ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-275; 50-323
License Nos.:	DPR-80; DPR-82
Report No.:	50-275/99-301; 50-323/99-301
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Nuclear Power Plant, Units 1 and 2
Location:	7 1/2 miles NW of Avila Beach Avila Beach, California
Dates:	January 25 to 28, 1999
Inspectors:	T. O. McKernon, Chief Examiner, Operations Branch R. E. Lantz, Examiner, Operations Branch
Accompanying Personnel:	Desiree Smith, Examiner in Training, Region III
Approved By:	John L. Pellet, Chief, Operations Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	Post Written Examination Comments
Attachment 3:	Written Examination and Answer Key

EXECUTIVE SUMMARY

Diablo Canyon Nuclear Power Plant, Units 1 and 2
NRC Inspection Report No. 50-275/99-301; 50-323/99-301

NRC examiners evaluated the competency of six senior operator applicants for issuance of operating licenses at the Diablo Canyon Power Plant, Units 1 and 2. The licensee developed the initial license examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8. NRC examiners reviewed, approved, and administered the examinations. The initial written examinations were administered on January 25, 1999. The NRC examiners administered the operating tests on January 26-28, 1999.

Operations

- All six applicants demonstrated the requisite knowledge and skills to satisfy the requirements of 10 CFR Part 55 and were issued senior operator licenses (Section O4.1).

- Overall, licensed operator applicants performed well during the examination. Operators demonstrated good 3-way communications practices, peer checking, and crew briefs. No generic performance weaknesses were identified (Section O4.2).

- The licensee's initial examination submittal was considered acceptable for administration requiring only minor enhancement suggestions. However, subsequent post-written examinations resulted in, the licensee commenting on six written examination questions, which required justification by the licensee and an explanation of how future post-examination comments will be minimized (Section O5.1).

Plant Support

- Housekeeping and material condition of the plant observed coincident to plant walkthroughs was good (Section F8.1).

Report Details

Summary of Plant Status

Both units operated at essentially 100 percent power for the duration of this inspection.

I. Operations

O4 Operator Knowledge and Performance

O4.1 Initial Written Examination

a. Inspection Scope

The licensee developed the written examination with dedicated training instructors on the security agreement and used facility training and operations staff on security agreement to validate the examination. The licensee proctored the administration of the written examination to the license applicants on January 25, 1999. The licensee staff proposed grading for the written examinations, analyzed the proposed results, and presented their evaluation and draft resultant comments for examination revision to the chief examiner on January 28, 1999. The licensee formally transmitted the examination comments to the NRC on February 1, 1999.

b. Observations and Findings

The minimum passing score was 80 percent. All applicants (six senior operators) passed with scores ranging from 80.8 to 88.8 percent, with an average score of 85.6 percent. The NRC specifically notified the licensee learning services representative of one individual, who passed with a score of 80.8 percent, for consideration of additional enhancement or remedial training. The grades reflected the results after incorporation of the accepted examination changes recommended by the licensee as a result of post-examination question analysis were incorporated. The NRC also revised one additional question based on post-examination analysis.

The licensee provided comments and the appropriate references for six questions as described in Attachment 2. Three questions were recommended for deletion: SRO Question 29 because of depth of knowledge; SRO Question 75 because three of the choices were correct answers, and Question 94 because of depth of knowledge. Questions 42, 50, and 57 were revised to accept two correct answers. The chief examiner reviewed and accepted some of the recommendations based on the technical merits of each recommendation and the material references provided by the licensee. However, the NRC did not accept the recommendation to delete Questions 29 and 75. Question 29 was not deleted because the correct choice did not require detailed knowledge of electrical schematics, but rather whether or not the spent fuel pump automatically started following an accident. Choice d was the only correct answer. Question 75 was not deleted but accepted with two possible correct choices, b or d. Since the auxiliary feedwater pump draws its steam supply from Steam Generators 1-2 and 1-3, the operator's decision to choose one of the two steam generators is valid.

However, Steam Generator 1-2 (choice b) should be selected rather than Steam Generator 1-3 (choice c) because pressure is lower and level is higher. Choice d, Steam Generator 1-4, which is not hot and dry, is also a correct answer in accordance with Procedure FR-H.1. The licensee's submitted examination comments are included as Attachment 2 to this inspection report.

The NRC also reviewed other questions missed by a majority of examinees and determined Question 60 to have two possible correct answers (choices b or d). Choice d was considered a correct choice because the wording of the distractor was not specific enough to discount it as a correct answer.

The licensee's post-examination test analysis indicated that more than half of the applicants missed the same ten questions. Six of the questions were submitted for comment. The chief examiner and the licensee determined that there were no significant inter-relationships to indicate generic weaknesses in knowledge or ability. The licensee stated that all missed questions would be reviewed with the individuals as part of the training department's remediation prior to assuming shift watch.

c. Conclusions

All six applicants demonstrated the requisite knowledge and skills to satisfy the requirements of 10 CFR Part 55 and were issued senior operator licenses.

O4.2 Initial Operating Test

a. Inspection Scope

The examination team administered the various portions of the operating test to the six applicants between January 26-28, 1999. Each applicant participated in three dynamic simulator scenarios and received a walkthrough test, which consisted of five system job performance tasks (except for the one senior reactor operator-instant applicant, who performed ten tasks), together with two followup questions for each system. Additionally, each applicant was tested on five subjects in four administrative areas by answering two questions or performing one task for each subject.

b. Observations and Findings

The examiners observed effective communications and good peer checks of control board activities during the dynamic simulator scenarios. Good status updates and crew briefs were practiced. Good plant and component awareness was observed during the walkthrough portion of the operating tests. The crews utilized effective three-way communications.

