

U.S. Nuclear Regulatory Commission
 Site-Specific Written Examination **Answer Key**

Applicant Information

Name:

Date: 13 April, 2007

Facility/Unit: DCPD / Units 1 & 2

Region: IV

Reactor Type: W

Start Time:

Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

RO/SRO-Only/Total Examination Values 74 / 25 / 99 Points

Applicant's Scores / / Points

Applicant's Grade / / Percent

Multiple Choice (Circle or X your choice)

NAME: **Answer Key**

If you change your answer, write your selection in the blank and initial.

- | | | | | | | | | | | | |
|--------------------------------------------------------------------------------------------------|-------------------------------------|-------------------------------------|-------------------------------------|-------------------------------------|-------|--------|-------------------------------------|-------------------------------------|-------------------------------------|-------------------------------------|-------|
| 001 | A | B | C | <input checked="" type="checkbox"/> | _____ | 026 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 002 | A | B | <input checked="" type="checkbox"/> | D | _____ | 027 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 003 | A | <input checked="" type="checkbox"/> | C | D | _____ | 028 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 004 | A | B | C | <input checked="" type="checkbox"/> | _____ | 029 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 005 | A | B | <input checked="" type="checkbox"/> | D | _____ | 030 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 006 | A | <input checked="" type="checkbox"/> | C | D | _____ | 031 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| <div data-bbox="64 635 178 682" style="border: 1px solid black; padding: 2px;">Deleted</div> 007 | A | B | <input checked="" type="checkbox"/> | D | _____ | 032 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 008 | <input checked="" type="checkbox"/> | B | C | D | _____ | 033 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 009 | A | <input checked="" type="checkbox"/> | C | D | _____ | 034 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 010 | A | <input checked="" type="checkbox"/> | C | D | _____ | 035 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 011 | <input checked="" type="checkbox"/> | B | C | D | _____ | ** 036 | <input checked="" type="checkbox"/> | <input checked="" type="checkbox"/> | C | D | _____ |
| 012 | <input checked="" type="checkbox"/> | B | C | D | _____ | 037 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 013 | A | B | <input checked="" type="checkbox"/> | D | _____ | 038 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 014 | A | <input checked="" type="checkbox"/> | C | D | _____ | 039 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 015 | A | B | <input checked="" type="checkbox"/> | D | _____ | 040 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 016 | A | <input checked="" type="checkbox"/> | C | D | _____ | 041 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 017 | A | B | C | <input checked="" type="checkbox"/> | _____ | 042 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 018 | A | B | <input checked="" type="checkbox"/> | D | _____ | 043 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 019 | <input checked="" type="checkbox"/> | B | C | D | _____ | ** 044 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 020 | A | B | C | <input checked="" type="checkbox"/> | _____ | 045 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 021 | A | <input checked="" type="checkbox"/> | C | D | _____ | 046 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 022 | A | <input checked="" type="checkbox"/> | C | D | _____ | 047 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 023 | A | B | <input checked="" type="checkbox"/> | D | _____ | 048 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 024 | A | <input checked="" type="checkbox"/> | C | D | _____ | 049 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 025 | A | B | C | <input checked="" type="checkbox"/> | _____ | 050 | A | <input checked="" type="checkbox"/> | C | D | _____ |

** Question 44: Original answer of "C" was revised to "B" after post-exam review

** Question 36: Two correct answers ("A" and "B")

Multiple Choice (Circle or X your choice)

NAME: **Answer Key**

If you change your answer, write your selection in the blank and initial.

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|-----|-------------------------------------|-------------------------------------|-------------------------------------|-------------------------------------|-------|-----|-------------------------------------|-------------------------------------|-------------------------------------|-------------------------------------|-------|
| 051 | A | <input checked="" type="checkbox"/> | C | D | _____ | 076 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 052 | A | <input checked="" type="checkbox"/> | C | D | _____ | 077 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 053 | A | B | <input checked="" type="checkbox"/> | D | _____ | 078 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 054 | A | B | <input checked="" type="checkbox"/> | D | _____ | 079 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 055 | A | B | C | <input checked="" type="checkbox"/> | _____ | 080 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 056 | A | <input checked="" type="checkbox"/> | C | D | _____ | 081 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 057 | A | B | <input checked="" type="checkbox"/> | D | _____ | 082 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 058 | <input checked="" type="checkbox"/> | B | C | D | _____ | 083 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 059 | <input checked="" type="checkbox"/> | B | C | D | _____ | 084 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 060 | A | B | C | <input checked="" type="checkbox"/> | _____ | 085 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 061 | <input checked="" type="checkbox"/> | B | C | D | _____ | 086 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 062 | A | B | <input checked="" type="checkbox"/> | D | _____ | 087 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 063 | <input checked="" type="checkbox"/> | B | C | D | _____ | 088 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 064 | A | B | <input checked="" type="checkbox"/> | D | _____ | 089 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 065 | <input checked="" type="checkbox"/> | B | C | D | _____ | 090 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 066 | <input checked="" type="checkbox"/> | B | C | D | _____ | 091 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 067 | A | B | C | <input checked="" type="checkbox"/> | _____ | 092 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 068 | A | B | <input checked="" type="checkbox"/> | D | _____ | 093 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 069 | A | <input checked="" type="checkbox"/> | C | D | _____ | 094 | A | B | <input checked="" type="checkbox"/> | D | _____ |
| 070 | A | B | <input checked="" type="checkbox"/> | D | _____ | 095 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 071 | <input checked="" type="checkbox"/> | B | C | D | _____ | 096 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 072 | A | <input checked="" type="checkbox"/> | C | D | _____ | 097 | A | B | C | <input checked="" type="checkbox"/> | _____ |
| 073 | <input checked="" type="checkbox"/> | B | C | D | _____ | 098 | <input checked="" type="checkbox"/> | B | C | D | _____ |
| 074 | A | B | <input checked="" type="checkbox"/> | D | _____ | 099 | A | <input checked="" type="checkbox"/> | C | D | _____ |
| 075 | A | B | <input checked="" type="checkbox"/> | D | _____ | 100 | <input checked="" type="checkbox"/> | B | C | D | _____ |

Question 1.

RO Question 28 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	003 K5.02	_____
	Importance Rating:	2.8	3.2

Proposed Question:

The plant is at 100% power, all rods out.

A reactor trip coincident with a loss of all offsite power occurs.

When reactor power drops to about 8-9% (due to prompt drop after the trip), core cooling exists because of:

- A. Negative MTC
- B. Trip with rods above the RIL
- C. Natural circulation flow
- D. Continued forced flow

Proposed Answer:

D. Continued forced flow

Explanation:

A incorrect. Negative MTC ensures negative reactivity as RCS temperature decreases following the trip.

B incorrect. Ensures SDM will be met.

C incorrect. Too soon for natural circulation.

D correct. Function of the flywheel - Increase pump coast down time, to extend the period of forced flow following a pump trip.

Technical Reference(s): STG A6

Proposed references to be provided to applicants during examination: None

Learning Objective: 35737 - State the purpose of RCP system and equipment

Question Source:

DCPP Bank A-0157

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: No changes to initial submittal. Expanded justification.

K/A: 003 K5.02 – Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP coastdown on RCS parameters (2.8/3.2)

Question 2.

RO Question 29 rev 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	004 A2.13	
	Importance Rating:	3.6	3.9

Proposed Question:

GIVEN:

- The plant is in MODE 4 when a LOCA occurs
- The crew is performing the actions of OP AP-24, Shutdown LOCA
- Charging injection valves have been opened

The crew has just started an SI pump when RWST level reaches 33%.

Which of the following actions should be taken by the crew?

- A. Monitor ECCS pumps for signs of cavitation due the decreasing RWST inventory.
- B. Immediately initiate makeup to the RWST to delay the time until the running pumps lose suction.
- C. Align the SI system for Cold Leg Recirculation using E-1.3, Transfer to Cold Leg Recirculation, to ensure flow is maintained to the core.
- D. Go to E-0 to align ECCS equipment, then transition to E-1.3, Transfer to Cold Leg Recirculation, to align the system for Cold Leg Recirculation to maintain core cooling.

Proposed Answer:

C. Align the SI system for Cold Leg Recirculation using EOP E-1.3, Transfer to Cold Leg Recirculation, to ensure flow is maintained to the core.

Explanation:

A incorrect. Action is to go use E-1.3.

B incorrect. This may be done later if there would be insufficient sump inventory.

C correct.

Per OP AP-24, caution:

CAUTION 1: IF RWST level decreases to less than 33%, the SI system should be aligned for cold leg recirculation using EOP E-1.3.

D incorrect. In another mode, this could be an appropriate action.

Technical Reference(s): OP AP-24, background

Proposed references to be provided to applicants during examination: None

Learning Objective: 3477 - Describe the major actions of abnormal operating procedures

Question Source:
Bank P-65994

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: Modified question to address both parts of KA, modified distractors B and D. Expanded justification.

K/A: 004 A2.13 – Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low RWST (3.6/3.9)

Question 3.

RO Question 30

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	005 K5.05	_____
	Importance Rating:	2.7	3.1

Proposed Question:

GIVEN:

- The plant is in MODE 5, solid
- One train of RHR in service
- Letdown Pressure Control Valve, PCV-135, is in AUTO
- RHR Letdown Flow is being maintained via CVCS HCV-133.
- Charging pump 1-2 is in service.

The running RHR pump trips.

Which of the following describes the RCS pressure response?

- A. RCS pressure INCREASES due to HCV-133 closing and charging is in service.
- B. RCS pressure INCREASES due to PCV-135 closing down and charging is in service.
- C. RCS pressure DECREASES due to PCV-135 opening.
- D. RCS pressure DECREASES due to the loss of RHR pump discharge pressure.

Proposed Answer:

- B. RCS pressure INCREASES due to PCV-135 closing down and charging is in service..

Explanation:

A incorrect. HCV-133 will not move.

B correct. OP L-5: "9.2.13 When operating in a solid condition with RCS pressure being maintained by CVCS 1-PCV-135, changes to the flow rate in the RHR loops or starting and stopping the RHR pumps will result in changes in RCS pressure. CVCS 1-PCV-135 should be operated in MANUAL and adjust as necessary to minimize RCS pressure transient."

When the pump trips, discharge pressure decreases. PCV-135 closes in an attempt to raise pressure. Letdown flow decreases, charging flow continues into a solid system and RCS pressure increases.

C incorrect. Operation is opposite of what occurs.

D incorrect. Pressure increases when the pump trips.

Technical Reference(s): OP L-5, Plant Cooldown From Minimum Load to Cold Shutdown

Proposed references to be provided to applicants during examination: None

Learning Objective: 9668 - Explain precautions and limitations within operating procedures

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: No changes to initial submittal.

K/A: 005 K5.05 - Knowledge of the operational implications of the following concepts as they apply the RHRS: Plant response during "solid plant": pressure change due to the relative incompressibility of water (2.7/3.1)

Question 4.

RO Question 31

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u> 2 </u>	<u> </u>
	Group:	<u> 1 </u>	<u> </u>
	K/A:	005 A4.03	
	Importance Rating:	2.8	2.7

Proposed Question:

GIVEN:

- Unit 1 cooldown is in progress
- RCS temperature is 300°F
- RCS pressure is 340 psig
- Train A Residual Heat Removal (RHR) pump is in service with flow set at 4000 total flow thru both RHR heat exchangers
- The current cooldown rate is 12°F/hr.

The SFM orders the cooldown rate increased from 12°F/hr to 40°F/hr over 10 minutes while maintaining RHR flow relatively constant.

Which of the following describes how the operator will increase the cooldown rate?

NOTE:

- HCV-670 – Bypass Flow Control Valve
 - HCV-637 – Flow Control Valve for Heat Exchanger 1-2
 - HCV-638 - Flow Control Valve for Heat Exchanger 1-1
- A. Close down on HCV-670. HCV-637 and HCV-638 will open to maintain a constant flow.
- B. Open both HCV-637 and HCV-638. HCV-670 will close down to maintain a constant flow.
- C. Open either HCV-637 or HCV-638 while opening HCV-670.
- D. Open both HCV-637 and HCV-638 while closing HCV-670.

Proposed Answer:

- D. Open both HCV-637 and HCV-638 while closing HCV-670.

Explanation:

A incorrect. Valves must be manually operated.

B incorrect. HCV-670 does not respond automatically.

C incorrect. An even cooldown requires both heat exchanger valves to be operated. HCV-670 must be closed not opened.

D correct. Cooldown is increased by increasing flow thru the heat exchangers (HCV-637/638) and maintaining a constant flow rate by closing HCV-670. None of the valves respond automatically.

Technical Reference(s): OP B-2:V

Proposed references to be provided to applicants during examination: None

Learning Objective: 35319 - State the alignments of the RHR system and components for various operational alignments.

Question Source:
Bank - INPO

Question History: Last NRC Exam: Seabrook 05/03

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No changes to initial submittal.

K/A: 005 A4.03 - Ability to manually operate and/or monitor in the control room: RHR temperature, PZR heaters and flow, and nitrogen (2.8/2.7)

Question 5.

RO Question 32 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	006 K6.02	_____
	Importance Rating:	3.4	3.9

Proposed Question:

What is the potential impact on the analyzed LOCA response if two Accumulators are INOPERABLE?

- A. Discharge of the accumulators occurs at a later time and will interfere with natural circulation core cooling.
- B. Discharge of the accumulators occurs at a later time and could initiate a rapid RCS cooldown and re-initiate break flow.
- C. Insufficient water will be available for reactor core refill which may result in excessive peak cladding temperatures.
- D. Insufficient water will be available in the recirculation sump to provide long-term core cooling ability during a post-LOCA recovery.

Proposed Answer:

C. Insufficient water will be available for reactor core refill which may result in excessive peak cladding temperatures.

Explanation:

A incorrect. Nitrogen injection is not the concern.

B incorrect. Injection will not result in RCS cooldown.

C correct.

From LMCDFRC - Larger LOCAs, which depressurize the RCS quickly, rely on the Accumulators and low head SI to refill the RCS. Without these systems, the core would quickly uncover and remain uncovered until the HPSI system could refill the RCS. An ICC condition may develop during this refilling period.

D incorrect. Most inventory is from the RWST.

Technical Reference(s): LMCDFRC

Proposed references to be provided to applicants during examination: None

Learning Objective: 5459 - Explain why ECCS Accumulators are isolated for various plant conditions.

Question Source:

Bank P-0545

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Expanded justification. Reference taken from Licensed Operator training material as required RO knowledge. No changes to question from initial submittal.

K/A: 006 K6.02 – Knowledge of the effect of a loss or malfunction of the following will have on the ECCS: Core flood tanks (accumulators) (3.4/3.9)

Difficulty taken from bank question.

Question 6.

RO Question 33

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	007 K4.01	
	Importance Rating:	2.6	2.9

Proposed Question:

PK 05-25, PRT Press/Lvl/Temp alarms. The operator reports it is due to high PRT temperature. Level and pressure are within the normal range.

Which of the following describes how PRT temperature will be lowered?

- A. Automatic opening of the Primary Water system supply valve.
- B. Manually opening the Primary Water system supply valve.
- C. Venting the PRT to the Waste Gas Header.
- D. Draining the PRT to the RCDT.

Proposed Answer:

B. Manually opening the Primary Water system supply valve.

Explanation:

A incorrect. The valve does not open automatically.

B correct. Temperature is lowered by manually opening RCS-1-8030 on VB2.

C and D incorrect. Pressure and level are normal - draining or venting is not necessary.

Technical Reference(s): AR PK05-25 - PRT Press/Lvl/Temp

Proposed references to be provided to applicants during examination: none

Learning Objective: 4950 - Explain the operation of PRT system

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No changes from initial submittal.

K/A: 007 K4.01 – Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: Quench tank cooling (2.6/2.9)

Question 7.

RO Question 34 rev1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	008 A1.02	_____
	Importance Rating:	2.9	3.1

Proposed Question:

GIVEN:

- Unit 1 is at 100% power.
- One CCW heat exchanger is in service.
- Containment temperature is 80°F
- 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.

Proposed Answer:

C. Seal Water Heat Exchanger.

Explanation:

A incorrect. AP-11 states the SFP may act as a heat sink and should not be isolated.

B incorrect. Not isolated unless the reactor is tripped.

C correct. Isolated by opening the bypass and closing the inlet and outlet valves. Then CCW is secured to the heat exchanger.

D incorrect. AP-11 states the CFCUs may act as a heat sink and should not be isolated.

Technical Reference(s): AP-11, appendix B.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3466 - Discuss the effects and actions associated with a loss of CCW.

Question Source:
New

Question History: Last NRC Exam: N /A

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: Replaced question to focus on CCW temperature limit and action to prevent exceeding limit.

K/A: 008 A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW temperature (2.9/3.1)

Question 8.

RO Question 35

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	008 A3.10	
	Importance Rating:	2.9	3.0

Proposed Question:

GIVEN:

- The plant is at full power.
- CCW pumps 1-1 and 1-2 are running.
- The AUTO/MANUAL switch for pump 1-1 is in MANUAL
- The AUTO/MANUAL switch for pump 1-2 is in MANUAL
- The AUTO/MANUAL switch for pump 1-3 is in AUTO
- The isolation valve for the lube oil pressure transmitter for CCW pump 1-3 was inadvertently left closed following maintenance.

A reactor trip and Safety Injection occurs.

Which CCW pumps will be running?

- A. All 3 CCW pumps will be running.
- B. Only 1-1 and 1-2.
- C. Only 1-1 and 1-3.
- D. Only 1-1.

Proposed Answer:

- A. All 3 CCW pumps will be running.

Explanation:

A correct. On LOP, the lube oil pressure is not required to start the CCW pump for SI, blocks low pressure start, therefore pump 1-3 will start.

B incorrect. The 1-3 pump will start.

C incorrect. The Auto/Manual switch does not impact the auto start on safety injection.

