal. Also, based on the number of unsatisfactory questions and questions requiring additional enhancement, the written examinations required substantial work by the NRC Chief Examiner and Diablo Canyon examination writing staff in order to prepare the written examination for administration. The licensee wrote Notification 50847217 to address the issues with the unsatisfactory reactor operator written examination submittal. As required in NUREG-1021 for an unsatisfactory submittal, “all future submittals should incorporate any lessons learned from this effort.”

The NRC examiners determined the operating tests initially submitted by the licensee were within the range of acceptability expected for a proposed examination.

.3 Operator Knowledge and Performance

a. Scope

On April 15, 2016, the licensee proctored the administration of the written examinations to all 13 applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis to the NRC on May 5, 2016.

The NRC examination team administered the various portions of the operating tests to all applicants on April 18 – April 23, 2016. One applicant withdrew from the remainder of the operating test after completing two JPMs and no simulator scenarios.

b. Findings

No findings were identified.

All 13 applicants passed the written examination and 12 of the 13 applicants passed all parts of the operating test. The final written examinations, the operating test, and post-examination analysis and comments may be accessed in the ADAMS system under the accession numbers noted in the attachment to Enclosure 1. The licensee requested and received approval by the NRC to withhold the written examinations from the public document room for 2 years after the administration date.

The examination team noted the following generic weaknesses during the operating tests:

1. When presented with a failed-open pressurizer power-operated relief valve (PORV) that initiated on a reactor trip, two of three crews failed to identify the open PORV prior to receiving a safety injection (SI) actuation. The SI actuation led to the pressurizer being filled and a subsequent rupture of the pressurizer relief tank into containment.

2. When presented with a ruptured steam generator with a failed open 10 percent steam dump valve that had been taken to manual and closed, two of four crews failed to manually control the ruptured steam generator pressure below the steam generator safety valve setpoint.

3. Several applicants failed to inform, or were slow to inform, the shift foreman when they had taken manual action to compensate for a component or system that should have actuated automatically.

4. On a simulator JPM that started in FR-C.1, Response to Inadequate Core Cooling, three applicants, in the absence of any available auxiliary feedwater pumps, chose to fully open all 10 percent steam dump valves. Eventually, the three applicants correctly established temporary core cooling via pressurizer PORVs and reactor vessel head vents.

The licensee wrote Notifications 50847245, 50847247, 50847248, and 50849195 to address the generic weaknesses.

Additionally, the licensee submitted one post-examination comment (Q3) that required review and disposition by the Chief Examiner. The Region IV Operations Branch Chief assigned a panel of examiners that were not part of the examination team effort at Diablo Canyon Power Plant to review the question challenge. The panel reviewed the question and recommended designating 'D' as the only correct answer. The Chief Examiner and Operations Branch Chief concurred with the NRC regional panel's recommendation that 'D' should be considered the only correct answer for Question #3. However, while administering the examination, the licensee provided clarification on the question stem to one applicant which led to the applicant choosing 'B' (the original correct answer). The licensee failed to follow the guidance contained within NUREG-1021, ES-402 D.3.b, which states, "Any question changes or clarifications shall be made on a chalk board or white board, if available, and called to the attention of all the applicants." The licensee wrote Notification 50848688 to address the failure to properly inform all applicants of the stem clarification. Therefore, the NRC has accepted 'B' as the correct answer <u>ONLY</u> for the applicant who received the additional clarification. For all other applicants, 'D' is considered the only correct answer. This grading decision was made in consultation with the NRC Headquarters operator licensing program office within the Office of Nuclear Reactor Regulation. More details are included in Enclosure 2 of this report and the entire licensee's post-examination comments and analysis can be found in ADAMS using Accession Number ML16145A323. Copies of all individual examination reports were sent to the facility training manager for evaluation and determination of appropriate remedial training.

.4 <u>Simulation Facility Performance</u>

a. <u>Scope</u>

The NRC examiners observed simulator performance with regard to plant fidelity during examination validation and administration.

b. <u>Findings</u>

No findings were identified.

.5 <u>Examination Security</u>

a. <u>Scope</u>

The NRC examiners reviewed examination security for examination development during both the on-site preparation week and examination administration week for compliance with 10 CFR 55.49 and NUREG-1021. Plans for simulator security and applicant control were reviewed and discussed with licensee personnel.

b. <u>Findings/Observations</u>

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The Chief Examiner presented the preliminary examination results to Ms. P. Gerfen, Station Director, and other members of your staff on April 22, 2016. A telephonic exit was conducted on May 11, 2016, between Mr. C. Cowdrey, Chief Examiner, and Ms. P. Gerfen, Station Director, and other members of your staff.

The licensee did not identify any information or materials used during the examination as proprietary.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Gerfen, Station Director
D. Gouveia, Manager, Operations
H. Hamzehee, Manager, Regulatory Services
E. Werner, Manager, Operations Training
J. Becerra, Supervisor, Training
J. Morris, Supervisor, Regulatory Services
R. Fortier, Examination Developer
L. Toribio, Instructor, Operations Training
H. Matteson, Operations Procedure Writer

NRC Personnel

J. Reynoso, Senior Resident Inspector
M. Stafford, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000275/2016301; 05000323/2016301-01	NCV	Insufficient Procedural Direction Contained Within Procedure EOP E-2, Faulted Steam Generator Isolation (Section 40A5)

ADAMS DOCUMENTS REFERENCED

Accession No. ML16132A400 - FINAL WRITTEN EXAMS
(delayed release May 31, 2018)

Accession No. ML16145A325 - FINAL OPERATING TEST

Accession No. ML16145A323 - POST EXAMINATION ANALYSIS (AND COMMENTS)

Enclosure 2: NRC Review of Diablo Canyon Power Plant Written Post-Examination Comment

Note: A complete text of the licensee's post-examination analysis and comments can be found in ADAMS under Accession Number ML16145A323.