Applicants displayed good knowledge of the location and operation of local plant components. The applicants responded accurately to the walkthrough followup questions, which indicated a depth of associated system knowledge.

c. Conclusions

All applicants passed all sections of the operating test. Operators demonstrated good 3-way communications practices and good peer checks during the dynamic simulator scenarios. Overall, operators performed well during the examination.

O5 Operator Training and Qualification

O5.1 Initial Licensing Examination Development

The licensee developed the initial licensing examination in accordance with NUREG-1021.

O5.1.1 Examination Outline

a. Inspection Scope

The licensee submitted the initial examination outline on September 25, 1998. The examiners reviewed the submittal against the requirements of NUREG-1021.

b. Observations and Findings

The chief examiner provided only minor enhancement suggestions related to a balanced mix of malfunctions and power maneuvers in the dynamic simulator scenarios. Some other minor enhancements were suggested to the scenarios to ensure that senior operator applicants were evaluated in exercising the facility's technical specifications.

c. Conclusions

The licensee's examination outline was acceptable. Minor enhancements suggested by the chief examiner were incorporated.

O5.1.2 Examination Package

a. Inspection Scope

The licensee submitted the initial examination package on November 20, 1998. The chief examiner reviewed the submittal against the requirements of NUREG-1021.

b. Observations and Findings

The licensee submitted 100 draft written examination questions. The chief examiner provided comments or questions on 13 questions. In resolving these comments, the licensee revised or replaced 10 questions. The remaining questions were found to be satisfactory. The majority of the chief examiner's comments were enhancements and not considered substantive. The examinations were acceptable for administration as submitted. Additional review of the examination against the audit examination resulted in other changes to the job performance measures and some of the scenarios to avoid

any duplication. As discussed in Section O4.1, following post-examination review, one question was deleted and credit for multiple answers on five questions was allowed. The failure to make these changes would not have invalidated the examinations or degraded their discriminatory value. The examinations were considered acceptable for administration as submitted. However, because the licensee submitted greater than 5 percent of the questions for comment the licensee was requested to respond with information related to changes in their examination development process, which will improve future examinations and preclude similar recurrences.

The licensee submitted one set of operating tests, which included a total of ten job performance measures, one administrative tests, three scenarios, and one backup scenario. The submitted scenarios were considered acceptable for administration. However, some enhancement suggestions were incorporated during NRC validation to add better balance to the scenarios. The submitted facility walkthrough subsection of the examination discriminated at the required level. Some enhancement suggestions were incorporated to better facilitate the test administration and eliminate any duplication with the audit examination. While some enhancements and revisions to the operating tests were made, the number of revisions was few and the changes did not impact administering the examination.

Final revisions to the examination were completed prior to the examination. The licensee's training department and operations department provided excellent support during the development and administration of the examination.

c. Conclusions

The licensee's initial examination submittal was considered acceptable for administration requiring only minor enhancement suggestions. However, the licensee commented on six written examination questions following, which required justification by the licensee and an explanation of how future post-examination comments will be minimized

O5.2 Simulation Facility Performance

The examiners observed simulator performance with regard to fidelity during the examination validation and administration. The simulation facility supported the examination administration well. No problems were observed.

IV. Plant Support

F8 Miscellaneous Fire Protection Issues

F8.1 General Comments

The examiners observed good plant housekeeping and condition of external panel and equipment coincident with the inplant walkthrough portion of the examination. The facility was reasonably clean, well lighted, and the floors were clear and free of debris.

V. Management Meetings

X1 Exit Meeting Summary

The examiners presented the inspection results to members of the licensee management at the conclusion of the inspection on January 28, 1999. The licensee acknowledged the findings presented.

The licensee did not identify any information or materials examined as proprietary during the inspection.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

S. Kettlesen, Supervisor, Licensing
G. Goelzer, Acting Operations Director
T. Blake, Learning Services Director
D. Burns, Training Instructor
R. Jett, Training Leader
J. Haynes, Training Leader
J. Molden, Operations Manager
B. Garrett, Operations Director
J. Becerra, Instructor

ATTACHMENT 2

Facility Initial License Written Examination Comments

Pacific Gas and
Electric Company

David H. Oatley
Vice President–Diablo Canyon
Operations and Plant Manager

Diablo Canyon Power Plant
PO Box 56
Avila Beach, CA 93424

805.545.6000

January 29, 1999

PG&E Letter DCL-99-012

Thomas O. McKernon, Chief Examiner
U.S. Nuclear Regulatory Commission, Region IV
611 Ryan Plaza Dr., Suite 400
Arlington, TX 76011-8064

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
NRC License Written Examination Formal Comments

Dear Mr. McKernon:

In accordance with NUREG 1021, Interim Revision 8, PG&E is providing the enclosed formal comments on the written examination administered to Diablo Canyon Power Plant license candidates on January 25, 1999.

PG&E appreciates the NRC staff efforts during the entire examination and review cycle.

If you have any questions, please contact Roger Jett, Operations Training Supervisor, at (805) 545-3439.