D incorrect. Pumps will start independent of the AUTO/MANUAL switch.

Technical Reference(s): STG F2, Component Cooling Water.

Proposed references to be provided to applicants during examination: None

Learning Objective: 35487- Analyze the control logic for the CCW system including automatic actions and trips.

Question Source:
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No changes to initial submittal.

K/A: 008 A3.10 - Ability to monitor automatic operation of the CCWS, including: CCW pump instruments and their respective sensors, including flow, pressure, oil level, and discharge temperature (2.9/3.0)

Question 9.

RO Question 36 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u> 2 </u>	<u> </u>
	Group:	<u> 1 </u>	<u> </u>
	K/A:	010 A2.01	
	Importance Rating:	3.3	3.6

Proposed Question:

INITIAL CONDITIONS:

- The plant is at full power
- Pressurizer heater groups 2 and 3 are aligned to their vital power supply
- Control switch on CC-1 for both groups is in AUTO

The following occurs:

- A steam generator tube rupture.
- The crew performs the actions of E-3, Steam Generator Tube Rupture.
- ECCS pumps have been secured and normal letdown and charging established.

At step 35, CONTROL RCS Pressure And Charging Flow To Minimize RCS-To-Secondary Leakage, the operator is to verify Pressurizer heaters ON.

What action, if any, would have to be taken by the crew in the Control Room for Pressurizer heater groups 2 and 3?

- A. None, the heaters energized when SI was reset.
- B. Turn the Control Switch for each group to OFF and back to ON.
- C. Dispatch a Nuclear Operator to locally reset the breakers at the vital buses.
- D. None, the heaters re-energized due to low RCS pressure as soon as Pressurizer level was restored.

Proposed Answer:

- B. Turn the Control Switch for each group to OFF and back to ON.

Explanation:

A incorrect. SI trips the breakers.

B correct. Breaker reset when cycled with the switch, (breaks the SI lockout).

C incorrect. The breakers have not tripped open.

D incorrect. This would be the normal response without SI actuated.

Technical Reference(s): E-3, SGTR, STG A4A, PPLC

Proposed references to be provided to applicants during examination: None

Learning Objective: 36923 - Analyze Pressurizer control logic, including:

- Pressurizer Pressure Control System
- Low Temperature Overpressure Protection (LTOP)
- Pressurizer Level Control System
- Power Operated Relief (PORV)

Question Source:

New

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: No change to question, however, distractors C and D were replaced.

K/A: 010 A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater Failures (3.3/3.6)

Question 10.

RO Question 37

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	010 G2.4.6	_____
	Importance Rating:	3.1	4.0

Proposed Question:

The crew is performing a cooldown and depressurization in accordance with E-0.3, Natural Circulation Cooldown With Steam Void in Vessel (With RVLIS).

Which of the following is the preferred method of depressurizing the RCS?

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

Proposed Answer:

B. Aux Spray

Explanation:

A incorrect. A PORV is the backup if aux spray unavailable.

B correct. Preferred method is to establish letdown and use Aux spray.

C incorrect. No RCPs running to use normal spray.

D incorrect. Not a method of pressure control.

Technical Reference(s): E-0.3, Natural Circulation Cooldown With Steam Void in Vessel (With RVLIS).

Proposed references to be provided to applicants during examination: None

Learning Objective: 7494 - Explain the actions to reduce effects of voiding in the RCS

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No changes from initial submittal.

K/A: 010 G2.4.6 – Pressurizer Pressure Control - Knowledge symptom based EOP mitigation strategies. (3.1/4.0)

Question 11.

RO Question 38

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	012 K1.03	_____
	Importance Rating:	3.7	3.8

Proposed Question:

The plant is at 100% power.

Which of the following failures would cause rods to insert and block both AUTO and MANUAL outward rod motion?

- A. N44 fails high.
- B. PT-505 fails low.
- C. Loop 1 Thot fails high.
- D. Controlling Pressurizer pressure channel fails low.

Proposed Answer:

- A. N44 fails high.

Explanation:

A correct. Rods will move in to the “increase” in reactor power. All outward motion is stopped by C-2 (coincidence 1/4)

B incorrect. Rods will move in, however, only auto outward motion blocked by C-5.

C incorrect, Rods move in, however, no outward rod motion is prevented (OT and OP dt are 2/4 coincidence).

D incorrect, No rod motion, coincidence is 2/4

Technical Reference(s): OIM B-6-3

Proposed references to be provided to applicants during examination: None

Learning Objective: 37048 - Analyze and interpret the logic for Reactor Protection System functions and components, including:

- Reactor Trip Signals
- ESFAS Actuation Signals
- Control Switches
- Permissives

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.2 - General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments: No changes from initial submittal.

K/A: 012 K1.03 – Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems: CRDS (3.7/3.8)

Question 12.

RO Question 39

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	013 K3.01	
	Importance Rating:	4.4	4.7

Proposed Question:

The plant trips from full power due to a loss of feedwater. AFW fails to actuate.

Without operator action, which of the following is the likely consequence of the AFW failure?

- A. Core uncover and overheating.
- B. RCS pressure exceeding design pressure.
- C. Return to criticality due to core boil off causing a loss of boron inventory.
- D. Steam Generator dryout and tube damage from excessive steam generator pressure.

Proposed Answer:

- A. Core uncover and overheating.

Explanation:

A correct. From FR-H.1 background, without operator action, once the secondary is lost as a heat sink, the RCS will heat up to saturation, boil off and result in core uncover and damage.

Technical Reference(s): FR-H.1, background

Proposed references to be provided to applicants during examination: None

Learning Objective: 11319 Describe the loss of feedwater event leading to core damage.

Question Source:

Bank modified M-0068

Question History: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No changes from initial submittal.

K/A: 013 K3.01 - Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel (4.4/4.7)

Question 13.

RO Question 40

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	013 A4.02	
	Importance Rating:	4.3	4.4

Proposed Question:

A steam break downstream of the MSIVs causes SI to actuate.

Current plant conditions:

- The B train RTB is closed.
- RCS pressure is 1900 psig and increasing.
- Pressurizer level is 35% and increasing.
- Narrow range levels in all Steam Generators are 30%.
- Steam Generator pressures are approximately 900 psig.
- The crew has entered E-1.1, SI Termination.

The operator resets both trains of Safety Injection.

Following the reset, what should be the status of PK02-02, SI Initiate and PK08-21, SI Actuate alarms?

- A. Only PK02-02 will be cleared.
- B. Only PK08-21 will be cleared.
- C. Both PK08-21 and PK02-02 will be cleared.
- D. Neither PK08-21 or PK02-02 will be cleared.

Proposed Answer:

- C. Both PK08-21 and PK02-02 will be cleared.

Explanation:

A incorrect. PK08-21 will also be cleared.

B incorrect. PK02-02 will also be cleared.

C correct. With no auto SI signal present, PK02-02 will clear. Going to reset will clear PK08-21. The failure of the RTB will not interfere with resetting SI. Auto SI train B is not blocked (PK08-22 will not acuate).

D incorrect. Both will be cleared.

Technical Reference(s): OIM page B-6-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 37050 - Analyze the indications of the Reactor Protection System including:

- Annunciator Windows
- Status Lights
- Monitor Light Boxes

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No changes from initial submittal.

K/A: 013 A4.02 - Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels (4.3/4.4)

Question 14.

RO Question 41 Rev 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u> 2 </u>	<u> </u>
	Group:	<u> 1 </u>	<u> </u>
	K/A:	022 A2.06	
	Importance Rating:	2.8	3.2

Proposed Question:

GIVEN:

- The plant is at full power.
- CFCUs 1-1, 1-2, 1-3 and 1-4 are running in HIGH speed.
- Containment temperature is 118°F, stable.
- Pressurizer safety loop seal temperatures are 290°F, stable.

CFCU 1-4 trips on overcurrent causing PK01-21, CONTMT FAN CLR, to alarm and the crew enters the alarm response procedure. The operator reports that both Containment temperature and Pressurizer loop seal temperatures are increasing.

Per the alarm response, what action should be taken by the crew?

- A. Restart CFCU 1-4 to prevent Containment temperature exceeding the Technical Specification limit.
- B. Start CFCU 1-5 to prevent Containment temperature exceeding the Technical Specification limit.
- C. Restart CFCU 1-4 to prevent Pressurizer safety loop seal temperature exceeding the Technical Specification limit.
- D. Start CFCU 1-5 to prevent Pressurizer safety loop seal temperature exceeding the Technical Specification limit.

Proposed Answer:

B. Start CFCU 1-5 to prevent Containment temperature exceeding the Technical Specification limit.

Explanation:

A incorrect. The tripped CFCU is not restarted until it has been investigated by maintenance.

B correct. Containment temperature is high, the non-running CFCU should be started to prevent exceeding the containment temperature upper limit.

C incorrect. Tripped CFCU not restarted until investigated by maintenance. Loop seal temperature limit is 221F (low) and 500F (high). Above 320F, there's the possibility there is a leaking safety.

D incorrect. Loop seal temperature limit is 221F (low) and 500F (high). Above 320F, there's the possibility there is a leaking safety. Inoperability due to high temperature, not likely due to strictly containment temperature, would require an increase of over 200F.

Technical Reference(s): AR PK01-21, PK05-23

Proposed references to be provided to applicants during examination: None

Learning Objective: 6149 - Describe the precautions and limitations for CFCUs.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

K/A: 022 A2.06 - Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of CCS pump (2.8/3.2)

Original question did not meet the first half of the KA. This question was rewritten.

Question 15.

RO Question 42 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	026 K4.09	_____
	Importance Rating:	3.7	4.1

Proposed Question:

Which of the following valves is interlocked with opening 9003A, residual heat removal (RHR) to containment spray rings isolation valve?

- A. 8809A, RHR cold leg injection isolation valve.
- B. 9001A, Containment spray pump discharge valve.
- C. 8982A, Containment recirculation sump suction valve.
- D. 9003B, RHR to containment spray rings isolation valve.

Proposed Answer:

C. 8982A, Containment recirculation sump suction valve.

Explanation:

A incorrect. Not part of the interlock for the 9003 valves. Possible the operator would feel that the injection valve would be closed as part of the alignment to provide more RHR flow to Containment spray.

B incorrect. Not part of the interlock for the 9003 valves. Possible the operator would feel the lineup would require isolation of the spray part to prevent runoff.

Only C correct.

9003A will open if both

- RHR-8982A, RHR pump 1 suction from Containment sump is open AND
- RHR-8701 OR -8702, RHR suction from RCS loop 4 is closed

D incorrect. 9003 interlocked with 8701 or 8702 loop suction valve and 8982A/B, Containment suction but not the other trains 9003 valve.

Technical Reference(s): STG I2, Containment Spray

Proposed references to be provided to applicants during examination: None

Learning Objective: 37375 - Analyze the control logic for Containment Spray system equipment, including:

Pumps
Valves

Question Source:

Bank P-0308

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Expanded justification. No other changes to question from initial submittal.

K/A: 026 K4.09 - Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Prevention of path for escape of radioactivity from containment to the outside (interlock on RWST isolation after swapover) (3.7/4.1)

Question 16.

RO Question 43 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	026 A4.01	
	Importance Rating:	4.5	4.3

Proposed Question:

GIVEN:

- A steam break inside Containment has occurred.
- Current Containment pressure is 24 psig.
- SI has not been reset
- One train of Containment Spray is running, one train did not actuate and is not running.

The operator goes to initiate Containment Spray manually, but instead presses both Containment Spray reset pushbuttons.

What affect, if any, did the operator's action have on actuating Containment Spray for the non-running train?

- A. No effect. To manually actuate, take the Phase B switch of the non-running train to Actuate.
- B. No effect. To manually actuate, take both Phase B switches to Actuate.
- C. Manual actuation is blocked until Containment pressure is less than 3 psig.
- D. Manual actuation is blocked until Containment pressure is less than 22 psig.

Proposed Answer:

B. No effect. To manually actuate, take both Phase B switches to Actuate.

Explanation:

B is correct. Reset will reset auto actuation (and manual) even with Containment pressure above the setpoint. Manual actuation is still available, however, both switches are required.

Technical Reference(s): STG B6A, RPS

Proposed references to be provided to applicants during examination: None

Learning Objective: 6017 - Explain the operation of the Containment Spray system

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: Modified last bullet concerning train status, no other changes from initial submittal.

K/A: 026 A4.01 - Ability to manually operate and/or monitor in the control room: CSS controls (4.5/4.3)

Question 17.

RO Question 44

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u> 2 </u>	<u> </u>
	Group:	<u> 1 </u>	<u> </u>
	K/A:	039 A1.10	<u> </u>
	Importance Rating:	2.9	3.0

Proposed Question:

Which of the following conditions would make the setpoint for the Steam Jet Air Ejector (SJAE) radiation monitor, RE-15, MORE conservative (alarm at a lower leak rate)?

- A. INCREASING SJAE Exhaust Flowrate or INCREASING RCS Noble Gas Concentration.
- B. INCREASING SJAE Exhaust Flowrate or DECREASING RCS Noble Gas Concentration.
- C. DECREASING SJAE Exhaust Flowrate or DECREASING RCS Noble Gas Concentration
- D. DECREASING SJAE Exhaust Flowrate or INCREASING RCS Noble Gas Concentration.

Proposed Answer:

D. DECREASING SJAE Exhaust Flowrate or INCREASING RCS Noble Gas Concentration.

Explanation:

A incorrect. Decreasing SJAE flow not increasing.

B incorrect. Decreasing SJAE flow and Increasing Noble gas.

C incorrect. Increasing noble gas.

D correct. Per O-4, increasing RCS Noble Gas concentration or decreasing SJAE flow makes the calculated setpoint more conservative.

Technical Reference(s): O-4, Primary to Secondary Steam Generator Tube Leak Detection

Proposed references to be provided to applicants during examination: O-4

Learning Objective: 37875 - Describe the Radiation Monitoring System parameters, indications, and alarms.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.11 - Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments: No changes from initial submittal.

K/A: 039 A1.10 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Air ejector PRM (2.9/3.0)

Question 18.

RO Question 45 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	039 G2.4.31	_____
	Importance Rating:	3.3	3.4

Proposed Question:

GIVEN:

- The plant is in MODE 3.
- RCS temperature is 485°F.
- RCS pressure is 1830 psig.

RCS pressure increases to 1850 psig and PK06-06, RCS - S/G PRESS MISMATCH is received in the Control Room.

What is the significance of this alarm?

- A. An MSI is about to occur.
- B. A sudden increase in steam flow has occurred.
- C. Plant conditions are approaching SI actuation on low steam pressure.
- D. Delta P across the Steam Generator U-tubes is close to the design limit.

Proposed Answer:

- C. Plant conditions are approaching SI actuation on low steam pressure.

Explanation:

C is correct. Alarm is at RCS pressure greater than 1850 psig and steam generators are less than 630 psig. If RCS pressure is allowed to increase above P-11, SI on low steam pressure would actuate.

Tsat for 485°F is about 590 psia (575 psig).

Technical Reference(s): AR PK06-06

Proposed references to be provided to applicants during examination: None

Learning Objective: 37593 - Describe the Main Steam system parameters, indications, and alarms.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Formatted initial conditions to bullets. No other changes.

K/A: 039 G2.4.31 – Main and Reheat Steam - Knowledge of annunciators alarms and indications, and use of the response instructions. (3.3/3.4)

Question 19.

RO Question 46

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	059 K1.05	_____
	Importance Rating:	3.1	3.2

Proposed Question:

GIVEN:

- Reactor power is 75%
- Feedwater control is in AUTOMATIC

Which of the following will cause RCS temperature to initially INCREASE?

- A. A running MFP trips.
- B. A 10% steam dump valve fails open.
- C. A steam generator narrow range level transmitter fails high.
- D. A steam generator main feedwater reg valve fails full open.

Proposed Answer:

- A. A running MFP trips.

Explanation:

A correct. Loss of feed causes a runback. Additionally, loss of the MFP initially will reduce feed to all S/Gs and cause RCS to increase.

B incorrect. Increased steam flow will cause temperature to decrease.

C incorrect. Control system will disregard a failed channel.

D incorrect. A large increase in feed to the steam generator occurs. This will cause RCS temperature to decrease.

Technical Reference(s): LPA-15

Proposed references to be provided to applicants during examination: None

Learning Objective: 7380 - Explain plant response to loss of feedwater

Question Source:

New

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: No changes to initial submittal.

K/A: 059 K1.05 - Knowledge of the physical connections and/or cause effect relationships between the MFW and the following systems: RCS (3.1/3.2)

Question 20.

RO Question 47

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	059 G2.1.27	_____
	Importance Rating:	2.8	2.9

Proposed Question:

A purpose of the Digital Feedwater Control System is to maintain which of the following?

- A. A constant Main Feed Pump speed for all power levels.
- B. A constant mass in the steam generators for all power levels.
- C. Inventory in the steam generators such that the U-tubes remain covered following a reactor trip.
- D. Steam generator water level within acceptable levels during normal unit transients.

Proposed Answer:

D. Steam generator water level within acceptable levels during normal unit transients.

Explanation:

A incorrect, MFP speed changes as power level changes.

B incorrect, mass in the generator changes greatly from 0 to 100%.

C incorrect, level is not above the U-tubes following a trip (from high power).

D correct, the purpose of DFWCS is to automatically maintain a programmed steam generator water level during steady-state operation and to restore and maintain the water level within acceptable levels during normal unit transients

Technical Reference(s): STG C8B, Digital Feedwater Control

Proposed references to be provided to applicants during examination: None

Learning Objective: 70366, State the purpose of the Main Feedwater Control System

Question Source:

Bank N-73157 modified

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No changes to initial submittal.

K/A: 059 G2.1.27 – MFW – Knowledge of system purpose and or function. (2.8/2.9)

Question 21.

RO Question 48

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	061 K5.03	
	Importance Rating:	2.6	2.9

Proposed Question:

Unit 1 trips from low power.

