UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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July 18, 2007

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SUBJECT: DIABLO CANYON POWER PLANT, UNITS 1 AND 2 - NRC EXAMINATION REPORT 05000275/2007301; 05000323/2007301

Dear Mr. Keenan:

On April 13, 2007, the US Nuclear Regulatory Commission (NRC) completed an examination at Diablo Canyon Power Plant, Units 1 and 2. The enclosed report documents the examination results, which were discussed on May 10, 2007, with Mr. James Becker, Vice President, Operations and Site Director, and other members of your staff via a telephonic call. On June 15, 2007, I conducted a telephonic exit meeting with Mr. Joseph Haynes, Training Manager.

The examination included the evaluation of 6 applicants for reactor operator licenses, 3 applicants for instant senior operator licenses and 2 applicants for upgrade senior operator licenses. The written examination and operating test were developed using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9. It was determined that 6 of the 11 applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

No findings of significance were identified during this examination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," 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 document 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/

Anthony T. Gody, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-275; 50-323
License: DPR-80; DPR-82

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NRC Examination Report 05000275/2007301;
05000323/2007301

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EXAMINATION REPORT

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-275, 50-323
Licenses:	DPR-80, DPR-82
Report:	05000275/2007301; 05000323/2007301
Licensee:	Pacific Gas and Electric Company (PG&E)
Facility:	Diablo Canyon Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, California
Dates:	April 2 through - June 15, 2007
Inspectors:	T. O. McKernon, Chief Examiner, Operations Branch Brian Larson, Operations Engineer Jim Drake, Operations Engineer
Approved By:	Anthony T. Gody, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000275/2007301; 05000323/2007301; 04/02-06/2007; Diablo Canyon Power Plant, Units 1 and 2; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of 6 applicants for reactor operator licenses, 3 applicants for instant senior operator licenses and 2 applicants for upgrade senior operator licenses at Diablo Canyon Power Plant, Units 1 and 2. The facility licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9. NRC examiners administered the operating test on April 2 - 6, 2007. The written examination was administered by the facility on April 13, 2007. The license examiners determined that 6 of the 11 applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

Report Details

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

The examiners reviewed all 11 license applications submitted by the licensee to ensure the applications reflected that each applicant satisfied relevant license eligibility requirements. The applications were submitted on NRC Form 398, "Personal Qualification Statement," and NRC Form 396, "Certification of Medical Examination by Facility Licensee." The 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 of significance were identified.

.2 Operator Knowledge and Performance

a. Examination Scope

The NRC examination team administered the various portions of the operating test to all 11 applicants on April 2 - 6, 2007. The 6 applicants for reactor operator licenses participated in two dynamic simulator scenarios, a control room and facilities walkthrough test consisting of 11 system tasks, and an administrative test consisting of 4 administrative tasks. Two of the 3 applicants seeking an instant senior operator license participated in two dynamic scenarios, a control room and facilities walkthrough test consisting of 10 system tasks, and an administrative test consisting of 5 administrative tasks. One of the 3 applicants seeking an instant senior operator license participated in three dynamic scenarios, a control room and facilities walkthrough test consisting of 10 system tasks, and an administrative test consisting of 5 administrative tasks. The 2 applicants for upgrade senior operator licenses participated in one dynamic simulator scenario, a control room and facilities walkthrough test consisting of 5 system tasks, and an administrative test consisting of 5 administrative tasks.

On April 13, 2007, the licensee proctored the administration of the written examinations to all 11 applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis to the NRC on April 23, 2007.

b. Findings

Six of the 11 applicants passed all parts of the initial licensing examination. Three applicants (1 reactor operator and 2 senior operators) failed the written portion of the examination. Four of the applicants (1 reactor operator and 3 senior reactor operators) failed the operating test portion of the examination. For the written examinations, the reactor operator applicants' average passing score was 85.7 percent and ranged from 81.1 to 93.2 percent; the senior operator applicants' average passing score was 89.9 percent and ranged from 85.9 to 91.9 percent.

Section ES-403 and Form ES-403-1 of NUREG 1021 require the licensee to analyze the validity of any written examination questions that were missed by half or more of the applicants. The licensee conducted this performance analysis for 10 questions that met this criteria and concluded that remediation was necessary on those questions. The examiners agreed with the licensee's conclusions.

Written Examination

The licensee submitted six examination questions for consideration of a change to the answer key for those questions. The licensee provided comments as well as plant reference material to support their submission. The license examiners reviewed the comments and material provided to determine if any changes to the examination answer key were justified. For the six questions submitted, the licensee recommended accepting the original and a second answer as correct. The examiners accepted the recommendation for one question, deleted one question, changed the correct answer from C to B on one question, and did not change the remaining three questions. Provided below are the six questions, a summary of the licensee's recommendation and justification, and the examiner's resolution and justification.