Sincerely,

J E Molden for

David H. Oatley

Enclosures

cc: Timothy M. Blake
David L. Burns
Roger L. Jett
David L. Proulx

TLH/1753

Question #	Question	Recommendation	Justification
29	The following conditions exist: • A LOCA has occurred • Safety Injection has actuated • Phase B isolation has actuated • 4 Kv buses are being powered from their respective diesels • All equipment has had time to sequence on. WHICH ONE (1) of the following is the expected response of the Spent Fuel Pool (SFP) cooling system without any operator actions? a. SFP pump1-1 restarts, SFP temperature increases due to CCW flow isolation to the SFP heat exchanger. b. SFP pump 1-2 restarts, SFP temperature decreases due to increased CCW flow. c. Selected SFP pump restarts, SFP temperature decreases due to increased CCW flow. d. Neither SFP pump restarts, SFP temperature increases due to CCW flow isolation to the SFP heat exchanger. ANSWER D	Delete question from exam.	The question is more appropriate as a JPM followup question where the examinee has access to the applicable electrical drawings. Depth of knowledge required for this question is too detailed, especially without references. It was an oversight by the facility not to have provided the necessary reference drawing as part of the written examination reference package.

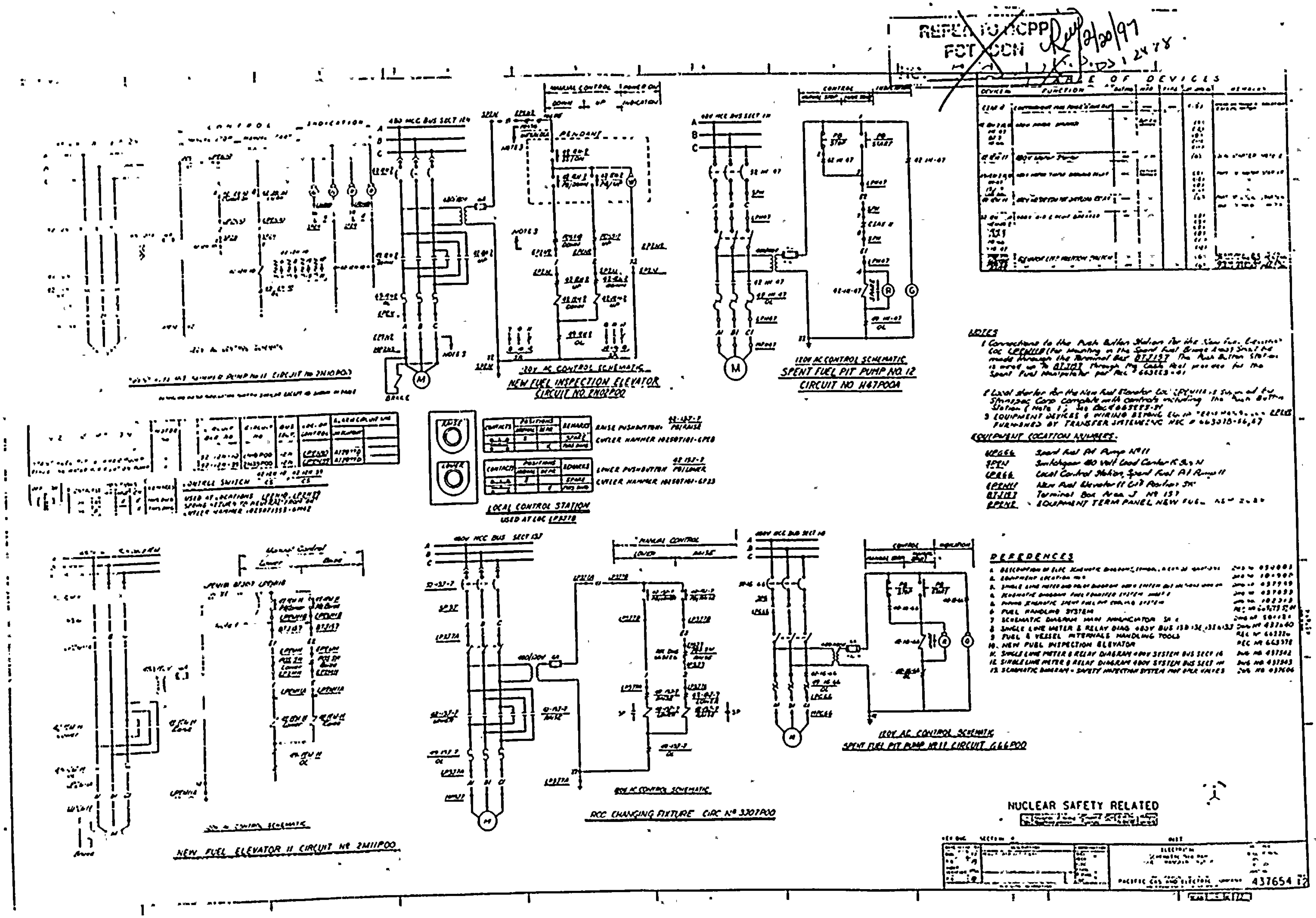

TABLE OF DEVICES
NEW FUEL INSPECTION ELEVATOR
CIRCUIT NO 2N02P00
120V AC CONTROL SCHEMATIC
SPENT FUEL PIT PUMP NO. 12
CIRCUIT NO H47P00A
LOCAL CONTROL STATION
NOTES
EQUIPMENT LOCATION NUMBERS
REFERENCES
RCC CHANGING FIXTURE CIRC Nº 3307P00
NEW FUEL ELEVATOR II CIRCUIT Nº 2M11P00
SPENT FUEL PIT PUMP Nº11 CIRCUIT G66P00
NUCLEAR SAFETY RELATED
437654