The operator has stopped flow to the Steam Generators from the MDAFW pumps by closing LCV-110, LCV-111, LCV-113 and LCV-115.

Which of the following describes the effect of closing the LCVs on pump head and volumetric flow rate for the MDAFW pumps?

- A. Pump head INCREASES and volumetric flowrate decreases to zero.
- B. Pump head INCREASES and volumetric flowrate decreases but remains greater than zero, due to recirc flow to the CST.
- C. Pump head DECREASES and volumetric flowrate decreases to zero.
- D. Pump head DECREASES and volumetric flowrate decreases but remains greater than zero, due to recirc flow to the CST.

Proposed Answer:

B. Pump head INCREASES and volumetric flowrate decreases but remains greater than zero, due to recirc flow to the CST.

Explanation:

A incorrect. Recirc maintains some flow.

B correct. With the discharge valves closed, head loss increases, which translates to an increase in pump head Flow thru the pump decreases but remains greater than zero due to recirc flow back to the CST.

C incorrect. Pump head increases.

D incorrect. Pump head increases.

Technical Reference(s): TH11 – Pump Theory, STG D1, Auxiliary Feedwater

Proposed references to be provided to applicants during examination: None

Learning Objective: 65911 - Draw, label, and explain the characteristic curves for centrifugal pumps, including the effects of system changes on pump operation.

Question Source:
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: No changes to initial submittal.

K/A: 061 K5.03 - Knowledge of the operational implications of the following concepts as they apply to the AFW: Pump head effects when control valve is shut (2.6/2.9)

Question 22.

RO Question 49

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u> 2 </u>	<u> </u>
	Group:	<u> 1 </u>	<u> </u>
	K/A:	062 K2.01	
	Importance Rating:	3.3	3.4

Proposed Question:

Unit 2 trips and Safety Injection actuates. A loss of offsite power results in the emergency diesels powering their respective Vital Buses.

Which of the following describes how RHR pump 2-1 is being powered at this time?

- A. Diesel 2-2 to Vital Bus G.
- B. Diesel 2-1 to Vital Bus G.
- C. Diesel 2-3 to Vital Bus F.
- D. Diesel 2-1 to Vital Bus F.

Proposed Answer:

- B. Diesel 2-1 to Vital Bus G.

Explanation:

Diesels supply power to the following vital buses:

- 1-1 Vital Bus H
- 1-2 Vital Bus G
- 1-3 Vital Bus F

- 2-1 Vital Bus G
- 2-2 Vital Bus H
- 2-3 Vital Bus F

RHR pumps are powered from Bus G (RHR 1-1 or 2-1) and Bus H (RHR 1-2 or 2-2).

A incorrect. Diesel 2-1 powers bus G.

B correct. RHR 2-1 is powered from Bus G, from emergency diesel 2-1.

C and D incorrect. This would be a logical assumption that the 2-1 RHR pump would come from the F (or A train) bus.

Technical Reference(s): OIM J-1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 7011 - State the power supply to the RHR pumps

Question Source:
Bank – P-0648 modified

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.8 - Components, capacity, and functions of emergency systems.

Comments: No changes to initial submittal.

K/A: 062 K2.01 – Knowledge of bus power supplies to the following: Major System Loads (3.3/3.4)

Question 23.

RO Question 50

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	063 K4.04	_____
	Importance Rating:	2.6	2.9

Proposed Question:

Which of the following describes the effect on a closed Circulating Water pump breaker if DC control power is lost to the breaker?

- A. The breaker immediately trips open and cannot be reclosed until control power is restored.
- B. The breaker can be tripped from the Control Room but automatic trip functions are not operable.
- C. Automatic trips are not operable and tripping the breaker from the Control Room is not possible.
- D. Automatic breaker trips are operable but tripping the breaker from the Control Room is not possible.

Proposed Answer:

C. Automatic trips are not operable and tripping the breaker from the Control Room is not possible.

Explanation:

A incorrect. The breaker will not open.

B incorrect. The breaker cannot be tripped remotely.

C correct. Only local operation is possible. The breaker cannot be operated from the control room.

D incorrect. Automatic trips are defeated.

Technical Reference(s): STG J9, DC Power

Proposed references to be provided to applicants during examination: None

Learning Objective: 5193 - Explain the effect of a loss of DC Power system on plant systems, instrumentation and controls.

Question Source:

Bank P-1514 modified

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.6 - Design, components, and function of reactivity control mechanisms and instrumentation.

Comments: No changes to initial submittal.

K/A: 063 K4.04 - Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following: Trips (2.6/2.9)

Question 24.

RO Question 51

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	064 K6.07	_____
	Importance Rating:	2.7	2.9

Proposed Question:

GIVEN:

- Unit 1 is at 100% power.
- A relief valve has failed open on Starting Air Receiver “A” for Diesel Generator 1-1.
- The leakage exceeds the capacity of the starting air compressor.
- The NO is responding to the associated alarm.

Which of the following describes the response of Diesel Generator 1-1 if a start signal occurs before any operator action?

- A. It will start in the normal time from Starting Air Receiver “B” via all four starting air solenoids.
- B. It will start in the normal time via the two starting air solenoids associated with the Starting Air Receiver “B”.
- C. It will take longer to start or may NOT start because the starting air system will be depressurizing/depressurized.
- D. It will take longer to start or may NOT start because the fuel rack fails to no fuel position and may not reset.

Proposed Answer:

B. It will start in the normal time via the two starting air solenoids associated with the Starting Air Receiver “B”.

Explanation:

A incorrect. Only two solenoids associated with B receiver.

B correct. The B receiver is capable of starting the diesel by opening the two solenoid valves for the air motors.

C incorrect. Receivers are not cross tied.

D incorrect. Loss of one receiver will not affect operation.

Technical Reference(s): STG 6B, Diesel Generators

Proposed references to be provided to applicants during examination: None

Learning Objective: 6431 - State the purpose of D/G subsystems and components.

Question Source:

Bank - INPO

Question History: Last NRC Exam: Salem 11/02

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.8 - Components, capacity, and functions of emergency systems.

Comments: No changes to initial submittal.

K/A: 064 K6.07 - Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers (2.7/2.9)

Question 25.

RO Question 52 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	073 K3.01	_____
	Importance Rating:	3.6	4.2

Proposed Question:

GIVEN:

- Reactor power is 45%.
- Circulating Water pump 1-1 is running. 1-2 is shutdown.
- A liquid radwaste discharge permit has been authorized.
- RE-18, Liquid Radwaste Monitor, is determined to be inoperable when adjusting the setpoint.

Which of the following actions should be taken prior to initiating the liquid radwaste discharge?

- A. Start Circulating Water pump 1-2.
- B. Install a temporary radiation monitor.
- C. Station an operator at RCV-18 to close the valve if necessary.
- D. Have a second independent sample analyzed and release rate calculations verified.

Proposed Answer:

- D. Have a second independent sample analyzed and release rate calculations verified.

Explanation:

A incorrect. Flow from one CWP is adequate and the discharge rate is adjusted accordingly.

B incorrect. Temporary monitor is not required.

C incorrect. No additional operators are required. The sample will be reanalyzed to assure its discharge is allowed.

D correct. Per G-1:II if RE-18 is out of service, a second sample must be analyzed and release rates verified by at least 2 staff members.

Technical Reference(s): OP G-1:II

Proposed references to be provided to applicants during examination: None

Learning Objective: 8443 - State the administrative requirements of Liquid Rad Waste system

Question Source:
Bank P-1245 modified

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
 55.43 _____

Comments: Expanded justification. No other changes to initial submittal.

K/A: 073 K3.01 - Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases (3.6/4.2)

Question 26.

RO Question 53 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	076 K2.01	
	Importance Rating:	2.7	2.7

Proposed Question:

What are the power supplies for the Unit 2 Auxiliary Saltwater Pumps, 2-1 and 2-2?

- | | <u>Pump 2-1</u> | <u>Pump 2-2</u> |
|----|-----------------|-----------------|
| A. | Bus F | Bus G |
| B. | Bus G | Bus H |
| C. | Bus F | Bus H |
| D. | Bus G | Bus F |

Proposed Answer:

- A. Bus F Bus G

Explanation:

- A correct. Pumps 1 and 2 are powered from Bus F and G respectively for both units.
B incorrect. Pump 2-1 is powered from F, 2-2 powered from G.
C incorrect. Pump 2-2 powered from G.
D incorrect. Pump 2-1 powered from F.

Technical Reference(s): STG E5, ASW

Proposed references to be provided to applicants during examination: None

Learning Objective: 5339 - State the power supply to the ASW Pumps.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Expanded justification. No other changes.

K/A: 076 K2.01 - Knowledge of bus power supplies to the following: Service water (2.7/2.7)

Question 27.

RO Question 54

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	078 A3.01	
	Importance Rating:	3.1	3.2

Proposed Question:

Instrument air is in a normal system lineup. All Instrument Air compressors are available.

Instrument air header pressure now indicates 89 psig.

What should be the status of the instrument air compressors?

- A. All compressors running and loaded.
- B. Only Rotary compressors 0-5 and 0-6 running and loaded
- C. Only Reciprocating compressors 0-1 through 0-4 running and loaded
- D. All compressors running but only Rotary compressors 0-5 and 0-6 loaded.

Proposed Answer:

- A. All compressors running and loaded.

Explanation:

A correct. Compressors 5 and 6 should start at 103 psig. 1 thru 4 start below 93 psig. All compressors should be running, loaded.

B incorrect. 1 thru 4 started at 93 psig.

C incorrect. 5 and 6 start at 103 psig.

D incorrect. All will be running and loaded.

Technical Reference(s): STG K1, Compressed Air

Proposed references to be provided to applicants during examination: None

Learning Objective: 68959 - Describe Automatic operations of the Compressed Air System.

Question Source:
Bank S-1254

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41. Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No changes.

K/A: 078 A3.01 - Ability to monitor automatic operation of the IAS, including: Air pressure (3.1/3.2)

Question 28.

RO Question 55 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>1</u>	_____
	K/A:	103 K3.01	
	Importance Rating:	3.3	3.7

Proposed Question:

Core offload is in progress.

Which of the following would require suspension of the core offload?

- A. Both personal air lock doors are open.
- B. Half the bolts are removed from the equipment hatch.
- C. Containment Purge Exhaust Fan, E-3, trips.
- D. Containment Purge Exhaust valves RCV-11 and RCV-12 fail to fully close when the system is secured.

Proposed Answer:

D. Containment Purge Exhaust valves RCV-11 and RCV-12 fail to fully close when the system is secured.

Explanation:

A incorrect, the doors may be open but must be capable of being closed.

B incorrect, removal of half the bolts would leave many more than the 4 required.

C incorrect, tripping of the purge exhaust is not a cause for stopping defueling.

D correct, not capable of being isolated.

Technical Reference(s): TS 3.9.4

Proposed references to be provided to applicants during examination: TS 3.9.4

Learning Objective: 96971 - Identify 3.9 Technical Specification LCOs

Question Source:

Bank - INPO modified

Question History: Last NRC Exam: DCPD 10/02

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Modified distractor C to be another possibility that is not a local indication.

K/A: 103 K3.01 – Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under shutdown conditions (3.3/3.7)

Question 29.

RO Question 56 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	001 K3.02	_____
	Importance Rating:	3.4	3.5

Proposed Question:

Unit 1 is exactly critical just below the point of adding heat when a single control rod drops into the core.

Assuming no operator or automatic actions occur, when the plant stabilizes, Intermediate Range indication will be _____ and average reactor coolant temperature will be _____.

- A. lower, the same
- B. lower, lower
- C. the same, the same
- D. the same, lower

Proposed Answer:

- A. lower, the same

Explanation:

A correct. Below the POAH, unlike at power, no temperature feedbacks to affect temperature. Temperature as set by the steam dumps will remain the same. The dropped rod will cause power to decrease.

B incorrect. Temperature will not change.

C incorrect. Power will decrease.

D incorrect. Power will decrease.

Technical Reference(s): LPA12

Proposed references to be provided to applicants during examination: None.

Learning Objective: 5024 - Explain the effect of dropped rod(s) on reactor operation.

Question Source:

Bank – F-40627

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.1 - Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Comments: changed “power” in question to “Intermediate Range indication” based on editorial comment.

K/A: 001 K3.02 - Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: RCS (3.4/3.5)

Question 30.

RO Question 57

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	002 K6.07	_____
	Importance Rating:	2.5	2.8

Proposed Question:

Unit 2 is at 25% power.

RCP 2-1 breaker trips open. The reactor does not trip.

Which of the following describes the response of Thot and Tcold and flow in Loop 2-1?

- A. Flow goes to zero. Tcold increases to approximately Thot.
- B. Flow goes to zero. Thot decreases to approximately Tcold.
- C. Flow goes to approximately 30%. Tcold increases to approximately Thot.
- D. Flow goes to approximately 30%. Thot decreases to approximately Tcold.

Proposed Answer:

D. Flow goes to approximately 30%. Thot decreases to approximately Tcold.

Explanation:

A and B incorrect. Flow reverses and goes to approximately 30%.

C incorrect, reverse flow causes loop temperature to go to Tcold not Thot.

D correct, flow reverses and will indicate approximately 30%. Loop temperature goes to Tcold.

Technical Reference(s): TH18T

Proposed references to be provided to applicants during examination: None

Learning Objective: 35742 Identify the basic interrelationships between the RCP system and other systems.

Question Source:

Bank P-72907, modified

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.2 - General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments: No changes.

K/A: 002 K6.07 – Knowledge of the effect or a loss or malfunction on the following RCS components: Pumps (2.5/2.8)

Question 31.

RO Question 58 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	011 K2.02	_____
	Importance Rating:	3.1	3.2

Proposed Question:

Unit 1 is at 50% power in a normal power lineup.

The feeder breaker to 480 bus 13E trips. No operator action has been taken.

What effect does the breaker trip have on the Pressurizer heaters?

- A. Heater Groups 1 and 2 are available.
- B. Heater Groups 2 and 3 are available.
- C. Heater Groups 2 and 4 are available.
- D. Heater Groups 3 and 4 are available.

Proposed Answer:

- A. Heater Groups 1 and 2 are available.

Explanation:

A correct. Power supply for Groups 1 and 2 is Bus 13D. Power supply for Groups 3 and 4 is 13E. If 13E trips, only Groups 1 and 2 are available.

B incorrect. 3 powered from 13E.

C incorrect. 4 powered from 13E.

D incorrect. 3 and 4 powered from 13E.

Technical Reference(s): STG 4A, PPLC

Proposed references to be provided to applicants during examination: None

Learning Objective: 9990 - Identify the power supply for RCS Pressurizer Pressure Control system major components

- Pressurizer heaters
- PORV Block Valves

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Modified answer and distractors for readability.

K/A: 011 K2.02 – Knowledge of bus power supplies to the following: PZR Heaters (3.1/3.2)

Question 32.

RO Question 59 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	016 A4.02	_____
	Importance Rating:	2.7	2.6

Proposed Question:

In accordance with OP AP-20, "CONDENSER TUBE LEAK," which of the following parameters is an initial indication of demineralizer (polisher) system breakthrough?

- A. PK12-10, S/G BLOWDOWN CATION CONDT'Y HI alarm in.
- B. PK12-14, COND TRAY/TUBE SKT LEAK DET CONDT'Y HI alarm in.
- C. Condensate pump cation conductivity >200 $\mu\text{S}/\text{cm}$ - PT 19 on Tracor Westronics Secondary Chemistry Recorder.
- D. Feedwater cation conductivity >0.25 $\mu\text{S}/\text{cm}$ - PT 32 on Tracor Westronics Secondary Chemistry Recorder.

Proposed Answer:

D. Feedwater cation conductivity >0.25 $\mu\text{S}/\text{cm}$ - PT 32 on Tracor Westronics Secondary Chemistry Recorder.

Explanation:

Only D correct.

Step 4 states the following:

Monitor for Condensate Demin Breakthrough:

Feedwater Cation Conductivity, PT 32 on Secondary Chemistry Recorder, is LESS THAN 0.25 $\mu\text{S}/\text{cm}$

Technical Reference(s): AP-20, Condenser Tube Leak

Proposed references to be provided to applicants during examination: None

Learning Objective: 4976 State the actions for increasing condensate cation conductivity

Question Source:
Bank P-5506

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: added "initial" (initial indication..) to question to make sure candidate does not draw the problem out so long that A could be argued as correct.

K/A: 016 A4.02 - Ability to manually operate and/or monitor in the control room:
Recorders (2.7/2.6)

Question 33.

RO Question 60

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	033 A3.02	_____
	Importance Rating:	2.9	3.1

Proposed Question:

The design of the Spent Fuel Pool will prevent inadvertent draining the pool below which of the following elevations?

- A. 127 feet
- B. 129 feet
- C. 131 feet
- D. 133 feet

Proposed Answer:

D. 133 feet

Explanation:

D correct. According to section 4.0 DESIGN FEATURES

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133 ft.

Technical Reference(s): Technical Specifications section 4.0, section 4.3.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 35694 - Describe the operation of the Spent Fuel Pool Cooling system.

- Make-up
- Normal alignments
- Refueling alignments
- Malfunctions

Question Source:

DCPP Bank R-55487

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No changes.

K/A: 033 A3.02 – Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Spent fuel leak or rupture (2.9/3.1)

Question 34.

RO Question 61

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	034 K4.02	_____
	Importance Rating:	2.5	3.3

Proposed Question:

Which of the following interlocks can be bypassed at the Spent Fuel Pool Bridge Crane pendant?

- A. Block at motor
- B. Hoist full down
- C. Hoist underload
- D. Upper slow zone

Proposed Answer:

C. Hoist underload

Explanation:

A incorrect, not bypassable.

B incorrect, GEMCO height limit switches (full up and down) can be bypassed using the push-button at the GEMCO box.