QUESTION # 7

GIVEN:
I. Unit 1 is at 100% power
II. One CCW heat exchanger is in service
III. Containment temperature is 80°F
IV. Spent Fuel Pool temperature is 90°F

CCW heat exchanger outlet temperature is 85°F and increasing. The crew is going to reduce CCW loads in accordance with OP AP-11, "Malfunction of Component Cooling Water System."

Which of the following CCW heat loads could the operators isolate to lower CCW temperature while the unit is at power?

A. SFP Heat Exchanger
B. RCP thermal barriers
C. Seal Water Heat Exchanger
D. Containment Fan Cooling Units

Answer:

C. Seal Water Heat Exchanger

Licensee Comment

The licensee recommended accepting both choices A and C as correct answers. This recommendation was based on the facts that the question, as written, was too high a level of difficulty for recall knowledge and that removal of the spent fuel pool heat exchanger would also lower the component cooling water system temperature. The facility also noted out that the information was contained in an appendix of Abnormal Procedure AP-11 that wasn't required from memory knowledge.

NRC Resolution

The licensee's recommendation was not accepted, however, the examiners agreed that the question, as written, was not appropriate for recall level of knowledge. Therefore, the question was deleted from the NRC examination. The question required the applicants to recall a step from the appendix of an abnormal procedure. Since this action was neither an immediate operator action nor a frequently performed action, recall from memory was not required.

QUESTION # 10

The crew is performing a cool down and depressurization in accordance with E-0.3, “Natural Circulation Cool Down With Steam Void in Vessel (With RVLIS).”

Which of the following is the preferred method of depressurizing the Reactor Coolant System?

A. PORV
B. Aux Spray
C. Normal Spray
D. Operating ECCS pumps as necessary

Answer:

B. Aux Spray

Licensee Comment

The licensee recommended accepting both choices A and B as correct answers. This recommendation was based on the fact that the question did not include the status of the reactor coolant letdown system. As such, the facility contended that use of a power-operated relief valve would accomplish the same effect of depressurizing the reactor coolant system as the auxiliary spray system.

NRC Resolution

The licensee's recommendation was not accepted. The examiners noted that the use of a power-operated relief valve to depressurize the reactor coolant system was a contingency action to be implemented only when letdown cannot be established or placed in service as identified in the “Response Not Obtained” column of Procedure E-0.3. Accordingly, the examiners concluded that the auxiliary spray system was the “preferred” method of depressurizing the reactor coolant system. Therefore, B was the only correct answer and the answer key was not changed.

QUESTION # 36

GIVEN:
I. The plant is at 90% power
II. Control Rods are at 228 steps
III. Boron concentration is 1600 ppm

A control valve malfunction results in a step 50 MWe increase in load.

How would the initial plant response be different if the same 50 MWe step increase occurred when boron concentration was 300 ppm?

A. The RCS temperature change would be smaller.
B. More positive reactivity would be added.
C. The power change would be smaller.
D. The power defect would be smaller.

Answer:

A. The RCS temperature change would be smaller.

Licensee Comment

The licensee recommended accepting both choices A and B as correct answers. The recommendation was based on the fact that the Doppler only power coefficient becomes more negative from beginning-of-life (1600ppm Boron) to end-of-life (300ppm Boron). A 50 MWe step increase in load at end-of-life would add more negative reactivity from the Doppler only power coefficient than the same increase in load at beginning-of-life. Therefore, even though the actual reactor coolant system temperature change would be smaller at end-of-life than beginning-of-life for the same increase in load (because of a more negative moderator temperature coefficient at end-of-life than beginning-of-life), it would have to add more positive reactivity to offset the increase in negativity reactivity from the Doppler effect.

NRC Resolution

During initial grading the recommendation was not accepted and the answer key was not changed. However, after licenses were issued, during the development of this examination report, the facility’s recommendation was re-evaluated by the examiners. Based on the re-evaluation, the licensee’s recommendation was accepted. The facility-provided reference material supports that a more negative Doppler only power

coefficient exists at end-of-life as compared to beginning-of-life. Thus, for the same size power increase, more negative reactivity would be added by the Doppler only power coefficient at end-of-life than at beginning-of-life. Since the net reactivity change of a power change is zero, more positive reactivity would be added by a reactor coolant system temperature change at end-of-life than beginning-of-life, even though the magnitude of the temperature change would be smaller (due to the more negative moderator temperature coefficient at end-of-life). As a result of accepting the facility's recommendation, the answer key was changed to accept A and B as correct, and all written examination grades were recalculated. It was determined that there were no changes to any pass/fail results.