Question #	Question	Recommendation	Justification
42	Given the following on Unit 1: • Unit is ramping to 100% • Reactor power is at 90% with Control rods in automatic • A Control Bank D group 1 rod drops into the core without causing a reactor trip; no trip is required. • Operators have implemented OP AP-12C, "Dropped Control Rod" WHICH ONE (1) of the following describes the required actions to establish initial recovery conditions 20 minutes following the dropped rod event? a. No action is required. b. Reduce turbine load to reduce Reactor Power to ~ 85%. c. Reduce turbine load to reduce Reactor Power to less than 50%. d. Set Tavg 1.5°F above Tref by withdrawing control bank rods as necessary. ANSWER: B.	Accept A & B as correct answers.	OP AP-12C, "Dropped Control Rod" requires that Reactor Power be reduced as necessary such that the steady state power level attained after the rod is recovered is less than 90%. The recovery actions will depend on the initial conditions (Rx power) and the stable power level at the time of rod recovery. Plant response should be as follows: • Single control rod drops adding negative reactivity. • Tavg decreases until rod control causes auto rod withdrawal to recover temp. • Procedure directs rod control to MANUAL, stopping auto rod withdrawal. • Both Reactor Power and Tavg are less than their initial value. • Procedure directs matching Tavg & Tref which will be accomplished by lowering turbine load. • At time of recovery $P < P_o$ & $Tavg \leq Tavg_o$ Therefore the actual power level at the time of rod recovery will be greater than 90% and require the action stated in "b" if the initial power was 100%. However if the initial power level was only 90% as stated in the stem of the question, then the power level will already be at approximately 85% and require no further action with regards to power.

TITLE: DROPPED CONTROL ROD

11/5/98
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure provides instructions for plant operation when a control rod becomes disengaged from its drive mechanism and drops into the core.

2. SYMPTOMS

2.1 Control rods stepping out (if select switch is in auto)

2.2 Rapid drop in TAVG indication

2.3 Rapid drop in reactor power

2.4 Rod bottom light (DRPI panel)

2.5 Possible power range high flux rate status light and P-250 printout

2.6 Possible Main Annunciator Alarms:

2.6.1 PWR RNGE DEV/QPTR (PK03-10)

a. Pwr Rnge Lower Quadrant Power Tilt

b. NIS Pwr Rnge Channel Flux Deviation

c. Pwr Rnge Upper Quadrant Power Tilt

2.6.2 DRPI Failure/Rod Bottom (PK03-21)

a. Rod Position Indicator Rod Bottom

2.6.3 TAVG DEVIATION FROM REF (PK04-03)

a. TAVG Deviation TAVG-TREF Lo

2.6.4 Rod Cont Urgent Failure (PK03-17)

a. Rod Cont Sys Urgent Failure

2.6.5 P-250 Rx Alm Axial Flux/Rod Pos (PK03-25)

a. P-250 Computer Axial Offset Alarm

b. P-250 Computer Rod Position Dev or Rod Bank Sequence

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ONLY One Control Rod Dropped	Trip the Reactor and GO TO EOP E-0, REACTOR TRIP OR SAFETY INJECTION
2. PLACE Rod Control in MANUAL	
3. STOP Any Load Change In Progress AND Allow Conditions To Stabilize:	
a. Reactor is critical.	a. Place the reactor in Mode 3 by fully inserting the control rods. GO TO OP L-5, PLANT COOLDOWN FROM MINIMUM LOAD TO COLD SHUTDOWN
4. ADJUST Turbine Load To Match TAVG AND TREF	
5. CHECK Axial Flux Difference Within Tech Spec Limits	Refer to Tech Spec 3.2.1.
Calculate QPTR per STP R-25:	
a. Verify LESS than 1.09.	a. Refer to Tech Spec 3.2.4 action b.
b. Verify LESS than 1.02.	b. Refer to Tech Spec 3.2.4 action a.
7. VERIFY Rod Control System Had No Urgent Failure:	
a. Verify Rod Cont Urgent Failure (PK03-17) - OFF	1. Do not attempt to move rods or reset the Urgent Failure. 2. Contact MS for trouble shooting. 3. Refer to AR PK03-17, ROD CONT URGENT FAILURE.

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

8. FIND And Correct Cause Of The Dropped Rod:

NOTE: See Appendix A for typical power cabinet fuse arrangement. Refer to STP R-1B, Attachment 7.4 (7.5 on Unit 2) for specific fuse locations.

a. Dispatch an operator to the affected rod control cabinet to check indicator fuses for the lift, moveable, and stationary grippers

b. Contact MS to initiate troubleshooting and repairs

c. Refer to Tech Specs 3.1.1.1 and 3.1.3.1

NOTE: The lift, stationary, or moveable gripper coils can have a blown fuse and not have an urgent failure alarm because the regulation failure cards look at auctioneered high current from all four coils.

9. CONTACT Reactor Engineering Regarding the Dropped Rod to Obtain:

a. Guidance on rate of control rod movement during recovery

b. Power level at which recovery should be performed

10. Record Adequate Data to Track:

a. How long rod has been dropped.

b. Movement of other control rods during the subsequent recovery

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **ESTABLISH Initial Recovery Conditions:**	
a. Check the time since the rod dropped - LESS THAN ONE hour	a. Reduce reactor power to LESS THAN 50%. Continue with Step 11.c when power is LESS THAN 50%.
b. Reduce power as necessary such that the steady state power level attained after the rod is recovered is LESS THAN 90% Reactor Power	
c. Set TAVG 1.5°F below TREF by inserting control bank rods as necessary	
12. **PREPARE For Rod Withdrawal:**	
a. Select the affected rod bank on the Bank Selector Switch	
b. Record the step counter position on the affected group Bank ____ Group _____ Step _____	
c. Reset the step counter to zero on the affected group only	
d. Locally open lift coil disconnect switches on all rods in the affected bank <u>except the dropped rod</u>. (Lift Coil Disconnect Cabinet 115' Elev Aux Bldg)	
e. Check dropped rod is in a control bank	e. GO TO Step 12.g.