C correct, bypassed on hoist pendant.

D incorrect, not bypassable.

Technical Reference(s): STG B8, Fuel Handling Equipment.

Proposed references to be provided to applicants during examination: None

Learning Objective: 36958 - Explain the operation of the SFP Bridge Crane including control/switch logic and indications.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: determined to be appropriate RO knowledge. Operation of SFP bridge is not a task of only SROs. Frequently Nuclear Operators are involved in moving spent fuel to support upcoming refuelings. Additionally, this knowledge item is testable RO systems knowledge. No changes made to question.

K/A: 034 K4.02 – (FHES) Knowledge of design feature(s) and/or interlock(s) which provide for the following: Fuel Movement (2.5/3.3)

Question 35.

RO Question 62 rev1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	035 A2.04	
	Importance Rating:	3.6	3.8

Proposed Question:

Unit 1 is at 100% power.

The following events occur:

- PK09-02, SG 1-2 PRESS, LVL, FLOW alarms.
- Steam flow from the 1-2 Steam Generator has increased above feed flow
- Steam Generator 1-2 level is 50% and increasing
- Reactor power is increasing
- Turbine load is stable

Which of the following states the probable failure and the action that should be taken by the crew?

- A. A Steam Generator Safety has opened, trip the reactor.
- B. A Steam Generator Safety has opened, reduce turbine load as necessary to maintain reactor power less than 100% while taking corrective action.
- C. A steam flow channel has failed high, go to AP-5, Malfunction of Eagle 21 Protection or Control Channel.
- D. A feed flow channel has failed low, go to AP-5, Malfunction of Eagle 21 Protection or Control Channel.

Proposed Answer:

- A. A Steam Generator Safety has opened, trip the reactor.

Explanation:

A correct. Given the plant conditions, an increase in steam flow has occurred. A steam generator safety passes sufficient steam flow to cause a steam/feed flow mismatch and swell due to increased steam flow. If a safety has opened at power, it is below the setpoint where it should be closed. Per AR PK09-02, if a steam generator safety is open, the action is to trip the reactor.

B incorrect, the action to take is trip the reactor, (it cannot be isolated).

C incorrect, no channel failure has occurred (reactor power and steam flow has increased).

D incorrect, no channel failure has occurred (reactor power and steam flow increasing)

Technical Reference(s): AR PK09-02, SG 1-2 PRESS, LVL, FLOW

Proposed references to be provided to applicants during examination: None

Learning Objective: 7158 - Identify alarm indications in the control room applicable to the steam generator system

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: question modified to address both parts of the KA.

K/A: 035 A2.04 – Ability to (a) predict the impacts of the following malfunctions or operations on the SGS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam flow/feed flow mismatch (3.6/3.8)

Question 36.

RO Question 63

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	045 K5.17	_____
	Importance Rating:	2.5	2.7

Proposed Question:

GIVEN:

- The plant is at 90% power
- Control Rods are at 228 steps
- 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.

Proposed Answer:

- A. The RCS temperature change would be smaller.

Explanation:

A correct. As load increases, temperature will decrease to add positive reactivity. At BOL, a small MTC will add less reactivity per degree change than at EOL. The temperature change at EOL will be less (more reactivity added per °F).

B incorrect. The power change is the same, the amount of positive reactivity that must be added by MTC will be the same (but less temperature change at EOL).

C incorrect. The power change will be the same. 50 MWe at BOL is about a 4.5% change (based on 1100 MWe), this is the same at EOL.

D incorrect. More power defect at EOL (4.5% change at EOL (2200 pcm) is about 100 pcm but only about 63 at BOL (1400 pcm)).

Technical Reference(s): R17-1F-3

Proposed references to be provided to applicants during examination: None

Learning Objective: 65587 - Describe the time effect of core age, moderator temperature and boron concentration on the moderator temperature coefficient.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.1 - Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Comments: No changes from initial submittal.

K/A: 045 K5.17 - Knowledge of the operational implications of the following concepts as they apply to the MT/B System: Relationship between moderator temperature coefficient and boron concentration in RCS as T/G load increases (2.5/2.7)

Question 37.

RO Question 64

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	068 G2.4.49	_____
	Importance Rating:	4.0	4.0

Proposed Question:

A Floor Drain Receiver is being discharged to the ocean when a reactor trip occurs.

The discharge:

- A. must be immediately secured.
- B. may continue unchanged.
- C. must immediately be reduced.
- D. must be diverted to the discharge of the other unit.

Proposed Answer:

- A. must be immediately secured.

Explanation:

A correct. Aux Building secures discharges starting from the Aux Building.

B, C and D incorrect. The discharge must be terminated.

Technical Reference(s): Operating Order O-19 Nuclear Operator Actions Following a Reactor Trip

Proposed references to be provided to applicants during examination: None

Learning Objective: 69248 - Explain Liquid Radwaste System local operator actions necessary to support emergency operating procedures.

Question Source:

Bank NS-54684

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.13 - Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments: No changes to initial submittal.

K/A: 068 G2.4.49 – Liquid Radwaste – Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (4.0/4.0)

Question 38.

RO Question 65

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>2</u>	_____
	Group:	<u>2</u>	_____
	K/A:	071 A1.06	_____
	Importance Rating:	2.5	2.8

Proposed Question:

Which of the following actions will automatically occur if the Gaseous Radwaste Discharge Header radiation monitor (RE-22) detects high radiation?

- A. PCV-94, 95 and 96 (GDT'S fill valves) will close.
- B. FCV-407, 408, 409 (GDT'S purge outlet header isolations) will close.
- C. RCV-17 (GDT'S to Plant Vent) will close.
- D. FCV-410 (GDT'S Outlet Header to Plant Vent) will close.

Proposed Answer:

C. RCV-17 (GDT'S to Plant Vent) will close.

Explanation:

- A incorrect. No auto closure on high radiation.
- B incorrect. No auto closure on high radiation.
- C correct, only RCV-17 closes (and is prevented from opening).
- D incorrect. No auto closure

Technical Reference(s): STG G2, Gaseous Radwaste

Proposed references to be provided to applicants during examination: None

Learning Objective: 37706 Describe the operation of the gaseous radwaste vent valves, including:

- Control switches
- Interlocks and logic
- Automatic actions

Question Source:
Bank R-52308

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.11 - Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments: No changes to initial submittal.

K/A: 071 A1.06 – Ability to predict and/or monitor changes in parameters(to prevent exceeding design limits) associated with Waste Gas Disposal System operating the controls including: Ventilation System (2.5/2.8)

Question 39.

RO Question 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	EPE 007 EK2.03	
	Importance Rating:	3.5	3.6

Proposed Question:

Unit 1 is at 6% power.

Which of the following would be the minimum conditions to cause PK04-12, Reactor Trip Initiate, to alarm?

- A. A single Pressurizer Pressure low bistable trips.
- B. Two Pressurizer Pressure low bistables trip.
- C. One Pressurizer Pressure channel fails high.
- D. Two Pressurizer Pressure channels fail high.

Proposed Answer:

D. Two Pressurizer Pressure channels fail high.

Explanation:

A incorrect, this would cause the Protect Channel Actuate alarm.

B incorrect, trip is blocked below 10%

C incorrect, this would cause the Protect Channel Actuate alarm.

D correct, logic is 2 of 4 and not blocked below 10%.

Technical Reference(s): PK04-12

Proposed references to be provided to applicants during examination: None

Learning Objective: 37050 - Analyze the indications of the Reactor Protection System including:

- Annunciator Windows
- Status Lights
- Monitor Light Boxes

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No change from initial submittal

K/A: EPE 007 EK2.03- Knowledge of the interrelations between a reactor trip and the following: Reactor trip status panel (3.5/3.6)

Question 40.

RO Question 2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	_____
	Group:	<u>1</u>	_____
	K/A:	APE 008 AA1.02	
	Importance Rating:	4.1	3.9

Proposed Question:

GIVEN:

- A Pressurizer PORV is partially open.
- The crew is performing the ECCS flow reduction actions of E-1.2, Post-LOCA Cooldown and Depressurization
- All Charging and SI pumps are running
- The RHR pumps are secured
- RCS subcooling is 100°F
- RCS temperature is 360°F
- Pressurizer level is 55%
- Conditions are met for securing one of the running Charging pumps

When the Charging pump is secured, RCS pressure begins to rapidly decrease.

Which of the following actions should be taken?

- A. Restart the RHR pumps.
- B. Immediately restart the Charging pump
- C. Proceed to the next step to determine if an SI pump can be secured.
- D. Wait for RCS pressure to stabilize or increase before taking any action.

Proposed Answer:

- D. Wait for RCS pressure to stabilize or increase before taking any action.

Explanation:

A incorrect, RHR pumps are restarted if subcooling is low when checking reduction criteria.

B incorrect, note states pressure should be allowed to stabilize or increase.

C incorrect, a period of time should pass to ensure RCS pressure will stabilize.

D correct - E-1.2 NOTE - After stopping any ECCS Pp, RCS Pressure should be allowed to stabilize or increase before stopping another ECCS Pp.

From E-1.2 background: After an SI pump is stopped, RCS pressure may decrease rapidly to a new equilibrium value where the reduced SI flow again matches leakage from the RCS. The criteria for stopping the next SI pump has been calculated assuming steady-state conditions. Hence, to ensure that these criteria are appropriate, RCS pressure and subcooling should be allowed to stabilize or increase before additional SI pumps are stopped.

Technical Reference(s): E-1.2, Post-LOCA Cooldown and Depressurization and background

Proposed references to be provided to applicants during examination: None

Learning Objective: 6743 - Explain PZR response during ECCS reduction sequence

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: No change from initial submittal

K/A: APE 008 AA1.02 - Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: HPI pump to control PZR level/pressure (4.1/3.9)

Question 41.

RO Question 3 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	EPE 009 EK3.26	
	Importance Rating:	4.4	4.5

Proposed Question:

The crew is performing ECCS reduction in E-1.2, Post LOCA Cooldown and Depressurization. One SI pump has been stopped.

Which of the following describes how the subcooling required changes and the basis for the amount of subcooling required for securing the second SI pump?

- A. Subcooling required increases to compensate for the anticipated void formation after the second pump is stopped.
- B. Subcooling required increases to ensure the RCS will remain subcooled after the second SI pump is stopped.
- C. Subcooling required decreases due to decreased break flow after stopping the second pump.
- D. Subcooling required decreases to allow pressure to decrease sufficiently after stopping the second pump to place RHR cooling in service.

Proposed Answer:

B. Subcooling required increases to ensure the RCS will remain subcooled after the second SI pump is stopped.

Explanation:

- A. incorrect. No void formation is expected or should exist because the RCS is subcooled.
- B. Correct. because throttling SI flow is not practical, pumps must be stopped. As pumps are stopped, a step decrease in flow is experienced. RCS pressure will decrease to reach equilibrium with break flow and ECCS flow. RCS subcooling and pressure control must be maintained as the reduction takes place. As flow is reduced, a large allowance is needed.
- C. Incorrect. Subcooling requirement increases.
- D. Incorrect subcooling requirement increases.

Technical Reference(s): E-1.2 background

Proposed references to be provided to applicants during examination: None

Learning Objective: 5770 - State the basis for required subcooling margin during accident conditions

Question Source:
Bank P-33554 modified

Question History: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: question modified to focus on the reason for maintaining subcooling.
Modified justification for distractor A.

K/A: EPE 009 EK3.26 - Knowledge of the reasons for the following responses as they apply to the small break LOCA: Maintenance of RCS subcooling (4.4/4.5)

Question 42.

RO Question 4

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	APE 015/017	AK1.01
	Importance Rating:	4.4	4.6

Proposed Question:

The plant is tripped from full power due to a loss of Component Cooling Water. The crew performs all appropriate actions and transitions to E-0.1, Reactor Trip Response while continuing to perform the actions of AP-11, Malfunction of the Component Cooling Water System.

Ten minutes after the trip, the following conditions exist:

- SG Pressures are all approximately 1000 psig and stable
- RCS pressure is 2230 psig and stable.
- That is approximately 575°F in all loops and slowly lowering.
- Core Exit TC's indicate approximately 580°F and stable.
- Tcold is approximately 543°F in all loops and stable.

Which of the following describes the status of RCS heat removal for the current plant conditions?

- A. Heat removal via natural circulation is not established at this time.
- B. Heat removal is via forced circulation and the condenser steam dumps.
- C. Heat removal is via natural circulation and the condenser steam dumps.
- D. Heat removal is via natural circulation and both the condenser and 10% atmospheric steam dumps.

Proposed Answer:

- C. Heat removal is via natural circulation and the condenser steam dumps.

Explanation:

A incorrect. For current plant conditions, all requirements are met.

B incorrect. RCPs are tripped in AP-11.

C correct. Per E-0.1 for natural circulation.

1) RCS subcooling based on core exit T/Cs GREATER THAN 20°F.

2) S/G Pressures stable or decreasing.

3) RCS Hot Leg Temperatures stable or decreasing.

4) Core Exit T/Cs stable or decreasing.

5) RCS Cold Leg Temperatures at saturation temperature for S/G Pressure.

The MSIVs should still be open, therefore, the condenser steam dumps are providing heat removal.

D incorrect. 10% steam dump setpoint is 1020 psig, therefore, they will be closed.

Technical Reference(s): E-0.1 and steam tables.

Proposed references to be provided to applicants during examination: steam tables

Learning Objective: 5810 - Interpret indication of natural circulation

Question Source:

Bank - INPO modified

Question History: Last NRC Exam: Beaver Valley 5/2005

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.14 - Principals of heat transfer, thermodynamics and fluid mechanics.

Comments: no change from initial submittal

K/A: APE 015/017 AK1.01 - Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Natural circulation in a nuclear reactor power plant (4.4/4.6)

Question 43.

RO Question 5 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	_____
	Group:	<u>1</u>	_____
	K/A:	APE 022 AK1.02	
	Importance Rating:	2.7	3.1

Proposed Question:

Which of the following describes the basis for opening a PORV in FR-S.1, Response to Nuclear Power Generation, if pressure is greater than 2335 psig?

- A. To prevent passing two phase flow through the safety valves.
- B. To ensure PTS limits will not be exceeded when the reactor is tripped and cools down.
- C. To minimize primary-to-secondary leakage in case of a SGTR, until other recovery actions can be taken.
- D. To allow sufficient borated injection flow into the RCS to ensure the addition of negative reactivity to the core.

Proposed Answer:

D. To allow sufficient borated injection flow into the RCS to ensure the addition of negative reactivity to the core.

Explanation:

A incorrect. 2 phase flow through safeties is a concern for accidents such as steam generator safeties and overfill, but not the bases for this check of pressure in FR-S.1

B incorrect. PTS is a concern for overcooling events, such as a steam break.

C incorrect. SGTR is not a concern in FR-S.1 at this time. The concern is inserting negative reactivity to shutdown power generation.

D correct. From FR-S.1 Background: *The check on RCS pressure is intended to alert the operator to a condition which would reduce charging or SI pump injection into the RCS and, therefore, boration.* The PRZR PORV pressure setpoint is chosen as that pressure at which flow into the RCS is insufficient. The contingent action is a rapid depressurization to a pressure which would allow increased injection flow. When primary pressure drops 200 psi below the PORV pressure setpoint, the PORVs should be closed. The operator must verify successful closure of the PORVs, closing the isolation valves, if necessary.

Technical Reference(s): FR-S.1 Background

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:

Bank – DCP P-0619

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No change to question. Expanded justification to address why A, B and C are incorrect.

K/A: APE 022 AK1.02 - Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Relationship of charging flow to pressure differential between charging and RCS (2.7/3.1)

Question 44.

RO Question 6

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	APE 025 AK2.03	
	Importance Rating:	2.7	2.7

Proposed Question:

GIVEN:

- Unit 1 has just entered MODE 5, cooling down for refueling.
- Both trains of RHR are in service.
- 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.

Proposed Answer:

- C. Unchanged to both heat exchangers.

Explanation:

A incorrect. Flow is unchanged to both heat exchangers..

B incorrect. Flow is unchanged to both heat exchangers.

C correct. Loads affected are:

- Boric Acid evaporator
- RCPs
- Letdown
- ASDR.
- Flow is not changed to the RHR heat exchangers

D incorrect. Flow to both heat exchangers is unchanged.

Technical Reference(s): OP SD-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 35316 - Identify the basic interrelationships between the RHR system and other systems.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No change from initial submittal

K/A: APE 025 AK2.03 - Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps (2.7/2.7)

Question 45.

RO Question 7 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	APE 027 AK3.01	
	Importance Rating:	3.5	3.8

Proposed Question:

GIVEN:

- The plant is at 25% increasing to full power.
- Rods are in MANUAL.
- PDP is in service.
- RCS pressure is 2265 psig.
- Pressurizer heaters on to equalize RCS and Pressurizer boron.

A loss of PY-11 occurs.

Which of the following actions would be appropriate for the current plant conditions?

- A. Take manual control of charging due to loss of PDP speed control.
- B. Verify the Pressurizer spray valves are closed due to the trip of the Pressurizer heaters.
- C. Perform the immediate actions of E-0, Reactor Trip or Safety Injection due to the energizing of a Source Range.
- D. Perform the actions of AP-29, Main Turbine Malfunction, due to the failure of the PT-505 and PT-506.

Proposed Answer:

- B. Verify the Pressurizer spray valves are closed due to the trip of the Pressurizer heaters.

Explanation:

A incorrect, PDP speed control is affected by a loss of air.

B correct. Loss of PY-11 causes letdown to isolate and the heaters to turn off, (Pressurizer level channel fails low). Additionally, sprays fail to MANUAL. If they are open, they will remain open and must be closed by the operator to prevent depressurizing. (B correct).

C incorrect, not at a power level which would result in a reactor trip.

D incorrect, only PT-505 fails on a loss of PY-11. Note: a loss of PY-12 causes both to fail low but the turbine does not trip.