The NRC Region IV Operations Branch Chief established a panel of two examiners that had no involvement in any part of the examination process for this examination at Diablo Canyon Power Plant to review the one post examination comment submitted by the licensee. The NRC resolution section below is a summary of the panel conclusions.

Question 3

A large break LOCA occurred on Unit 1. The operators have begun aligning for cold leg recirculation.

The operator has just opened 8804B, RHR Heat Exchanger 1-2 Discharge to Charging and SI Pumps Suction valve.

The following valves have been operated prior to opening 8804B:

1. 8974B, SI Pps Recirc Stop valve - Close
2. 8982B, RHR Pp 1-2 Suct From Contmt Recirc Sump - Open
3. 8700B, RHR Pump Suction Isolation – Close

Which of the valves operated were necessary to open 8804B?

A. 1 only

B. 1 and 2 only

C. C. 2 only

D. D. 1, 2, and 3

Original Proposed Answer: B. 1 and 2 only

Revised Proposed Answer: D. 1, 2, and 3

Licensee Comments for Question 3:

As mentioned in NUREG 1021, ES 403, Section D.1.b, sometimes errors are identified after the exam is administered. As described in the first example of this section, the facility believes question #3 is "a question with unclear stem that confused the applicants or did not provide all the necessary information".

The lack of clarity in the stem, leaves room for interpretation by the candidates as to which interlocks are of interest. This is confirmed by a question raised during the exam by one of the candidates. Based on the wording, it is a reasonable assumption to believe the aggregate of all interlocks are to be considered. The stem indicates 3 valves were operated prior to opening 8804B, and asks which of those manipulations were necessary in order to do so. Every valve listed is required to be operated in order to open 8804B. As such, the key answer “B” is clearly incorrect. The facility believes the only correct answer is “D” and requests the key to be revised based to indicate the correct answer as “D”, based on the following.

<u>INTERLOCKS (Reference Lesson Plan LB-2 attached):</u>

- In order to open 8804B, the following valves must be in the indicated alignment:
  - 8701 or 8702 Closed
  - 8974A or **<u>8974B</u>** Closed
  - **<u>8982B</u>** Open
- In order to open 8982B:
  - **<u>8700B</u>** must be closed.

<u>PROCEDURAL FLOW PATH (Reference E-1.3 Transfer to Cold Leg recirculation attached)</u>:

In the situation described, the crew is implementing E-1.3. Prior to manipulating any valves, the expected alignment is 8700B Open, 8982B closed, 8974B Open.

- The first valve operation in the body of E-1.3 is close 8700B (E 1.3 step 3)
- As 8700B is stroking closed, 8974A and 8974B are closed (E 1.3 step 7.b)
- Once 8700B is closed, 8982B is opened (E1.3 step 9.b)
- 8804B is opened afterwards (E1.3 step 9e)

Finally, a test item analysis was performed, which identified this question as high miss. The review indicates 31% chose answer “D”, 38% chose answer “B” and the remaining 31% chose A or C, indicating there was confusion regarding the intent of this question.

NRC Resolution of Question 3

The licensee performed a post-examination analysis and determined that the correct answer for Question #3 should be changed from ‘B’ to ‘D’. NRC Region IV formed a panel of examiners who had not participated in any way in the development or administration of the examination to evaluate the question challenge. The panel developed the following recommended response:

> The main issue is whether the operations needed to open valve 8804B involve only operations associated with the interlock unique to valve 8804B, or if it includes other operations as well. We believe that opening valve 8804B requires its interlock conditions to be satisfied, as well as any conditions necessary to enable the interlock to be met. Valve 8700B being in the closed position is necessary in order for valve 8982B to be opened (an interlock condition).

If valve 8700B is not in the closed position, valve 8982B cannot be opened and, therefore, valve 8804B cannot be opened.

Therefore, the three valves operated, based on the stem information, are all necessary to open valve 8804B. We concur with the recommendation to change the correct answer from B to D.

The Chief Examiner and Region IV Operations Branch Chief concurred with the NRC regional panel's recommendation that 'D' should be considered the only correct answer for Question #3. However, while administering the examination, the licensee provided clarification on the question stem to one applicant which led to the applicant choosing 'B' (the original correct answer). The licensee failed to follow the guidance contained within NUREG-1021, ES-402 D.3.b, which states, "Any question changes or clarifications shall be made on a chalk board or white board, if available, and called to the attention of all the applicants." The licensee wrote Notification 50848688 to address the failure to properly inform all applicants of the stem clarification. Therefore, the NRC will accept 'B' as the correct answer <u>ONLY</u> for the applicant who received the additional clarification. For all other applicants, 'D' will be considered the only correct answer.