-THIS STEP CONTINUED ON NEXT PAGE-

Question #	Question	Recommendation	Justification
50	Unit 1 has the following conditions: • Emergency Boration is required due to stuck rods. • Emergency Boration via normal makeup results in 25 gpm flow with the Boric Acid pump in high speed. • Emergency Boration is accomplished by using CVCS-8104. WHICH ONE (1) of the following describes the method used to determine the total number of gallons of boric acid added? a. FI-113A (Emergency Boration Flow meter on Vertical Board 2) and the duration the valve is open. b. FR-110 (Boric Acid and Primary water flow recorder on Control Console 2). c. YIC-110 (Boric Acid Integrator on Control Console 2). d. XFIT-113 (Emergency Boration Flow Transmitter in the Cable Spreading Room) and the duration the valve is open. ANSWER: D.	Accept A & D as correct answers.	Per OP AP-6, "Emergency Boration," the note preceding step 2 indicates that "Emergency Boration Flowmeter FI-113 may peg high at 50 GPM. XFIT in the Cable Spreading room may be used for higher flowrates or to determine total gallons of boric acid added via the Emergency Boration flowpath." The conditions stated in the stem of the question do not specify the specific problem/ reason for the decreased boration flowrate via the normal makeup mode. It is quite possible that the blockage/problem is such that when 8104 is opened, flow would not necessarily peg FI-113 in the control room. In this case using FI-113 and duration of valve opening is acceptable. XFIT is only required if FI-113 pegs high at greater than 50 gpm but can also be used for any range of flow rates. The stem of the question does not indicate that the boration flow rate is necessarily greater than 50 gpm.

TITLE: EMERGENCY BORATION

APPROVED: ______________________ 06/04/97 DATE 06/05/97 EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure covers situations which require emergency boration and the methods for accomplishing this operation. Various options of emergency boration are discussed in this procedure.

1.2 The preferred option is using the VCT Makeup System. The next option is borating through the emergency boration valve (CVCS-8104). The next alternate option is the use of the RWST. The use of manual emergency borate valve CVCS-8471 is too involved and takes so much time that it is ONLY USED as the LAST option.

NOTE: Emergency boration is defined as a flow GREATER THAN 30 GPM of 7,000 to 7,700 PPM boron or equivalent. Channel inaccuracies have been included where appropriate yielding flow values of greater than 30 GPM.

2. SYMPTOMS

Any one of the following conditions requires emergency boration of the specified amount:

2.1 Control rods inserted below the low-low insertion limit when critical.

ROD BANK LO LO INSERTION LIMIT (PK03-14)

2.2 Failure of any 2 control rods to fully insert following a reactor trip as indicated by rod position indication and rod bottom lights.

2.3 Uncontrolled Reactor Coolant System cooldown following a reactor trip with no ESF action.

2.4 Uncontrolled or unexplained reactivity increase as indicated by:

2.4.1 Unexplained control rod insertion.

2.4.2 Increasing TAVG or nuclear power with no increased load demand.

2.4.3 Unexpected increasing count rate when shutdown.

2.5 When boration is required and normal boration through the VCT makeup system is not possible.

2.6 Shutdown margin less than acceptable minimum limits per Tech Spec 3.1.1.1, 3.1.1.2 and 3.9.1.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE 1: 900 gallons of 4% boric acid provides 100 ppm INCREASE TO THE RCS, BOL. Calculated values may be used in place of this thumbrule.

NOTE 2: If Letdown is NOT in service, then it will be necessary to cool down 50°F per hour while injecting Boric Acid at 30 gpm in order to maintain a constant pressurizer level.

1. INITIATE Emergency Boration:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Verify GREATER THAN 55 gpm charging flow to the RCS	a. GO TO OP AP-17, LOSS OF CHARGING.
b. Place VCT make up control in BORATE position	
c. Set boron flow controller HC-110 pot setting to 9.5 turns	c. Increase demand manually to 100% on HC-110.
d. Set integrator for desired gallons of boric acid. Refer to Appendix A for boration requirements	
e. Place M/U controller 1/MU in START position - Adjust HC-110 pot setting to obtain GREATER THAN 32 GPM of boric acid flow	e. Perform the following: 1) Verify BA Transfer Pp - HIGH SPEED 2) IF VCT pressure GREATER THAN 30 PSIG, THEN Vent the VCT by opening CVCS-8101 until LESS THAN 30 PSIG 3) IF Boric acid flow remains LESS THAN 32 GPM, THEN GO TO Step 2.
f. GO TO Step 3	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Emergency Boration Flowmeter FI-113 may peg high at 50 GPM. XFIT-113 in the Cable Spreading room may be used for higher flowrates or to determine total gallons of boric acid added via the Emergency Boration flowpath.

2. INITIATE Alternate Boration Method

a. OPEN CVCS-8104 and verify greater than 33 GPM Emergency Boration Flow

RESPONSE NOT OBTAINED:

a. Perform one of the following in order of preference:

1) Swap Charging Pp suction to the RWST.

a. OPEN 8805A AND 8805B.

b. CLOSE LCV-112B AND LCV-112C.

c. VERIFY GREATER THAN 105 GPM charging flow.

OR

2) Locally OPEN CVCS-8471 (100' Blender Room).