Technical Reference(s): OIM A-4-2b

Proposed references to be provided to applicants during examination: none

Learning Objective: 4303 - Explain the consequences of de-energizing 120 VAC power supplies.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: Changed charging pump in service (PDP) to make distractor A (also changed) a credible distractor. Added "pressurizer" to final bullet to delineate which heaters are on.

K/A: APE 027 AK3.01 - Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Isolation of PZR spray following loss of PZR heaters (3.5/3.8)

Question 46.

RO Question 8

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	EPE 029 EK2.06	
	Importance Rating:	2.9	3.1

Proposed Question:

GIVEN:

- Reactor Trip testing is in progress on Train "A"
- Reactor Trip Breaker "A" is open, the bypass breaker is closed.

A transient occurs requiring a reactor trip. The RO attempts to manually trip the reactor but the reactor does NOT trip.

Which of the following describes a malfunction that could have contributed to the failure of the reactor to trip?

- A. Reactor Trip Breaker "B" Shunt Trip coil failed to deenergize.
- B. Reactor Trip Breaker "B" Undervoltage Trip coil failed to energize.
- C. Reactor Trip Bypass Breaker "A" Shunt Trip coil failed to energize.
- D. Reactor Trip Bypass Breaker "B" Undervoltage Trip coil failed to deenergize.

Proposed Answer:

C. Reactor Trip Bypass Breaker "A" Shunt Trip coil failed to energize.

Explanation:

A incorrect, shunt trip coils must energize to trip.

B incorrect, RTB "B" UV trip coils are normally energized. Deenergizes on trip signal.

C correct, Bypass breakers have shunt trip coils which are energize to actuate.

D incorrect, bypass breaker A is closed.

Technical Reference(s): OIM A-3-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 37051 - Explain how to diagnose and respond to Reactor Protection System problems, including:
Failure of P6 to automatically reset
SSPS Failures

Reactor Trip Initiates, but fails to Actuate

Question Source:
Bank - INPO

Question History: Last NRC Exam: Beaver Valley 12/02

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.6 - Design, components, and function of reactivity control mechanisms and instrumentation.

Comments: No change from initial submittal.

K/A: EPE 029 EK2.06 - Knowledge of the interrelations between the following and an ATWS: Breakers, relays, and disconnects (2.9/3.1)

Question 47.

RO Question 9

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	EPE 038 G2.4.31	
	Importance Rating:	3.3	3.4

Proposed Question:

GIVEN:

- The crew is performing the actions of E-3, Steam Generator Tube Rupture
- SI has been terminated
- Charging and letdown are in service

While isolating Accumulators at step 34, PK05-07, Subcooling Margin Lo/Lo-Lo alarms. The CO reports the valid alarm inputs are:

- 1151 - Subcooled Mon Lo-Lo Margin From RVLIS Train A (setpoint <20°F)
- 1160 - Subcooled Mon Lo-Lo Margin From RVLIS Train B (setpoint <20°F)

What action should be taken by the crew?

- A. Turn on Pressurizer heaters.
- B. Continue with the isolation of the Accumulators.
- C. Restart ECCS pumps and return to step 9 to re-perform the RCS cooldown and restore subcooling.
- D. Restart ECCS pumps and go to ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired.

Proposed Answer:

D. Restart ECCS pumps and go to ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired.

Explanation:

A incorrect, this is an action from the table at step 35 to control RCS pressure.

B incorrect, action to address the low subcooling is necessary.

C incorrect, alarm is an indication of LOCA, transition is required by foldout page.

D correct, per foldout page, once SI has been terminated, (step 25), subcooling below 20F requires ECCS pumps to be restarted and a transition to ECA-3.1.

Technical Reference(s): AR PK05-07 and E-3, foldout page.

Proposed references to be provided to applicants during examination: E-3 foldout page.

Learning Objective: 6776 - State the purpose of subcooling and pressurizer level check after terminating ECCS flow in E-3

7336 - State contents of foldout page

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No change from initial submittal.

K/A: EPE 038 G2.4.31 – SGTR - Knowledge of annunciators alarms and indications, and use of the response instructions. (3.3/3.4)

Question 48.

RO Question 10

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	_____
	Group:	<u>1</u>	_____
	K/A:	APE 040	AA2.01
	Importance Rating:	4.2	4.7

Proposed Question:

The plant trips from full power due a steam break.

Plant conditions 5 minutes later:

- SI actuated
- MSI actuated
- FWI actuated
- All equipment operated as designed
- Steam Generator Pressures:
 - 11 – 900 psig, stable
 - 12 – 300 psig, decreasing
 - 13 – 300 psig, decreasing
 - 14 – 870 psig, stable.

Which of the following is a possible location for the steam break?

- A. Upstream of Steam Generator 12 MSIV FCV-42.
- B. Downstream of Steam Generator 13 MSIV FCV-43.
- C. On the line to the TDAFW pump, upstream of FCV-95.
- D. On the line to the TDAFW pump, downstream of FCV-95.

Proposed Answer:

- C. On the line to the TDAFW pump, upstream of FCV-95.

Explanation:

A incorrect, both 12 and 13 Steam Generator pressures are low. Check valves between the generators (off the AFW pump line) prevent both from depressurizing.

B incorrect, MSIVs are shut which would have isolated the break if this was the location.

C correct. Both feed the AFW pump. A break on the line upstream of the FCV would cause both to depressurize.

D incorrect. The plant tripped due to the steam break. FCV-95 is normally closed. A break here would not have affected plant operation at power.

Technical Reference(s): OVID 106704 sheet 4

Proposed references to be provided to applicants during examination: None

Learning Objective: 7240 - Identify the location of Main Steam system valves and piping.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: No change from initial submittal.

K/A: APE 040 AA2.01 - Ability to determine and interpret the following as they apply to the Steam Line Rupture: Occurrence and location of a steam line rupture from pressure and flow indications (4.2/4.7)

Question 49.

RO Question 11 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	APE 054 G2.2.22	
	Importance Rating:	3.4	4.1

Proposed Question:

The purpose of which of the following components in the Main Feedwater system is to isolate the safety portion of the system from the non-safety portion of the system in the event of a feedline break?

- A. Feedwater Isolation valves only.
- B. Feedwater Main Feed Reg valves only.
- C. Feedwater Main Feed Reg and Bypass valves only.
- D. Feedwater Isolation valves, Main Feed Reg and Bypass valves.

Proposed Answer:

- A. Feedwater Isolation valves only.

Explanation:

A correct. –The MFIVs isolate the non-safety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops....

B incorrect. The MRV valves are in the non-safety portion of the system.

C incorrect. The bypass valves are in the non-safety portion of the system.

D incorrect. The MRV and bypass valves are in the non-safety portion of the system.

Technical Reference(s): C8A, Main Feedwater

Proposed references to be provided to applicants during examination: None

Learning Objective: 69103 - Explain the importance of the Main Feedwater system to plant safety.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41.4 - Secondary coolant and auxiliary systems that affect the facility.

Comments: expanded question to state the purpose is for a feedline break. This taken from the TS bases but is required RO knowledge and is covered in the System Training Guide under the importance to safety section.

K/A: APE 054 G2.2.22 – Loss Of Main Feedwater - Knowledge of limiting conditions for operations and safety limits. (3.4/4.1)

NOTE: information in the System Training Guide is taken from the basis for TS 3.7.3. The objective given is required for Reactor Operators. Removed reference to TS 3.7.3.

Question 50.

RO Question 12 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	EPE 055 EK1.02	
	Importance Rating:	4.1	4.4

Proposed Question:

The crew is performing a secondary depressurization to reduce RCS pressure and inject Accumulators in accordance with ECA-0.0, Loss of All AC Power.

Which of the following actions or cautions in the step is designed to prevent an interruption of natural circulation?

- A. Stopping when RCS temperature is 270°F
- B. Stopping when secondary pressure is 240 psig.
- C. Depressurizing the secondary at the maximum rate.
- D. Maintaining intact Steam Generator levels between 20 and 50%.

Proposed Answer:

- B. Stopping secondary depressurization at 240 psig.

Explanation:

A incorrect, limit ensures a challenge to RCS integrity does not occur.

B correct, ensures that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS.

C incorrect, The S/Gs are being depressurized at maximum rate to minimize RCS inventory loss due to anticipated RCP Seal failures.

D incorrect, normal band is not required. Limit is 6% (U-tubes covered)

Technical Reference(s): EOP ECA-0.0 and background step 17

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.8 - Components, capacity, and functions of emergency systems.

Comments: Modified question - changed “steps or cautions in this step...” to “actions or cautions in this step...”

K/A: EPE 055 EK1.02 – Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling (4.1/4.4)

Question 51.

RO Question 13

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	APE 057 AA2.19	
	Importance Rating:	4.0	4.3

Proposed Question:

A loss of a 120 VAC vital instrument bus causes the loss of MANUAL power to a controller which is in AUTOMATIC.

How are the indications and operation of the controller affected?

- A. Lights on the controller go out; the controller fails to Manual.
- B. Lights on the controller go out; the controller fails to Auto-Hold.
- C. Lights on the controller remain lit; the controller fails to Manual.
- D. Lights on the controller remain lit; the controller fails to Auto-Hold.

Proposed Answer:

- B. Lights on the controller go out; the controller fails to Auto-Hold.

Explanation:

A incorrect. Controller fails to Auto-Hold.

B correct. From O-2: NOTE 1: Manual power supplies power to the Hagan controller lights. Therefore, if manual power is lost, the lights on the controller will go out and the controller will go to the AUTO-HOLD setpoint, indicated by the controller demand meter indicating a value above zero. The controller will remain in AUTO-HOLD until manual power is restored. Once manual power is restored, the Hagan controller will transfer to Manual in 5-30 seconds.

C incorrect. Lights go out and controller fails to Auto-Hold.

D incorrect. Lights go out.

Technical Reference(s): OP O-2, attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: 7970 - Identify failure modes associated with Hagan controllers

Question Source:
DCPP Bank P-73313

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No change from initial submittal.

K/A: APE 057 AA2.19 – Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus. (4.0/4.3)

Question 52.

RO Question 14 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	APE 062	AA2.02
	Importance Rating:	2.9	3.6

Proposed Question:

GIVEN:

- Unit 1 is in a normal full power lineup.
- STP M-9A, Diesel Engine Generator Routine Surveillance Test is being performed on DEG 1-2.
- The diesel has been paralleled with offsite and loaded to 2.5 MW.

The running ASW pump, 1-1 trips.

Which of the following describes what could have tripped ASW pump 1-1 and response of ASW pump 1-2?

- A. Pump 1-1 tripped on overcurrent. Pump 1-2 will not start.
- B. Pump 1-1 tripped on overcurrent. Pump 1-2 will start on low system pressure.
- C. Pump 1-1 tripped on low system pressure. Pump 1-2 will not start.
- D. Pump 1-1 tripped on low system pressure. Pump 1-2 will start when the breaker for Pump 1-1 opens.

Proposed Answer:

B. Pump 1-1 tripped on overcurrent. Pump 1-2 will start on low system pressure.

Explanation:

A incorrect. Power supplies for ASW pumps are 11 – bus F, 12 – bus G. When the diesel auto transfers to the 4 kv bus (DEG 12 supplies bus G), the auto start on low pressure is blocked. However, since the diesel is currently powering the bus (not due to an auto transfer), the standby pump will start.

B correct. The standby pump will start on low system pressure.

C and D incorrect. Low system pressure is an auto start, but not a trip for the ASW pump.

Technical Reference(s): STG E5, ASW. STG J6A, 4kv.

Proposed references to be provided to applicants during examination: None

Learning Objective: 5365 - Analyze ASW pump control logic.

Question Source:
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Minor answer and distractor wording change (trips to tripped, 11 to 1-1), no other changes from initial submittal.

K/A: APE 062 AA2.02 - Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss

Question 53.

RO Question 15

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	_____
	Group:	<u>1</u>	_____
	K/A:	APE 065	AA1.02
	Importance Rating:	2.6	2.8

Proposed Question:

Unit 1 is at full power.

An air leak develops upstream of instrument air valve 221.

Which of the following is the effect on the instrument air headers to the penetration area and containment?

- A. Both headers will rapidly depressurize until FCV-584 closes to isolate both headers.
- B. Both headers will rapidly depressurize until FCV-584 closes to isolate containment. The penetration area will continue to depressurize.
- C. A check valve between the break and the Backup Air Receiver will minimize the effect on both headers.
- D. A check valve between the break and the Backup Air Receiver will minimize the effect on the containment header. The penetration area header will depressurize.

Proposed Answer:

C. A check valve between the break and the Backup Air Receiver will isolate the Backup Air Receiver and air to Containment from the break.

Explanation:

A incorrect. Check valve will isolate the break.

B incorrect. FCV-584 does not close on low pressure.

C correct. A check valve just before the Air Receiver will act to isolate the break from the Receiver and Containment air.

D incorrect. The check valve is before the Receiver.

Technical Reference(s): STG K1, Compressed Air, OVID 106725 sheet 25

Proposed references to be provided to applicants during examination: OVID 106725 sheet 25

Learning Objective: 7209 - Explain the effect of malfunctions in the station air system:
Containment
Effect on the plant
Effect on air operated valves

Question Source:
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No change from initial submittal.

K/A: APE 065 AA1.02 – Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Components served by instrument air to minimize drain on system (2.6/2.8)

Question 54.

RO Question 16 rev1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	_____
	Group:	<u>1</u>	_____
	K/A:	EPE E04 EK3.2	_____
	Importance Rating:	3.4	4.0

Proposed Question:

GIVEN:

- The crew is performing ECA-1.2, LOCA Outside Containment
- RWST level is 30%.
- 8809A (RHR to cold legs 1 and 2) and 8809B (RHR to cold legs 3 and 4) have been closed and reopened when RCS pressure continued to decrease

Which procedure should the crew enter from ECA-1.2 and why?

- A. E-1.3, Transfer to Cold Leg Recirculation, because swap over to the Cold Leg Recirculation is required.
- B. E-1, Loss of Primary or Secondary Coolant, because the leak must be in another location.
- C. ECA-1.1, Loss of Emergency Coolant Recirculation, because there is no inventory in the Containment sump.
- D. ECA-1.1, Loss of Emergency Coolant Recirculation, because the leak must be in another location

Proposed Answer:

C. ECA-1.1, Loss of Emergency Coolant Recirculation, because there is no inventory in the Containment sump.

Explanation:

A incorrect. Although below the setpoint for transfer to cold leg recirc, there is no sump inventory to align to. Additionally, unlike other EOPs, there is no fold out page transition criteria. This is one of the few procedures a transition to E-1.3 is not appropriate.

B incorrect. Although the leak is not isolated, E-1 presumes a leak inside containment. Plant indications to drive the crew to ECA-1.2 indicate this is not the case. Additionally, the procedure entered due to decreasing pressure is ECA-1.1. E-1 is entered if pressure is increasing.

C correct. The leak is not isolated and a transition to ECA-1.1 is appropriate due to the lack of inventory in the sump.

D incorrect. This is not the reason to go to ECA-1.1. ECA-1.1 will deal with minimizing ECCS flow and make up to the RWST.

Technical Reference(s): ECA-1.2 background

Proposed references to be provided to applicants during examination: none

Learning Objective: 5433 - Identify exit conditions for the EOPs

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: changed focus of original question to have candidate determine which procedure is appropriate and the reason for the procedure selection. No longer an open reference question.

K/A: E04 EK3.2 - Knowledge of the reasons for the following responses as they apply to the (LOCA Outside Containment): Normal, abnormal and emergency operating procedures associated with (LOCA Outside Containment). (3.4/4.0)

Question 55.

RO Question 17

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	_____
	Group:	<u>1</u>	_____
	K/A:	E05 G2.1.27	_____
	Importance Rating:	2.8	2.9

Proposed Question:

What is the purpose of opening all the Pressurizer PORVs when addressing a loss of secondary heat sink?

- A. Prevent lifting the Pressurizer safeties.
- B. Delay the time to steam generator dryout.
- C. Depressurize the RCS to allow automatic SI actuation.
- D. Provide adequate RCS heat removal until restoration of a heat sink.

Proposed Answer:

D. Provide adequate RCS heat removal until restoration of a heat sink.

Explanation:

A incorrect. For this accident, the PORVs are not operated to prevent safeties lifting but to provide adequate heat sink.

B incorrect. Steam generators are not a heat sink when bleed and feed initiated.

C incorrect. This is Feed and Bleed which is not performed.

D correct. If operated in accordance with FR-H.1, the PORVs will allow SI injection of subcooled water which will remove RCS decay heat removal until a secondary heat sink can be restored.

Technical Reference(s): FR-H.1 Background

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No change from initial submittal.

K/A: E05 G2.1.27 - Loss of Secondary Heat Sink – Knowledge of system purpose and or function. (2.8/2.9)

Question 56.

RO Question 18

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>1</u>	<u> </u>
	K/A:	EPE E11 EA1.1	
	Importance Rating:	3.9	4.0

Proposed Question:

GIVEN:

- The crew is performing the actions of ECA-1.1, Loss of Emergency Coolant Recirculation.
- All ECCS pumps are running.
- All vital buses are energized from their Emergency Diesel Generators.
- SI has been reset.

One train of ECCS is to be stopped.

What action must be taken prior to the operator stopping the first charging pump?

- A. Locally reset each 4 kV Vital Bus Auto Transfer Trip Relay, only.
- B. Depress Vital 4KV Auto Transfer pushbutton for each bus on VB4, only.
- C. Depress Vital 4KV Auto Transfer pushbutton for each bus on VB4 and locally reset each 4 kV Vital Bus Auto Transfer Trip Relay.
- D. Depress Vital 4KV Auto Transfer pushbutton for each bus on VB4 and place each Bus Auto Xfer to S/U Cutout Switch to CUTOOUT.

Proposed Answer:

- B. Depress Vital 4KV Auto Transfer pushbutton for each bus on VB4, only.