QUESTION # 44

GIVEN:
I. Unit 1 has just entered MODE 5, cooling down for refueling
II. Both trains of RHR are in service
III. CCW pump 13 is out of service

One of the two running CCW pumps trip. Due to increasing CCW temperatures the crew is reducing heat loads in accordance with OP AP-SD-4, "Loss of Component Cooling Water."

When loads have been reduced, what will be the status of CCW flow to the RHR heat exchangers?

A. Isolated to both heat exchangers.
B. Reduced to both heat exchangers.
C. Unchanged to both heat exchangers.
D. Reduced to one heat exchanger and unchanged to other.

Answer:

C. Unchanged to both heat exchangers.

Licensee Comment

The licensee recommended accepting both choices B and C as correct answers. This recommendation was based on the fact that since overall component cooling water system flow would be partially reduced upon loss of a running pump, flow to both heat exchangers would be reduced.

NRC Resolution

The licensee's recommendation was not accepted. In accordance with Procedure APSD-4, Step 1.1.3 (applicable for implementation of procedure steps 7-10), the applicant is directed to respond to component cooling water system low flow conditions, such as the trip of one component cooling water pump while two pumps were running. Procedure actions state, "if only one component cooling water pump can be run, then component cooling water flow is reduced to within the capacity of one CCW

pump." In accordance with data and component cooling water flow plots provided by the facility, overall component cooling water system flow will be reduced as a result of losing one running component cooling water pump. Therefore, the answer key was changed to accept B as the only correct answer.

QUESTION # 96

Unit 1 is in MODE 5.

A component required to be OPERABLE in MODES 1 – 4 is removed from service.

In accordance with OP1.DC17, "Control of Equipment Required by the Plant Technical Specifications or Other Designated Programs," how, if at all, is the inoperable equipment tracked?

A. Tracking is not done.
B. Using an Info TS sheet.
C. Using an Active TS sheet.
D. Documenting the inoperability each night in the Shift Manager Logs.

Answer:

A. Tracking is not done.

Licensee Comment

The licensee recommended accepting choices A and B as correct answers. This recommendation was based on the fact that the question was not clear as to the current status of the plant, in that, if the component was removed just prior to Mode 4 entry then an information only technical specification sheet would be used to track the inoperable equipment.

NRC Resolution

The licensees recommendation was not accepted. The stem of the question clearly states that the unit is in Mode 5. No applicant asked a question concerning the intent or the initial conditions as directed by NUREG-1021, Appendix E, Part B.7. Part B.7. also states that when answering a question, applicants are not to make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. Given the information provided in the stem of the question and directions provided by Part B of Appendix E, choice A is the only correct answer and the answer key was not changed.

QUESTION # 99

Unit 1 is at 50 percent power. All systems are in Auto.

The operator reports the following:
I. Pressurizer pressure is 2250 psig and slowly increasing
II. Pressurizer spray valves are closed
III. Letdown Orifice Valves are closed

IV. Charging flow is decreasing
V. Pressurizer Level 54% and increasing
VI. Reactor Cavity Sump levels are 0%

Which of the following abnormal operating procedures would be appropriate to address the current plant conditions?

A. AP-5, Malfunction of Eagle 21 Protection or Control Channel
B. AP-9, Loss of Instrument Air
C. AP-13, Malfunction of Reactor Pressure Control System
D. AP-18, Letdown Line Failure

Answer:

B. AP-9, Loss of Instrument Air

Licensee Comment

The licensee recommended accepting both choices A and B as correct answers. The recommendation was based on the fact that a pressurizer level instrument malfunction could produce the symptoms listed in the stem of the question. Specifically, the licensee stated the following, "A pressurizer level backup control channel failure could produce the symptoms listed in the stem of the question. A failure of backup level control channel will isolate the letdown orifice isolation valves and cause the pressurizer level to increase. Charging flow will decrease to attempt to maintain pressurizer level on program. Pressurizer pressure would increase due to the pressurizer level increase. It is entirely reasonable for a relatively inexperienced candidate to rule out a letdown line failure with 0% indicated level in the reactor cavity sump and not necessarily equate this with a loss of instrument air to containment."