3. CHECK Sufficient Boric Acid Available:

In Service Boric Acid Tank level GREATER THAN required gallons of Boric Acid per Appendix A

RESPONSE NOT OBTAINED:

a. Stop the Boric Acid Transfer Pp not aligned to the blender.

b. Locally OPEN CVCS-8476, Boric Acid Transfer Pp crosstie. (100' Behind Suction to BA Transfer Pp 1-1/2-2).

WHEN Sufficient BA inventory restored,
THEN Realign the system per OP B-1C:II, 4% BORIC ACID SYSTEM - PLACE IN SERVICE.

Question #	Question	Recommendation	Justification
57	Given the following initial conditions: • Unit 1 is at 100% power and ramping to 50% power for condenser tube cleaning. How will the loss of MANUAL power supply to HC-459D, "Pressurizer Master Level Controller," affect pressurizer level control during the ramp? a. No operator actions should be necessary. b. Throttle open on HCV-142, "RCP Seal Flow control valve." c. Throttle closed on FCV-128, "CCP flow control valve." d. Adjust HC-459-D, "Pressurizer Master Level Controller. ANSWER: A.	Accept A & C as correct answers.	Theoretically as power decreases, Tavg decreases, causing program reference level to decrease. Actual pressurizer level will decrease, following reference level, due to RCS inventory shrink, therefore charging flow should not have to be adjusted (manually or automatically) during the ramp. Realistically (based on operational experience), whether or not charging flow will need to be adjusted to compensate for a difference in actual vs reference pressurizer level depends on the ramp rate and adjustments to keep ΔI on target. The slower, more controlled ramp to 50% could require no operator action whereas a more aggressive ramp of 50MW/min to 200MW/min will require charging flow adjusting (manually if auto is OOS). The question does not specify the ramp rate. It also indicates that the reason for the ramp is to conduct condenser tube cleaning which could be interpreted to imply that the reason for the tube cleaning is elevated condenser DPs and therefore a sense of urgency to pursue a more aggressive ramp to 50%.

<table>
<tr><th>Question #</th><th>Question</th><th>Recommendation</th><th>Justification</th></tr>
<tr>
<td>75</td>
<td>Unit 1 has experienced a feedwater line break inside containment on Steam Generator 1-1 and a total loss of feedwater. FR-H.1 has been entered and feed and bleed of the RCS has been initiated. Shortly after opening the PORVs, the Turbine Driven Auxiliary Feedwater pump is returned to service and a source of feedwater is available. The operators are directed to restore a steam generator for a heat sink per FR-H.1 with the following plant conditions:

<table>
<tr><th>Indication</th><th>Loop 1</th><th>Loop 2</th><th>Loop 3</th><th>Loop4</th></tr>
<tr><td>S/G WR level (%)</td><td>0</td><td>12</td><td>7</td><td>3</td></tr>
<tr><td>S/G pressure (psig)</td><td>0</td><td>650</td><td>675</td><td>645</td></tr>
<tr><td>RCS hot leg temp (°F)</td><td>545</td><td>553</td><td>554</td><td>548</td></tr>
</table>
• Containment pressure = 3.5 psig
• Core exit T/Cs are stable at an average value = 560°F

Which steam generator should be fed first?

a. S/G 1-1
b. S/G 1-2
c. S/G 1-3
d. S/G 1-4

ANSWER D</td>
<td>Possible 3 correct answers. Delete question from exam.</td>
<td>The intent of the question is to test Hot Dry steam generator criteria, foldout page for FR-H.1 Based only on this criteria, S/Gs 1-2 and 1-3 are Hot and Dry and therefore S/G 1-4 which is not Hot and Dry SHOULD be fed first.

However based on the specific conditions given in the stem of the question, the only source of feedwater to the S/Gs is the Turbine Driven AFW pump which draws its steam supply from S/Gs 1-2 and 1-3. The risk of losing AFW is a valid concern under these circumstances. The procedure gives the latitude to make the decision to feed other than the non Hot Dry S/G by stating that the non Hot Dry S/G "SHOULD" be fed first vice "SHALL" be fed first.

The question stem gave no reason to believe that there were any other potential sources of feedwater since it was necessary to commence bleed and feed due to levels being less than 23% WR. Therefore under these circumstances feeding S/G 1-2 or 1-3 to maintain a feedwater source is an acceptable decision.</td>
</tr>
</table>

1.0 **SECONDARY INTEGRITY CRITERIA**

IF Any S/G Pressure is decreasing in an Uncontrolled manner or has completely depressurized, AND has NOT been isolated, unless it is needed for RCS cooldown,
THEN GO TO EOP E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.

2.0 **BLEED AND FEED CRITERIA**

IF a. WR S/G Level in any 3 S/Gs LESS THAN 23% [34%], AND ALL NR S/G Levels are LESS THAN 6% [16%],
OR
b. PZR Pressure is GREATER THAN 2335 PSIG due to a loss of secondary heat sink,
THEN STOP ALL RCPs
AND Initiate Bleed and Feed, Steps 12 through 18.

3.0 **RESTART SAFEGUARDS EQUIPMENT AFTER LOSS OF OFFSITE POWER**

IF Offsite Power is lost AFTER SI RESET,
THEN
- o Restart Safeguards equipment as necessary
- o IF In recirculation mode,
  THEN CCPs should be held in STOP/RESET until RHR is in service.
- o REFER TO APPENDIX A for guidance.

4.0 **COLD LEG RECIRCULATION SWITCHOVER CRITERION**

IF RWST Level decreases to LESS THAN 33%,
THEN GO TO EOP E-1.3, TRANSFER TO COLD LEG RECIRCULATION.