Explanation:

A incorrect. No local action is required. This relay is for resetting the auto transfer not operating ECCS equipment.

B correct. This is the only action required.

C incorrect. No local action required (relay is for auto transfer)

D incorrect. The CUTOOUT switches are used for transferring buses.

Technical Reference(s): ECA-1.1 and STG J15.

Proposed references to be provided to applicants during examination: None

Learning Objective: 37857, Explain the operation of auto transfer components.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.8 - Components, capacity, and functions of emergency systems.

Comments: No change from initial submittal.

K/A: E11 EA1.1 - Ability to operate and / or monitor the following as they apply to the (Loss of Emergency Coolant Recirculation): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (3.9/4.0)

Question 57.

RO Question 19

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	APE 001 G2.4.31	
	Importance Rating:	3.3	3.4

Proposed Question:

Unit 1 is at 65% power. Control systems are in AUTO.

Rods begin to step out at 8 steps per minute.

The following alarms are received:

- PK04-03, Tavg Deviation From Tref
- PK05-16, Pzr Pressure High
- PK05-22, Pzr Level Hi/Lo

The operator reports Tref is unchanged.

Which of the following actions should be taken by the operator?

- A. Trip the reactor due to a vapor space leak.
- B. Place rods in Manual due to a dropped rod.
- C. Place rods in Manual due to a malfunction of rod control.
- D. Place rods in Manual due to a power range instrument failure.

Proposed Answer:

C. Place rods in Manual due to a malfunction of rod control.

Explanation:

A incorrect, no indication of trip required, for a vapor space break, Tave would not change.

B incorrect, a dropped rod would not cause high pressure.

C correct, indication of outward rod motion. Action is to place rods in Manual.

D incorrect, no power range instrument failure would cause high pressure, high level and increasing temperature and outward rod motion.

Technical Reference(s): OP AP-12A, Continuous Withdrawal or Insertion of a Control Rod Bank.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3476 - Explain the general purpose/function of abnormal operating procedures

Question Source:
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No change from initial submittal.

K/A: APE 001 G2.4.31 – Continuous Rod Withdrawal -Knowledge of annunciators alarms and indications, and use of the response instructions. (3.3/3.4)

Question 58.

RO Question 20

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	APE 028 AK2.02	
	Importance Rating:	2.6	2.7

Proposed Question:

GIVEN:

- Unit 1 is at 50% power
- All control systems are in their normal alignment
- Pressurizer Level Control is selected to 459/461

Which of the following Pressurizer level failures would initially cause a large increase in charging flow?

- A. LT-459 fails low.
- B. LT-461 fails low.
- C. LT-459 fails high.
- D. LT-461 fails high.

Proposed Answer:

- A. LT-459 fails low.

Explanation:

A correct. Charging flow is driven by the controlling channel. If the controlling channel fails low, charging will increase to raise level

B incorrect, backup channel does not affect charging flow.

C incorrect, failing high causes charging to decrease.

D incorrect, charging not affected by backup channel.

Technical Reference(s): OIM A-4-2b

Proposed references to be provided to applicants during examination: None

Learning Objective: 36926 - Explain the effects of Pressurizer system failures, including: Transmitters

Question Source:

DCPP P-1509

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: No change from initial submittal.

K/A: APE 028 AK2.02 – Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Sensors and detectors (2.6/2.7)

Question 59.

RO Question 21 rev 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	APE 033 AK3.01	
	Importance Rating:	3.2	3.6

Proposed Question:

GIVEN:

- Plant startup in progress
- Reactor is critical at 1500 cps.

N36 loses control power and fails low.

Which of the following describes why the startup will be terminated?

- A. The reactor has tripped.
- B. P-6 actuation has de-energized both Source Ranges.
- C. Two Intermediate Range channels are required to be OPERABLE below P-6.
- D. Two Intermediate Range channels are required to be OPERABLE below P-10.

Proposed Answer:

- A. The reactor has tripped.

Explanation:

A correct, when the IR loses control power, the High Flux trip (1 of 2) will cause the reactor to trip.

B incorrect, P-6 automatically energizes the Source Ranges, it has not turned off the Source Ranges.

C incorrect, Not required below P-6.

D incorrect, IR operability applies above P-6 (action is to reduce power to less than P-6).

Technical Reference(s): OIM B-4-2, TS 3.3.1

Proposed references to be provided to applicants during examination: Tech Spec 3.3.1

Learning Objective: 36973 - Explain the effect of Excore NIS channel failures, including:

- Individual Channel Failures
- Loss of power supplies/fuses

Question Source:
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: Question rewritten to focus on why startup would be terminated to align it with the KA.

K/A: APE 033 AK3.01 – Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Termination of startup following loss of intermediate range instrumentation (3.2/3.6)

Question 60.

RO Question 22 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	APE 036	AA2.03
	Importance Rating:	3.1	4.2

Proposed Question:

In order to limit the release due to a dropped fuel assembly, ECG 42.4, Refueling Operations - Crane Travel - Fuel Handling Building, requires immediate operator action if load over the Spent Fuel Pool exceeds _____ pounds.

- A. 1750
- B. 2000
- C. 2250
- D. 2500

Proposed Answer:

- D. 2500

Explanation:

A incorrect. This is close to the weight of hoist with an assembly (1790 psig).
B incorrect. Hoist and assembly with approximately a 10% allowance.
C incorrect. Overload is 2180 psig. This would be close to that limit.
D correct. ECG 42.4 requires the operator to immediately place the crane in a safe location if load over the Spent Fuel Pool exceeds 2500 pounds. In the basis, the reason is to limit a potential release to a single fuel assembly and prevent possibly causing a critical array.

Technical Reference(s): ECG 42.4.

Proposed references to be provided to applicants during examination: None

Learning Objective: 66070 - Discuss the requirements of System 42 ECGs.

Question Source:
INPO

Question History: Last NRC Exam: Braidwood, 2001

Question Cognitive Level:
Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.8 - Components, capacity, and functions of emergency systems.

Comments: Modified justification. No change to question from initial submittal.

K/A: APE 036 AA2.03 - Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release (3.1/4.2)

Question 61.

RO Question 23

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	APE 059	AA2.05
	Importance Rating:	3.6	3.9

Proposed Question:

A HIGH RADIATION alarm is received on RE-18, the liquid radwaste rad monitor.

Which of the following automatic action(s) occur?

- A. RCV-18 (liquid radwaste overboard) closes and FCV-477 (recirc to the EDR) opens.
- B. RCV-18 (liquid radwaste overboard) closes and FCV-477 (recirc to the EDR) closes.
- C. RCV-18 closes and all liquid radwaste pumps that are running receive a trip signal.
- D. RCV-18 (liquid radwaste overboard) remains open and FCV-477 (recirc to the EDR) opens.

Proposed Answer:

- A. RCV-18 (liquid radwaste overboard) closes and FCV-477 (recirc to the EDR) opens.

Explanation:

A correct. The discharge valve closes and recirc to the EDR opens.

B incorrect, the EDR recirc opens.

C incorrect, pumps do not trip.

D incorrect, RCV-18 closes.

Technical Reference(s): OIM G-3-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 69251 - Explain the automatic actions associated with the Liquid Radwaste system.

Question Source:

Bank - DCP P-0179

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.11 - Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments: No change from initial submittal.

K/A: APE 059 AA2.05 – Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: The occurrence of automatic safety actions as a result of a high PRM system signal (3.6/3.9)

Question 62.

RO Question 24

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	_____
	Group:	<u>2</u>	_____
	K/A:	APE 069 AK3.01	
	Importance Rating:	3.8	4.2

Proposed Question:

GIVEN:

- Containment pressure is 24 psig.
- The crew is performing the actions of FR-Z.1, Response to High Containment Pressure.
- The operator verifies the Containment Fan Cooler Units (CFCUs) are running in LOW.

Why are the CFCUs operated in LOW instead of HIGH?

- A. CCW cooling is isolated to the units.
- B. To prevent overwhelming the drain system.
- C. To prevent the units from tripping on overcurrent.
- D. Design flow through the annular ring would be exceeded.

Proposed Answer:

C. To prevent the units from tripping on overcurrent.

Explanation:

A incorrect, cooling is not isolated to the units.

B incorrect, the drain system can handle the units running in HIGH.

C correct, Emergency operations call for CFCU fans to be in LOW speed to prevent the fans tripping on overcurrent due to the extremely dense atmosphere after a LOCA.

D incorrect, not a design concern.

Technical Reference(s): Westinghouse Background FR-Z.1, step 4, STG H2

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Emergency operations call for CFCU fans to be in LOW speed to prevent the fans tripping on overcurrent due to the extremely dense atmosphere after a LOCA.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No change from initial submittal.

K/A: APE 069 AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Containment Integrity: Guidance contained in EOP for loss of containment integrity (3.8/4.2)

Question 63.

RO Question 25 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	APE 076 AK2.01	
	Importance Rating:	2.6	3.0

Proposed Question:

Unit 1 has been operating at 100% power for three weeks with a 5 gpd tube leak in Steam Generator 1-2.

Which of the following will increase if RCS activity increases due to a fuel defect?

- A. Steam Jet Air Ejector Monitor, RM-15, counts.
- B. RCS Argon-40.
- C. Steam Line Radiation Monitor, RM-73, counts.
- D. Steam Generator 1-2 tube leak flow.

Proposed Answer:

- A. Steam Jet Air Ejector Monitor, RM-15 counts.

Explanation:

A correct, the purpose of the Steam Jet Air Ejector (SJAE) Monitor is to monitor noble gas flow and its release concentration prior to its exhaust to the plant vent.

B incorrect, Argon is not an indication of failed fuel.

C incorrect, the purpose of Monitors RM-71, 72, 73, and 74 is to continuously monitor the main steam lines for gamma radiation that could indicate a steam generator tube leak. RM-73 monitors Steam Generator 13

D incorrect, only a further failure of the tube will increase the leak flow.

Technical Reference(s): O-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 33808 - Interpret indications of core damage

Question Source:

Bank - INPO

Question History: Last NRC Exam: DCPD 10/02

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.11 - Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments: added names to Radiation Monitors.

K/A: APE 076 AK2.01 – Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors (2.6/3.0)

Question 64.

RO Question 26 R1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	EPE E02 EK1.1	
	Importance Rating:	3.2	3.8

Proposed Question:

The following plant conditions exist after stabilizing from a small break LOCA:

- RCS Pressure = 1750 psig
- Pressurizer Level = 20% stable
- Subcooling = 45°F
- AFW flow = 500 gpm
- 1 CCP running
- 2 SI pumps running

How will Pressurizer level be affected if both SI pumps are shut down?

- A. Increase
- B. Decrease
- C. No change
- D. Initial sharp decrease, then slowly stabilize

Proposed Answer:

C. No change

Explanation:

A incorrect. Turning off the SI pumps will not result in more flow and subsequent repressurization.

B incorrect. RCS pressure is greater than shutoff head. Pressure should not change.

C correct. RCS pressure is above SI pump shutoff head. When the SI pumps are secured, there will not be an RCS pressure response or temperature response.

D incorrect. This is a typical response when pumps are secured during ECCS reduction sequence.

Technical Reference(s): STG B3

Proposed references to be provided to applicants during examination: None

Learning Objective: 6743 Explain PZR response during ECCS reduction sequence

Question Source:

Bank P-6157

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.8 - Components, capacity, and functions of emergency systems.

Comments: Expanded justification. No change to question from initial submittal.

K/A: EPE E02 EK1.1 – Knowledge of the operational implications of the following concepts as they apply to the (SI Termination) Components, capacity, and function of emergency systems. (3.2/3.8)

Question 65.

RO Question 27

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>1</u>	<u> </u>
	Group:	<u>2</u>	<u> </u>
	K/A:	EPE E07 EA1.3	
	Importance Rating:	3.5	3.9

Proposed Question:

The crew has entered and is performing the actions of FR-C.3, Response to Saturated Core Conditions.

The desired outcome of this procedure is to isolate RCS vent paths and which of the following conditions?

- A. Subcooling restored.
- B. ECCS pumps secured.
- C. RCS temperature less than 200°F.
- D. Core exit thermocouples stable or decreasing.

Proposed Answer:

- A. Subcooling restored.

Explanation:

A correct. Procedure verifies ECCS pumps running (based on RCS pressure) and vents, such as PORVs closed to restore subcooling .

B incorrect. Procedure verifies pump injecting not secured.

C incorrect. No cooldown in the procedure.

D incorrect. No guidance regarding core exit thermocouples.

Technical Reference(s): FR-C.3, Response to Saturated Core Conditions

Proposed references to be provided to applicants during examination: None

Learning Objective: 3551 - State the major action categories for EOPs

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No change from initial submittal.

K/A: EPE E07 EA1.3 - Ability to operate and / or monitor the following as they apply to the (Saturated Core Cooling): Desired operating results during abnormal and emergency situations. (3.5/3.9)

Question 66.

RO Question 66

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u> 3 </u>	<u> </u>
	Group:	<u> 1 </u>	<u> </u>
	K/A:	G2.1.9	
	Importance Rating:	2.5	4.0

Proposed Question:

GIVEN:

- Unit 1 is at full power.
- Unit 2 is in MODE 6.
- The Unit 2 Shift Foreman is making a plant tour.
- The Unit 2 CO is the only operator “at the controls”
- The Shift Manager is in the Shift Manager’s office.

A currently Licensed SRO procedure writer wishes to enter the Unit 2 area to obtain valve and meter numbers on VB2.

What action should the procedure writer take to gain access to VB2?

- A. Ask the Unit 2 CO.
- B. Locate and ask the Unit 2 Shift Foreman.
- C. Go to the Shift Manager for permission.
- D. Enter the area; an operator with a current license may enter the area at any time.

Proposed Answer:

- A. Ask the Unit 2 CO.

Explanation:

A correct. if the CO is the only one in the Control Room he is asked.

B incorrect, SFM permission not required.

C incorrect, the CO may grant permission.

D incorrect, only operators on watch may enter the area without permission.

Technical Reference(s): Operations Policy A-21, Control Room Protocol

Proposed references to be provided to applicants during examination: None

Learning Objective: 3076 - Explain the requirements for control of access to the Control Room

Question Source:
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.45.12 - Demonstrate the knowledge and ability as appropriate to the assigned position to assume the responsibilities associated with the safe operation of the facility.

Comments: No changes to initial submittal.

K/A: G2.1.9 – Ability to direct personnel activities inside the control room. (2.5/4.0)

Question 67.

RO Question 67

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>1</u>	_____
	K/A:	G2.1.20	_____
	Importance Rating:	4.3	4.2

Proposed Question:

While operating at 100% power, a valid reactor trip signal is received but the reactor did not automatically trip.

The BOPCO successfully initiates a reactor trip by de-energizing 480v buses 13D and 13E when the crew enters E-0, "Reactor Trip or Safety Injection," and the reactor indicates TRIPPED.

The SFM immediately transitions to FR-S-1, "Response to Nuclear Power Generation/ ATWS," and begins performing step 1.

The procedural response to this situation was:

- A. Correct, FR-S.1, Step 1, RNO instructs the operator to de-energize 480v buses 13D and 13E.
- B. Correct, direct entry in FR-S.1 is allowed because the reactor failed to automatically trip.
- C. Incorrect, E-0, Step 1, should have been performed prior to going to FR-S.1.
- D. Incorrect, E-0 should have been continued once the reactor was tripped at Step 1, RNO.

Proposed Answer:

D. Incorrect, E-0 should have been continued once the reactor was tripped at Step 1, RNO.

Explanation:

A incorrect. This is part of the immediate action of E-0 and not FR-S.1

B incorrect. FR-S.1 entered thru E-0 when auto and manual trip fails.

C incorrect. Step 1 of E-0 was performed and entry conditions do not warrant entry into FR-S.1

D correct. Step 1 of E-0 was performed successfully. The proper action is to continue with E-0. Transitioning to FR-S.1 is not appropriate.

Technical Reference(s): E-0, Reactor Trip or Safety Injection.

Proposed references to be provided to applicants during examination: None

Learning Objective: 9693 - State the steps and transitions in procedures that are considered immediate actions

Question Source:

Bank P-68832

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No changes to initial submittal.

K/A: G2.1.20 – Ability to execute procedure steps. (4.3/4.2)

Question 68.

RO Question 68

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>2</u>	_____
	K/A:	G2.2.12	
	Importance Rating:	3.0	3.4

Proposed Question:

A surveillance, to verify offsite sources available, is required to be performed every 8 hours to satisfy a Technical Specification requirement while an emergency diesel is out of service.

What type of surveillance is this considered?

- A. Periodic
- B. Recurring
- C. Conditional
- D. Mode Specific

Proposed Answer:

C. Conditional

Explanation:

Only C correct. Per AD13.DC1 - Conditional Surveillance

Any surveillance that is performed on other than the normal periodic time based surveillance interval. Conditional surveillances may be required for events such as inoperable alarms and monitors or for changing plant conditions such as entering a new mode.

Note: The offsite surveillance is a 7 day surveillance normally.

Technical Reference(s): AD13.DC1 - Control of the Surveillance Testing Program

Proposed references to be provided to applicants during examination: None

Learning Objective: 3588 Identify the definitions associated with the surveillance testing and inspection program

Question Source:
Bank P-69640

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No changes to initial submittal.

K/A: G2.2.12 - Knowledge of surveillance procedures. (3.0/3.4)

Question 69.

RO Question 69 rev 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>2</u>	_____
	K/A:	G2.2.22	_____
	Importance Rating:	3.4	4.1

Proposed Question:

Unit 1 is in MODE 3. The SFM is performing STP R-19, Shutdown Margin Determination, to determine current Shutdown Margin (SDM).

If the SDM is NOT satisfactory, how quickly must a boration be initiated?

- A. Immediately.
- B. Within 15 minutes.
- C. Within 30 minutes.
- D. Within 1 hour.