NRC Resolution

The licensee's recommendation was not accepted. System design documents and facility provided references indicate that while a malfunction of the pressurizer level instrument in the low direction would result in similar indications as a loss of instrument air, it would not result in an indicated reactor cavity sump level of 0 percent. Reactor cavity sump level is maintained at a level greater than 0 percent during normal plant operations. An indication of 0 percent could only be explained by the answer choice involving a loss of instrument air to the reactor sump level instrument, therefore, answer B is the only correct answer and the answer key was not changed.

<u>Operating Test</u>

The examiners observed several generic weaknesses in operator performance. Examples included: (1) Weakness in diagnosing malfunctions based upon plant instrumentation and indications, (2) Weakness in understanding integrated plant operations, (3) Hesitancy of operators to take immediate manual actions when appropriate, and (4) Knowledge weakness in applying given conditions and determining emergency action levels.

The first and second weaknesses were observed during a dynamic simulator scenario events that included malfunction of a steam pressure transmitter, a steam dump failure, and an inadvertent start of the turbine driven auxiliary feedwater pump.

The third weakness was observed during a simulated malfunction in which safety injection was required but did not actuate. Additionally, during a dynamic simulator system task, the majority of applicants failed to take the appropriate immediate operator actions for a tripped main feedwater pump.

The fourth weakness was observed during the performance of an administrative task that involved calculation of the total effective dose equivalent and the thyroid committed dose equivalent given plant conditions, meteorological data, and radioactive material release information. The applicants were then required to determine the applicable emergency action level. The majority of applicants either failed to correctly calculate the total effective dose equivalent and committed dose equivalent, or failed to correctly identify the applicable emergency action level.

.3 Initial Licensing Examination Development

a. Examination Scope

The licensee developed the examinations in accordance with NUREG-1021, Revision 9. All licensee facility training and operations staff involved in examination preparation and validation were on a security agreement. The facility licensee submitted both the written examination and operating test outlines on January 9, 2007. Examiners reviewed the outlines against the standards of NUREG-1021 and provided comments to the licensee on January 25, 2007. The facility licensee submitted the draft examination package on January 31, 2007. Examiners reviewed the draft examination package against the standards of NUREG-1021 and provided comments to the licensee on February 8, 2007. The examiners conducted an onsite validation of the operating test and provided further comments during the week of February 12, 2007. The licensee satisfactorily completed comment resolution during the week of March 5, 2007. Also during the week of March 5, 2007, the licensee requested and NRC Region IV approved a delay in administering the written examination and operating test. The written examination date was changed from March 12 to April 13, 2007, and the operating test dates were changed from the weeks of March 13 and 19, 2007, to the week of April 2, 2007. The licensee requested the delay in order to provide the applicants more training as well as balancing simulator availability for outage preparation and the biennial requalification inspection. The duration of the operating test was reduced from two to one week due to a reduction in the number of applicants following the licensee's audit examination.

b. Findings

The NRC approved the initial examination outline with minor comments and advised the licensee to proceed with the examination development.

The examiners determined that the written examinations and operating test initially submitted by the licensee were within the range of acceptability expected for a proposed examination.

No findings of significance were identified.

.4 Simulation Facility Performance

a. Scope

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

b. Findings

No findings of significance were identified.

.5 Examination Security

a. Scope

The examiners reviewed examination security for examination development and during the onsite preparation week and examination administration week for compliance with the standards contained in NUREG-1021. Plans for simulator security and applicant control were reviewed and discussed with licensee personnel.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The examination team presented a debrief to Mr. James Welsch, Operations Manager, and other members of the licensee's management staff at the conclusion of the examinations on April 6, 2007. The licensee acknowledged the findings presented. On May 10, 2007, the examination results were discussed with Mr. James Becker, Vice President, Operations and Site Director, and other staff members via a telephonic call. During this telephonic call, the licensee committed to conduct remedial training for individual reactor operators and senior operators on observed weaknesses prior to their assumption or resumption of licensed operator duties. On June 15, 2007, Mr. Anthony Gody, Chief, Operations Branch, conducted a final telephonic exit meeting with Mr. Joseph Haynes, Training Manager.

The licensee did not identify any information or materials used during the examination as proprietary. Materials of a proprietary nature submitted with the post-examination comments were redacted as necessary.

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

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James Welsch, Operations Manager
Gary Anderson, Assistant Operations Manager
Paul Roller, Operations Director
Joseph Haynes, Training Manager
David Burns, Operations Training Supervisor
Larry Parker, Regulatory Services
Steve Wilson, Initial License Class Supervisor
Michael Kennedy, Operations Liaison to Shift Manager
John Buckley, Operations Instructor
Jack Blackwell, Operations Instructor
Ron Fortier, Operations Instructor

NRC Personnel

T. Jackson, Senior Resident Inspector