5.0 **CONTMT SPRAY INITIATION CRITERIA**

IF Contmt Pressure is GREATER THAN 22 PSIG,
THEN Initiate Contmt Spray.

6.0 **ESTABLISHING FEED TO A HOT DRY STEAM GENERATOR CRITERIA**

A "Hot Dry" S/G is a S/G with Hot Leg temperature GREATER THAN 550°F AND WR level LESS THAN 7% [17%]. Feeding a Hot Dry S/G should be performed only when another intact S/G is NOT available for cooldown. (When depressurizing a S/G to inject a low pressure water source per Step 18, all S/Gs do not have to be Hot and Dry prior to feeding.)
- o IF Hot Leg temperatures are increasing,
  THEN Feed ONE Hot Dry S/G at MAXIMUM rate.
  WHEN Hot Leg temperature is LESS THAN 550°F
  THEN Check for SGTR. Use another S/G if a SGTR exists.
- o IF Hot Leg temperatures are stable or decreasing,
  THEN IMPLEMENT EOP FR-H.5.

7.0 **AFW SUPPLY SWITCHOVER CRITERION**

IF CST level decreases to LESS THAN 10%,
THEN IMPLEMENT OP D-1:V, ALTERNATE AFW SUPPLIES.

Question #	Question	Recommendation	Justification
94	WHICH ONE (1) of the following conditions would require the termination of a containment vent or purge? a. Failure of either RM-44A or B while in Mode 6 with movement of irradiated fuel in containment. b. Failure of either RM-44A or B while in Mode 4. c. Failure of either RM-44A or B while in Mode 6 during core alterations. d. RM-44A out of service and RM-44B failure in Mode 5. ANSWER: B.	Delete question from exam.	The question is more appropriate as a JPM followup question where the examinee has access to the applicable Equipment Control Guidelines and Technical Specifications. Depth of knowledge required for this question is too detailed, especially without references since it is a combination of an ECG and a Technical Specification and the required action is not 1 hour or less. It was an oversight by the facility not to have provided the necessary reference procedures as part of the written examination reference package.

39.0 INSTRUMENTATION

39.4 Radioactive Gaseous Effluent Monitoring Instrumentation

ECG 39.4 The Radioactive Gaseous Effluent monitoring instrumentation channels shown in Table 39.4-1 shall be OPERABLE* with their alarm/trip setpoints set to ensure the limits of the Radiological Monitoring and Controls Program (AP A-81) are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the offsite dose calculation procedure (CAP A-8).

APPLICABILITY: In accordance with Table 39.4-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
One or more required radiation monitors channels listed in Table 39.4-1 inoperable.	Perform Required Actions as specified in Conditions referenced in Table 39.4-1.	As specified in applicable ACTION conditions.

(continued)

* As described in the Diablo Canyon Power Plant Technical Specifications.

Table 39.4-1

Radiological Gaseous Effluent Monitoring Instrumentation

FUNCTION	REQUIRED NUMBER OF CHANNELS	REQUIRED MODE	ECG 39.4 ACTION CONDITION
1. GASEOUS RADWASTE SYSTEM			
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RM-22)	1	At all times	A
2. Plant Vent System			
a. Noble Gas Activity Monitor - Providing Alarm RM-14 or 14R	1 per Unit	At all times	B
b. Iodine Sampler (the cartridge and filter only, associated with): RF-24 or RF-24R	1	At all times	C
c. Particulate Sampler (the cartridge and filter only, associated with): RF-28 or RF-28R	1	At all times	C
d. Plant Vent Flow Rate Monitor FR-12 (Fed from FT-12 or FT-12R)	1	At all times	D
e. Iodine Sampler Flow Monitor: FT-813 or FT-814	1	At all times	D
3. Containment Purge System (In Accordance With Tech Spec)	TS 3.3.2 TS 3.3.3.1	TS 3.3.2 TS 3.3.3.1	TS 3.3.2 TS 3.3.3.1
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release Unit 1: (RM-44A and 44B) Unit 2: (RM-44A and 44B)	2(1) per Unit	1,2,3,4,6 (2)	

(1) Only 1 channel required in Mode 6.

(2) During CORE ALTERATIONS* or movement of irradiated fuel within containment.

* As described in the Diablo Canyon Power Plant Technical Specifications.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Values column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 24 months. Each test shall include at least one train such that both trains are tested at least once per 48 months and one channel per function such that all channels are tested at least once per N times 24 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Fan Cooler Units, and Component Cooling Water)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3, 4	20
d. Pressurizer Pressure-Low	4	2	3	1, 2, 3#	20
e. DELETED					
f. Steam Line Pressure-Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	20

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2.. Containment Spray (coincident with SI signal)					
a. Manual	2	2 with 2 coincident switches	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-High	4	2	3	1, 2, 3, 4	17
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	19
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual	2	2 with 2 coincident switches	2	1, 2, 3, 4	19

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure-High-High	4	2	3	1, 2, 3, 4	17
c. Containment Ventilation Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	18
2) Deleted					
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)	2	1	2	1, 2, 3, 4	18
4. Steam Line Isolation					
a. Manual	1 manual switch/steam line	1 manual switch/steam line	1 manual switch/operating steam line	1, 2, 3, 4	24

TABLE NOTATIONS

\# Trip function may be blocked in this MODE below the P11 (Pressurizer Pressure Interlock) Setpoint.

\## Trip function automatically blocked above P-11 (Pressurizer Pressure Interlock) Setpoint and is automatically blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.