Proposed Answer:

- B. Within 15 minutes.

Explanation:

A incorrect. This LCO is not immediate.

B correct. Technical Specification 3.1.1 requires initiation of boration to restore SDM within 15 minutes.

C incorrect. There are 30 minute LCOs (3.1.4 e.g.), however, this LCO is 15 minutes..

D incorrect. There are 1 hour LCOs (3.2.3 e.g.), however, this LCO is 15 minutes.

Technical Reference(s): Technical Specification 3.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

9697A - Identify 3.1 Technical Specification LCOs

Question Source:

Bank – B0125

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content:

55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments: Item resampled. Original KA was deemed to be SRO knowledge and already selected for the SRO exam.

K/A: G2.2.22 – Knowledge of limiting conditions for operations and safety limits.
(3.4/4.1)

Question 70.

RO Question 70 rev1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>2</u>	_____
	K/A:	G2.2.34	_____
	Importance Rating:	2.8	3.4

Proposed Question:

Which of the following surveillances is performed to verify the calibration of the NIS power range channels?

- A. PEP R-23, Determination and Update of Target Axial Flux Difference
- B. STP R-64, BEACON Core Power Distribution Surveillance
- C. STP R-2B1, PPC Operator Heat Balance
- D. PEP R-5, Burnup Tracking

Proposed Answer:

C. STP R-2B1, PPC Operator Heat Balance.

Explanation:

A incorrect. procedure determines the target axial flux difference (AFD) in order to maintain control rods at the desired position..

B incorrect. BEACON Power Distribution Monitoring System (PDMS) core power distribution surveillance summary report is generated. Core power peaking factors are evaluated and compared to the limits of the Technical Specifications. In addition, BEACON incore Axial Flux Difference (AFD) is compared to excore AFD to determine the need for an incore/excore NIS power range calibration.

C correct. Procedure provides a method of performing a secondary side heat balance using the PPC. The results of the heat balance (given in terms of reactor power level) are then used to verify the calibration of the NIS power range channels.

D incorrect, procedure provides a method for calculating correct and consistent burnup values as needed.

Technical Reference(s): STP R-2B1

Proposed references to be provided to applicants during examination: None

Learning Objective: 78013 - Describe the fundamentals of BEACON operation and why it's expected to approximate core behavior reliably.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.43.1 - Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

55.43.6 - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Comments: Question replaced, original question did not match KA.

K/A: G2.2.34 – Knowledge of the process for determining the internal and external effects on core reactivity (2.8/3.4)

Question 71.

RO Question 71

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>3</u>	_____
	K/A:	G2.3.1	
	Importance Rating:	2.6	3.0

Proposed Question:

10CFR20 limits the radiation exposure (dose) to a qualified radiation worker to _____ per year.

Diablo Canyon guideline limits the radiation dose to a qualified worker to _____ per year without extension.

	<u>10CFR20 Limit</u>	<u>Diablo Canyon Guideline</u>
A.	5000 mrem	2000 mrem
B.	5000 mrem	4500 mrem
C.	3000 mrem	2000 mrem
D.	3000 mrem	2700 mrem

Proposed Answer:

A. 5000 mrem 2000 mrem

Explanation:

A correct. Per RP1.ID6, admin guideline is 2 rem. Federal limit is 5 rem.

B incorrect. 4500 is 90% of the Federal limit and this is the administrative limit. Anything above the guideline requires permission.

C incorrect. 3 rem is old Federal limit.

D incorrect. 3 rem is old Federal limit. 2700 is 90% of this “old” limit.

Technical Reference(s): RP1.ID6, attachment 8.1

Proposed references to be provided to applicants during examination: None

Learning Objective: GET

Question Source:
INPO - modified

Question History: Last NRC Exam: Kewaunee 2/06

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.12 - Radiological safety principles and procedures.

Comments: No changes from initial submittal.

K/A: G2.3.1 – Knowledge of 10 CFR: 20 and related facility radiation control requirements. (2.6/3.0)

Question 72.

RO Question 72

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>3</u>	_____
	K/A:	G2.3.10	_____
	Importance Rating:	2.9	3.3

Proposed Question:

Both units are at full power.

PK15-06, Cont Room Vent alarms on Unit 1 and Unit 2.

UNIT 1 Radiation Monitors	Unit 2 Radiation Monitors
RE-25 and RE-26 - 2.5 mrem/hr.	RE-25 and RE-26 - 5.0 mrem/hr.
RE-51 and RE-52 - 2.3 mrem/hr.	RE-53 and RE-54 - 5.2 mrem/hr

What action should be taken by the operators for both units?

- A. Place Unit 1 selector switches in MODE 3. Place Unit 2 selector switches in MODE 4.
- B. Place Unit 1 selector switches in MODE 4. Place Unit 2 selector switches in MODE 3.
- C. Place the selector switches in MODE 3 on both Units 1 and 2.
- D. Place the selector switches in MODE 4 on both Units 1 and 2.

Proposed Answer:

B. Place Unit 1 selector switches in MODE 4. Place Unit 2 selector switches in MODE 3.

Explanation:

A incorrect. Unit 1 radiation level at the pressurization (RE-51 and 52) intake is lower than Unit 2's (RE-53 and 54).

B correct. The Unit with the lower reading (Unit 1) selector switches are placed in MODE 4. The other unit is placed in MODE 3.

C incorrect. Both units are placed in different modes depending on radiation levels.

D incorrect. Both units are placed in different modes depending on radiation levels.

Technical Reference(s): AR PK15-06, STG H5, CRVS

Proposed references to be provided to applicants during examination: AR PK15-06

Learning Objective: 4714 - Explain the operation of the Control Room Ventilation system.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.13 - Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments: No change from initial submittal.

K/A: G2.3.10 - Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (2.9/3.3)

Question 73.

RO Question 73

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>3</u>	_____
	K/A:	G2.3.11	
	Importance Rating:	2.7	3.2

Proposed Question:

GIVEN:

- The crew has entered E-3, Steam Generator Tube Rupture.
- The ruptured S/G has been identified.

What action will the operator perform on the ruptured steam generator 10% Steam Dump controller?

- A. The setpoint will be raised to 1040 psig to minimize radiological releases.
- B. The controller will be placed in MANUAL and closed to minimize radiological releases.
- C. The setpoint will be raised to 1040 psig to increase ruptured steam generator pressure and decrease primary to secondary leakage.
- D. The controller will be placed in MANUAL and closed to increase ruptured steam generator pressure and decrease primary to secondary leakage.

Proposed Answer:

- A. The setpoint will be raised to 1040 psig to minimize radiological releases.

Explanation:

A correct. The valve setpoint is raised but left in AUTO to actuate if necessary.

B incorrect. Controller left in AUTO.

C incorrect. Action is not performed to minimize primary to secondary leakage.

D incorrect. Controller left in AUTO, not performed to minimize primary to secondary leakage.

Technical Reference(s): E-3, Background step 3

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:

Bank - P-27379 - modified

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.13 - Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments: No change from initial submittal.

K/A: G2.3.11 - Ability to control radiation releases. (2.7/3.2)

Question 74.

RO Question 74

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u> 3 </u>	<u> </u>
	Group:	<u> 4 </u>	<u> </u>
	K/A:	G2.4.21	
	Importance Rating:	3.7	4.3

Proposed Question:

Which of the following Critical Safety Function RED paths can be directly caused by procedurally directed operator action?

- A. Subcriticality
- B. Core Cooling
- C. Heat Sink
- D. Containment

Proposed Answer:

C.Heat Sink

Explanation:

A incorrect. Caused by power generation greater than 5%.

B incorrect. Caused by overheating thermocouples.

C correct. Can be caused by the operator throttling AFW flow below 435 gpm while no steam generator is above 6%, as in the case of ECA-2.1 or FR-P.1

D incorrect. Caused by high containment pressure.

Technical Reference(s): FR-H.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 9704 - Identify entry conditions for the FRPs

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: No changes from initial submittal.

K/A: G2.4.21 - Knowledge of the parameters and logic used to assess the status of safety functions including:

1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.

(3.7/4.3)

Question 75.

RO Question 75

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	<u>3</u>	_____
	Group:	<u>4</u>	_____
	K/A:	G2.4.48	
	Importance Rating:	3.5	3.8

Proposed Question:

Unit 1 is at full power, in a normal lineup with no equipment out of service.

For which of the following set of valid conditions could the BOPCO take corrective action without obtaining prior permission, and then report their actions to the SFM "after the fact"?

- A. The operating CCP 1-2 trips on overcurrent.
- B. Containment temperature increases 5°F with 3 CFCUs running in LOW.
- C. Main Feedwater pump 1-1 trips.
- D. PK01-01, "ASW SYS HX dp/HDR PRESS" alarms with only 1 ASW heat exchanger in service.

Proposed Answer:

C. Main Feedwater pump 1-1 trips

Explanation:

A, B and D are abnormal conditions which would be promptly reported to the SFM.

C correct. An immediate action in AP-15 requires a reactor trip if power is greater than 80%.

Technical Reference(s): OP1.DC11 – Conduct of Operations – Abnormal Plant Conditions.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3705 - Explain the responsibilities of the control room staff positions during emergencies

Question Source:

Bank – Modified R-55094

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No changes from initial submittal.

K/A: G2.4.48 - Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions. (3.5/3.8)

END OF RO EXAM

Question 76.

SRO Question 76

Examination Outline Cross-Reference:	Level	RO	<u>SRO</u>
	Tier:	_____	<u>1</u>
	Group:	_____	<u>1</u>
	K/A:	_____	EPE 009 EA2.21
	Importance Rating:		3.9

Proposed Question:

Unit 1 is operating at 100% power in a normal full power, steady state lineup. Charging pump 1-1 is in service.

The following events occur:

- PK11-21, High Radiation, input 669, In Core Seal Table Area Mon Hi Rad RE 7, alarms.
- The operator reports RM-7 is reading 55 mr/hr and trending up on the PPC.
- Additionally, trends are increasing for Containment Radiation Monitors, RM-11 and RM-12.
- Containment temperature is increasing.
- Charging flow is 115 gpm.
- RCP seal injection flow is 8 gpm to each RCP.
- Pressurizer level is 60% and steady
- Letdown is not isolated

Based on the radiation monitors and system response, what action should be taken by the Shift Foreman?

- A. The leakage is small. Enter OP O-21, "Isolating Leaking Thimble Tubes" to find and isolate the leak.
- B. The leak is in excess of leakage allowed by Technical Specifications. Enter AP-25, "Rapid Load Reduction" to quickly shutdown the unit.
- C. The leak within the capacity of normal charging. Enter AP-1, "Excessive Reactor Coolant System Leakage" and attempt to isolate the leak.
- D. Leakage is in excess of allowable for continued operation. Direct a reactor trip, SI actuation and enter E-0, "Reactor Trip or Safety Injection."

Proposed Answer:

- C. The leak within the capacity of normal charging. Enter AP-1, "Excessive Reactor Coolant System Leakage" and attempt to isolate the leak.

Explanation:

A incorrect, increasing containment radiation and size of the leak precludes using O-21.

B incorrect, RCS leakage is not an entry condition for AP-25.

C correct, leakage is less than 50 gpm and within the capacity of normal charging.

D incorrect, this is for larger leaks (greater than 50 gpm).

Technical Reference(s): O-21, Isolating Leaking Thimble Tubes.

AP-1, Excessive Reactor Coolant System Leakage

Proposed references to be provided to applicants during examination: **None**

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No change from initial submittal.

K/A: EPE 009 EA2.21 - Ability to determine or interpret the following as they apply to a small break LOCA: Containment radiation trend recorder (3.9)

Question 77.

SRO Question 77

Examination Outline Cross-Reference:	Level	RO	<u>SRO</u>
	Tier:	_____	<u>1</u>
	Group:	_____	<u>1</u>
	K/A:	_____	APE 015/017G2.4.4
	Importance Rating:		4.3

Proposed Question:

GIVEN:

- PK 05-01, RCP No. 11 in alarm due to inputs:
 - 1393 RCP 1-1 Seal Leakoff Flo Hi
 - 1259 RCP 1-1 No. 2 Seal Leakoff Flo Hi
- RCP 1-1 Number 1 seal DP is 1200 psid
- RCP 1-1 Number 1 seal leakoff flow is 5.8 gpm.
- RCP 1-1 bearing outlet and number 1 seal outlet temperatures are stable.
- The crew has entered AP-28, RCP Malfunction, section B, RCP No. 1 Seal Failure

Which of the following actions should be taken by the Shift Foreman?

- A. Based on the number 1 seal leakoff flow high alarm, direct a reactor trip and a trip of RCP 1-1 and go to E-0.
- B. Based on the number 2 seal leakoff flow high alarm, direct a reactor trip and a trip of RCP 1-1 and go to E-0.
- C. Based on total seal leakoff flow greater than allowed, direct a reactor trip and a trip of RCP 1-1 and go to E-0.
- D. Based on high number 1 seal leakoff flow, direct a unit shutdown per AP-25, Rapid Load Reduction.

Proposed Answer:

- B. Based on the number 2 seal leakoff flow high alarm, direct a reactor trip and a trip of RCP 1-1 and go to E-0.

Explanation:

A incorrect. Number 1 seal leakoff flow does not warrant a reactor trip and RCP trip unless bearing outlet or seal outlet temperatures are increasing.

B correct, the presence of the alarm for number 2 seal leakoff flow high requires a reactor and RCP trip.

C incorrect, total leakoff flow is less than 8 gpm based on number 1 seal leakoff less than 7 gpm.

D incorrect, AP-25 entry is warranted if seal leakoff is greater than 6 but less than 7 gpm.

Technical Reference(s): AP-28, RCP Malfunction, rev2.

Proposed references to be provided to applicants during examination: OP AP-28, section B

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No change from initial submittal.

K/A: APE 015 G2.4.4 - RCP Malfunction - Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (4.3)

Question 78.

SRO Question 78

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>1</u>
	Group:	_____	<u>1</u>
	K/A:	_____	APE026 AA2.06
	Importance Rating:		3.1

Proposed Question:

Unit 1 is operating at 100% power.

The following alarms are received:

- PK05-01 (RCP No. 11)
- PK05-02 (RCP No. 12)
- PK05-03 (RCP No. 13)
- PK05-04 (RCP No. 14)
- PK01-08, CCW Header C

The CO reports the following RCP conditions:

- The cause for PK01-08 is FCV-357, CCW Thermal Barrier Return valve is closed.
- Cause for the RCP alarms is radial bearing temperature above 160°F
- RCP Radial Bearing Outlet temperatures are all reading approximately 205°F and stable
- RCP Seal Outlet temperatures are all reading approximately 215°F and stable
- RCP Motor Shaft Vibrations are all 2 to 3 mils and stable
- RCP Pump Shaft Vibrations are all 4 to 5 mils and stable

What action should be taken by the Shift Foreman for the current plant conditions?

- A. RCP temperatures are greater than allowed. Immediately direct a reactor trip, all RCPs tripped and go to E-0, Reactor Trip or Safety Injection.
- B. A loss of RCP thermal barrier cooling has occurred. If flow is not restored within 5 minutes, direct a reactor trip, all RCPs tripped and go to E-0, Reactor Trip or Safety Injection.
- C. RCP Radial bearing temperatures are high. Direct the operator to lower temperature by opening the Number 1 Seal Bypass Valve for each RCP in accordance with the alarm response for each RCP.
- D. A loss of RCP thermal barrier cooling has occurred. Enter OP AP-11, Malfunction of the Component Cooling Water System to investigate and correct the loss.

Proposed Answer:

D. A loss of RCP thermal barrier cooling has occurred. Enter OP AP-11, Malfunction of the Component Cooling Water System to investigate and correct the loss.

Explanation:

A incorrect, all temperatures are below the threshold to trip. Temperature is above motor bearing setpoint but not radial bearing setpoint (225°F) or seal outlet temperature (235°F)

B incorrect, a trip is required in 5 minutes if all CCW has been lost to the RCPs. Only thermal barrier flow has been lost (motor oil cooler still supplied by CCW)

C incorrect, the bypass valves are not opened if RCS pressure is above 1000 psig.

D correct, per AP-11, there is no trip criteria for a loss of CCW to thermal barrier. The cause is investigated, and AP-1 implemented.

Technical Reference(s): AR PK05-01, AP-11, Malfunction of CCW System, SECTION E: LOSS OF CCW FLOW TO RCPs

Proposed references to be provided to applicants during examination: None

Learning Objective: 8121 - State the effects of RCP operations during loss of CCW

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No changes from initial submittal.

K/A: APE 026 AA2.06 - Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The length of time after the loss of CCW flow to a component before that component may be damaged. (3.1)

Question 79.

SRO Question 79

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u> 1 </u>
	Group:	_____	<u> 1 </u>
	K/A:	_____	APE 054 G2.1.33
	Importance Rating:		4.0

Proposed Question:

The crew is stabilizing the plant at approximately 50% following a ramp from 70% in accordance with AP-15, Loss of Feedwater Flow, due to the trip of a Main Feed pump.

Why must Chemistry be notified?

- A. To determine current boron concentration.
- B. To determine if RCS pH adjustment is required.
- C. A sample for DOSE EQUIVALENT I-131 is required by Technical Specifications.
- D. A sample for gross specific activity, $100/\dot{E}$, of the RCS is required by Technical Specifications.

Proposed Answer:

C. A sample for DOSE EQUIVALENT I-131 is required by Technical Specifications.

Explanation:

A incorrect, a boron concentration will not have changed.

B incorrect, pH not changed.

C correct, IAW Tech Spec SR 3.4.16, a sample for DE I-131 following a power change of 15% or more in one hour is required 2 to 6 hours after the power change.

D incorrect. This is determined on a periodic frequency, not after power changes.

Technical Reference(s):

OP AP-15, step 18 of section A.

Technical Specification 3.4.16

Proposed references to be provided to applicants during examination: NONE.