\### For Mode 3, the Trip Time Delay associated with the Steam Generator Water Level-Low-Low channel must be less than or equal to 464.1 seconds.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, declare the affected Emergency Diesel Generator(s) inoperable and comply with the ACTION statements of Specification 3.8.1.1; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, declare the affected Emergency Diesel Generator(s) inoperable and comply with the ACTION statements of Specification 3.8.1.1; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves (RCV-11, 12, FCV 660, 661, 662, 663, 664) are maintained closed.

ACTION 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Fuel Handling Building				
a. Storage Area				
1) Spent Fuel Pool	1	*	≤ 75 mR/hr	30 & 32**(a)
2) New Fuel Storage	1	*	≤ 15 mR/hr	30 & 32**(a)
b. Gaseous Activity Fuel Handling Building Ventilation Mode Change(b)	1	*	Per the ODCP	32**
2. Control Room Ventilation Mode Change	2***	All	≤ 2 mR/hr	34
3. Containment				
a. Gaseous Activity				
1) Deleted				
2) RCS Leakage	1	1, 2, 3, 4	N.A.	31
3) Containment Ventilation Isolation (RM-44A or 44B)	1	6	Per the ODCP	33
b. Particulate Activity				
1) Containment Ventilation Isolation (RM-44A or 44B)	1	6	Per the ODCP	33
2) RCS Leakage	1	1, 2, 3, 4	N.A.	31

*With fuel in the spent fuel pool or new fuel storage vault.
**With irradiated fuel in the spent fuel pool.
***One channel for each normal intake to the Control Room Ventilation System (common to both units).
(a)Action 32 is not applicable to the Fuel Storage Area Monitors following installation of RM-45A and 45B.
(b)The requirements for Fuel Handling Building Ventilation Mode Change are applicable following installation of RM-45A and 45B.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 30 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint or an individual qualified in radiation protection procedures with a radiation dose rate monitoring device is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1. The provisions of Specification 3.0.4 are not applicable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the Control Room Ventilation System in a recirculation mode with the HEPA filter and charcoal adsorber bank in operation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

November 22, 1999

Gregory M. Rueger, Senior Vice President
and General Manager
Nuclear Power Generation Bus. Unit
Pacific Gas and Electric Company
Nuclear Power Generation, B32
77 Beale Street, 32nd Floor
P.O. Box 770000
San Francisco, California 94177

SUBJECT: NRC INSPECTION REPORT 50-275/99-05; 50-323/99-05

Dear Mr. Rueger:

NRC Inspection Report 50-275/99-05; 50-323/99-05 for the corrective action program inspection has been rescheduled for the year 2000 and a new report number will be issued for this inspection. Therefore, no inspection report will be issued for this report number.

Sincerely,

John L. Pellet, Chief
Operations Branch
Division of Reactor Safety

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

cc:
Dr. Richard Ferguson
Energy Chair
Sierra Club California
1100 llth Street, Suite 311
Sacramento, California 95814

Nancy Culver
San Luis Obispo Mothers for Peace
P.O. Box 164
Pismo Beach, California 93448

PDR ADOCK 9912350129

Chairman
San Luis Obispo County Board of
Supervisors
Room 370
County Government Center
San Luis Obispo, California 93408

Truman Burns\Mr. Robert Kinosian
California Public Utilities Commission
505 Van Ness, Rm. 4102
San Francisco, California 94102

Robert R. Wellington, Esq.
Legal Counsel
Diablo Canyon Independent Safety Committee
857 Cass Street, Suite D
Monterey, California 93940

Steve Hsu
Radiologic Health Branch
State Department of Health Services
P.O. Box 942732
Sacramento, California 94234-7320

Christopher J. Warner, Esq.
Pacific Gas and Electric Company
P.O. Box 7442
San Francisco, California 94120

David H. Oatley, Vice President
Diablo Canyon Operations and Plant Manager
Diablo Canyon Nuclear Power Plant
P.O. Box 3
Avila Beach, California 93424

Managing Editor
Telegram-Tribune
1321 Johnson Avenue
P.O. Box 112
San Luis Obispo, California 93406

bcc to DCD (IE01)

bcc distrib. by RIV:
Regional Administrator
Resident Inspector
DRP Director
RIV File
DRS Director
RITS Coordinator
Senior Project Inspector (DRP/E)
Branch Chief (DRP/TSS)

DOCUMENT NAME: R:_DC\dc905rp-lmb.wpd
To receive copy of document, indicate in box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

RIV:C:OB	N						
JLPellet/lmb JP							
11/1?/99							

OFFICIAL RECORD COPY

Distri24.txt

Distribution Sheet

Priority: Normal

From: Elaine Walker

Action Recipients:	Copies:	
S Bloom	1	Not Found
NRR/DLPM/LPD4-2	1	Not Found

Internal Recipients:		
TTC C-TNN	1	Not Found
RidsEdoMailCenter	1	OK
RGN FILE 01	1	Not Found
OGC/RP	1	Not Found
OE	1	Not Found
NRR/DRIP/REXB	1	Not Found
NRR/DIPM/OLHP	1	Not Found
NRR/DIPM/IIPB	1	Not Found
FILE CENTER	1	Not Found
DEDRO	1	Not Found
ACRS	1	Not Found

External Recipients:		
NRC PDR	1	Not Found
NOAC	1	Not Found
INEEL Marshall	1	Not Found

Total Copies: 16

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 993350189

Subject:
NRC INSPECTION REPORT 50-275/99-05; 50-323/99-05

Body:
PDR ADOCK 05000275 Q

Docket: 05000275, Notes: N/A

Docket: 05000323, Notes: N/A