Learning Objective: 3477 - Describe the major actions of abnormal operating procedures

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments: No change from initial submittal.

K/A: APE 054 G2.1.33 – Loss of Main Feedwater – ability to recognize indications for system operating parameters which are entry conditions for technical specifications. (4.0)

Question 80.

SRO Question 80

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>1</u>
	Group:	_____	<u>1</u>
	K/A:	_____	APE 058 AA2.03
	Importance Rating:		3.9

Proposed Question:

Unit 1 is at full power.

The following events occur:

- A loss of Vital DC bus 1-2 occurs and the crew enters AP-23, Loss of Vital DC Bus.
- The reactor is tripped. The Shift Foreman is performing the actions of AP-23.
- The Work Control Shift Foreman is implementing E-0, Reactor Trip or Safety Injection.

What action should be taken by the Work Control Shift Foreman at step 3 of E-0?

- A. Verify from the BOPCO that Vital Buses F and H are energized and implement AP-27, Loss of a Vital 4 kV and/or 480 V Bus.
- B. Verify the BOPCO has closed the breaker for Emergency Diesel 12 to energize bus G, that buses F and H are energized, then continue in E-0.
- C. Verify from the BOPCO that Vital Buses F and H energized and GO TO EOP ECA-0.3.
- D. Verify from the BOPCO that Vital Buses F and H energized and refer to EOP ECA-0.3 for restoration of Bus G while continuing in E-0.

Proposed Answer:

D. Verify from the BOPCO that Vital Buses F and H energized and refer to EOP ECA-0.3 for restoration of Bus G while continuing in E-0.

Explanation:

A incorrect. 2 vital buses energized from offsite, however, E-0 has action in the RNO to energize the de-energized bus/

B incorrect. 2 vital buses are energized. No control over the diesel at this time.

C incorrect. Do not transition to ECA-0.3. Immediate action step is to refer to ECA-0.3

D incorrect. Action is to refer to ECA-0.3

Technical Reference(s): E-0, step 3. AP-23, Loss of Vital DC Bus, and attachment B.

Proposed references to be provided to applicants during examination: None

Learning Objective: 7116 - Explain the consequences of loss of DC vital bus

Question Source:
Modified DCPD Bank A-0735

Question History: Last NRC Exam: N/A

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No change from initial submittal.

K/A: APE 058 AA2.03 - Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems. (3.9)

Question 81.

SRO Question 81

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>1</u>
	Group:	_____	<u>1</u>
	K/A:	_____	W/E11 EA2.2
	Importance Rating:		4.2

Proposed Question:

GIVEN:

- The crew is currently implementing ECA-1.1, Loss of Emergency Coolant Recirculation.
- Bus F is de-energized
- RWST level is 30%
- RCS Pressure is 450 psig
- Containment pressure is 24 psig

Per ECA-1.1, what action, if any, needs to be taken to achieve the REQUIRED combination of Containment Fan Cooler Units and Containment Spray Pumps for the current plant conditions?

- A. No action required, the required number of Containment Spray pumps are running.
- B. Stop either Containment Spray pump 1-1 or 1-2.
- C. Stop both Containment Spray pumps 1-1 and 1-2.
- D. Stop Containment Spray pump 1-2, Containment Spray pump 1-1 is de-energized.

Proposed Answer:

- B. Stop either Containment Spray pump 1-1 or 1-2.

Explanation:

A incorrect. CS pumps 1-1 and 1-2 are powered from Bus G and H respectively. Required number is 1 (2 CFCUs are de-energize), one must be secured. This would be correct if either pump was powered from bus F.

B correct. CS pumps are both running. Required number for plant conditions is 1 because 2 CFCUs are powered from Bus F.

C incorrect. This would be correct if 4 CFCUs were still available, instead of 3.

D incorrect. Neither CS pump is powered from bus F.

Technical Reference(s): ECA-1.1 step 6 and attachment A

Proposed references to be provided to applicants during examination: ECA-1.1 page 3 (step 6)

Learning Objective: SIM0152 Demonstrate the Proper Use of the EOPs

Question Source:

Modified - Indian Point NRC Exam 12/04

Question History: Indian Point 12/04

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No change from initial submittal. Only giving page 6, (step 3)

K/A: W/E11 EA2.2 Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. (4.2)

Question 82.

SRO Question 82 rev1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>1</u>
	Group:	_____	<u>2</u>
	K/A:	_____	APE 003 G2.4.49
	Importance Rating:		4.0

Proposed Question:

Unit 1 is recovering from a dropped control rod in accordance with OP AP-12C, "Dropped Control Rod."

At step 13, while opening the lift coils to recover the dropped rod, a second control rod drops into the core.

What action should the Shift Foreman take?

- A. Direct a trip of the reactor and GO TO EOP E-0, Reactor Trip or Safety Injection.
- B. Direct the operator to reduce turbine load to match Tave and Tref, finish the current rod recovery, then repeat the procedure for the other rod.
- C. Direct the operator to terminate the recovery and not to reset any URGENT alarms and contact I&C immediately to investigate the problem.
- D. Direct the operator to terminate the recovery and within one hour, verify SDM to be within the limits provided in the COLR and make preparations to be in MODE 3 within the next 6 hours.

Proposed Answer:

- A. Direct a trip of the reactor and GO TO EP E-0, Reactor Trip or Safety Injection.

Explanation:

Only A is correct. Currently the crew would be at step 13 of the procedure, preparing the dropped rod for recovery. When the second one drops, there are now two dropped rods. Per step 1 of the procedure (not a continuous action), if there are 2 dropped rods, then the reactor is tripped. It would not be appropriate to continue with the recovery with a second dropped rod.

Technical Reference(s): AP-12C step 1 and step 13, rev10.

Proposed references to be provided to applicants during examination: None

Learning Objective: 9934 - Explain the operator actions if more than 1 control rod drops

Question Source:

Bank DCP P-5466

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.6 - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Comments: Reformatted question, answer and distractors.

K/A: APE 003 G2.4.49 – Dropped Control Rod, Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (4.0)

Question 83.

SRO Question 83

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>1</u>
	Group:	_____	<u>2</u>
	K/A:	_____	APE 036 AA2.02
	Importance Rating:		4.1

Proposed Question:

Unit 1 refueling is in progress.

The RED overload light comes on at the Containment Manipulator Crane Console. The SRO suspects the fuel assembly being moved has been damaged.

Which of the following would be a positive indication to the SRO that the fuel element has been damaged?

- A. Increased drag indicated on the Dillon Cell.
- B. Increasing radiation levels on RE-58 and RE-59.
- C. Report from the Control Room of increasing Source Range counts.
- D. Gas bubbles coming to the surface of the water in the refueling cavity.

Proposed Answer:

- D. Gas bubbles coming to the surface of the water in the refueling cavity.

Explanation:

A incorrect, increased drag would be an indication of the assembly possibly resisting movement but not that it has actually been damaged. NOTE, if the limit is exceeded, then it's a possible, additional indication.

B incorrect, these radiation monitors are in the SFP area.

C incorrect, this would be an indication of approaching criticality but not fuel damage.

D correct, a damaged fuel assembly would release the gas in the assembly. The gas comes to the surface which would indicate the integrity of the assembly is compromised.

Technical Reference(s): AP-21, rev 10, step 2 - Symptoms.

Proposed references to be provided to applicants during examination: None

Learning Objective: 6619 - Explain the actions for fuel damage, actual or suspected

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.43.7 - Fuel handling facilities and procedures.

Comments: No changes from initial submittal.

K/A: APE 036 AA2.02 - Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurrence of a fuel handling incident (4.1)

Question 84.

SRO Question 84

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>1</u>
	Group:	_____	<u>2</u>
	K/A:		APE 037 G2.1.33
	Importance Rating:		4.0

Proposed Question:

GIVEN:

- Unit 1 is at full power
- RCS IDENTIFIED and primary to secondary leakage at 1200 is as follows:
 - Total IDENTIFIED LEAKAGE is 9.5 gpm
 - Primary to Secondary leakages:
 - 0.03 gpm to Steam Generator 11
 - 0.03 gpm to Steam Generator 12
 - 0.00 gpm to Steam Generator 13
 - 0.04 gpm to Steam Generator 14

Subsequent Steam Generator leakages are determined as follows:

TIME	1230	1300	1330
Steam Generators 11	0.06	0.08	0.08
Steam Generator 12	0.08	0.09	0.11
Steam Generator 13	0.00	0.00	0.00
Steam Generator 14	0.06	0.08	0.09

For the times above, when, if at all, was the Shift Foreman required to enter LCO 3.4.13, RCS OPERATIONAL Leakage, ACTION A, for leakage in excess of Technical Specifications?

- A. 1230
- B. 1300
- C. 1330
- D. No leakage limits have been exceeded as of 1330.

Proposed Answer:

- C. 1330

Explanation:

A incorrect. At 1230, Identified is 9.4 gpm + 0.20 gpm or 9.6 gpm. No steam generators are above 150 gpd.

B incorrect. At 1300 Identified is 9.4 + 0.25 or 9.65. No steam generators are above 150 gpd.

C correct. At 1330, identified is 9.4 + 0.28 for a total of 9.68, however, primary to secondary limits have been exceeded in the 12 steam generator - limit is 150 gpd or .104 gpm. Currently leakage is 0.11 gpm or 158.4 gpd.

D incorrect. The LCO was entered at 1330.

Technical Reference(s): TS 3.4.13, TS definitions

Proposed references to be provided to applicants during examination: None

Learning Objective: 9697D - Identify 3.4 Technical Specification LCOs

Question Source:

New

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments: No changes from initial submittal.

K/A: APE 037 G2.1.33 – Steam Generator Tube Leak - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (4.0)

Question 85.

SRO Question 85

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>1</u>
	Group:	_____	<u>2</u>
	K/A:	_____	W/E16 EA2.1
	Importance Rating:		3.3

Proposed Question:

GIVEN:

- A large LOCA has occurred
- Containment pressure is 3 psig
- ECCS system is aligned for Cold Leg Recirculation
- The BOPCO reports Containment radiation is 10 R/hr on RM-30/31.
- After leaving E-1.3, Transfer to Cold Leg Recirculation, the Shift Foreman enters FR-Z.3, Response to High Containment Radiation Level

When informed of the Containment radiation levels, the TSC recommends starting a Containment Spray pump to reduce the radiation levels.

Which of the following actions should be taken by the Shift Foreman?

- A. Direct the BOPCO to start a Containment Spray pump.
- B. Inform the TSC that Containment Spray is not an option at this time and return to E-1, Loss of Reactor or Secondary Coolant.
- C. Refer to FR-Z.1, Response to High Containment Pressure, for guidance on starting a Containment Spray pump.
- D. Refer to E-1.3, Transfer to Cold Leg Recirculation, for guidance on starting a Containment Spray pump.

Proposed Answer:

D. Refer to E-1.3, Transfer to Cold Leg Recirculation, for guidance on starting a Containment Spray pump.

Explanation:

A incorrect. Because the plant is in recirc, starting the pump at this time will be starting it without a suction path.

B incorrect. Containment spray is an option.

C incorrect. Guidance is in E-1.3.

D correct. Because of the current plant alignment, guidance in E-1.3 is necessary.

Technical Reference(s): FR-Z.3, page 2.

Proposed references to be provided to applicants during examination: None

Learning Objective: 6177 - State the means of reducing containment radiation levels

Question Source:

Bank P-49251, modified

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No changes from initial submittal.

K/A: E16 EA2.1 - Ability to determine and interpret the following as they apply to the (High Containment Radiation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (3.3)

Question 86.

SRO Question 86 rev1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u> 2 </u>
	Group:	_____	<u> 1 </u>
	K/A:		003 A2.05
	Importance Rating:		2.8.

Proposed Question:

GIVEN:

- The crew is making preparations to start RCP 1-2 using Attachment B in E-0.2, “Natural Circulation Cooldown”.
- Power is available
- Lift oil pump is running
- RCS pressure is 350 psig
- Seal Injection flow is 8 gpm to each RCP
- Seal Leakoff flow is 0.15 gpm from each RCP
- Number 1 seal leakoff temperature is 100°F
- All bearing temperatures are less than 150°F
- VCT pressure is 50 psig
- Both Spray valves are in AUTO

Which of the following actions should the Shift Foreman direct the operator to perform?

- A. Direct the operator to start RCP 1-2, all conditions are satisfied.
- B. Direct the operator to start RCP 1-2, all conditions are not satisfied but are not required.
- C. Direct the operator to decrease VCT pressure to increase seal leakoff flow, then RCP 1-2 may be started.
- D. Direct the operator to open RCP No. 1 Seal Bypass Valve valve, 8142 to increase seal leakoff flow, then RCP 1-2 may be started.

Proposed Answer:

C. Direct the operator to decrease VCT pressure to increase seal leakoff flow, then RCP 1-2 may be started.

Explanation:

A incorrect, not all conditions are met, seal leakoff flow is low.

B incorrect, this is not a procedure in which RCPs are started if normal conditions are not met..

C correct, decreasing VCT pressure will increase seal leakoff flow.

D incorrect, all conditions are met to allow opening 8142, however, the valve should not be opened unless either the pump radial bearing outlet temperature or the No. 1 seal leakoff temperature approaches its alarm level, neither of these conditions exist.

Technical Reference(s): E-0.2, attachment B and OP A-6:I

Proposed references to be provided to applicants during examination: None

Learning Objective: 4892 - State the cause/effect relationship between VCT and RCPs

Question Source:

Diablo Canyon 2005 SRO

Question History: Last NRC Exam: DCPD 2005

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Question reformatted to be reformatted to SRO level.

K/A: 003 A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of VCT pressure on RCP seal leakoff flows (2.8)

Question 87.

SRO Question 87

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>2</u>
	Group:	_____	<u>1</u>
	K/A:		006 G2.4.31
	Importance Rating:		3.4

Proposed Question:

Unit 1 is at full power.

The following events occur:

- PK 02-05, ACCUM PRESSURE HI-LO and PK 02-10, ACCUM LEVEL HI-LO alarm
- The operator reports Accumulator 1-1 pressure is 590 psig, level is 59% and both are decreasing slowly.
- The level and pressure stabilize at 585 psig and level is 50%, once the Nuclear Operator verifies NSS-1-9352A, Accumulator Sample Line Isolation, closed.

Which of the following describes the OPERABILITY status of Accumulator 1-1 and the action that should be taken first by the Shift Foreman?

- A. Inoperable. Refer to AR PK02-05 to raise pressure.
- B. Inoperable. Refer to AR PK02-10, to raise level.
- C. OPERABLE. Refer to AR PK02-05 to raise pressure.
- D. OPERABLE. Refer to AR PK02-10, to raise level.

Proposed Answer:

- B. Inoperable. Refer to AR PK02-10, to raise level.

Explanation:

A incorrect. Accumulator is inoperable, however, the action that should be taken is to raise level and then if necessary adjust pressure.

B correct. Accumulator is inoperable. The action to take is to raise level, which will raise pressure. Once level is restored, pressure is adjusted, if necessary.

C and D incorrect. Per STP 1-1A, TS level is 52%. Pressure is above the required pressure, however, Accumulator are inoperable due to the low level.

Technical Reference(s): TS 3.5.1, STP 1A

Proposed references to be provided to applicants during examination: STP I-1A (at least Accumulator page) and TS 3.5.1

Learning Objective: 35326 - State the ECCS parameters that produce alarms in the control room.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No changes to initial submittal.

K/A: 006 G2.4.31 – ECCS - Knowledge of annunciators alarms and indications, and use of the response instructions. (3.4)

Question 88.

SRO Question 88 r1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>2</u>
	Group:	_____	<u>1</u>
	K/A:	_____	061 G2.2.25
	Importance Rating:		3.7

Proposed Question:

According to the Bases in Technical Specifications for auxiliary feedwater, (AFW), what is the MINIMUM AFW alignment for adequate decay heat removal following an assumed feedline break accident?

- A. Two motor driven AFW pumps or the turbine driven AFW pump to deliver flow to a total of 3 steam generators.
- B. One motor driven AFW pump and the turbine driven AFW pump to deliver flow to a total of 3 steam generators.
- C. One motor driven and the turbine driven AFW pump to deliver flow to two steam generators.
- D. One motor driven AFW pump to deliver flow to two steam generators.

Proposed Answer:

D. One motor driven AFW pump to deliver flow to two steam generators

Explanation:

A incorrect. Only one MDAFW pump to 2 SGs assumed.

B incorrect. Flow assumed to only 2 SGs.

C incorrect. TDAFW pump not assumed.

D correct. According to TS Bases 3.7.5 - The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and *a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the broken MFW header at the pump maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.*

Technical Reference(s): B3.7.5

Proposed references to be provided to applicants during examination: None

Learning Objective: 9694G - Discuss 3.7 Technical Specification bases

Question Source:

Bank – DCPB B-0401

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content:

55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments: Reformatted question for clarity. Removed acronym for TDAFW, replaced with turbine driven AFW

K/A: 061 G2.2.25 - AFW, Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (3.7)

Question 89.

SRO Question 89 r2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u> 2 </u>
	Group:	_____	<u> 1 </u>
	K/A:		064 A2.19
	Importance Rating:		2.7

Proposed Question:

GIVEN:

- Unit 1 is at 50% power
- Diesel Generator 1-2 is fully loaded and running in accordance with STP-M9A, Diesel Engine Generator Routine Surveillance Test.
- Load is 2.5 MW and 1.5 MVARs out

The crew is preparing to start Circulating Water pump 1-1.

Which of the following actions should be taken by the Shift Foreman?

- A. Direct the BOPCO to reduce MVARs, then allow the pump to be started.
- B. Direct the BOPCO to reduce MW load, then allow the pump to be started.
- C. Allow the pump to be started, there should be no effect on diesel MW loading or MVARs.
- D. Direct the BOPCO to closely monitor MVARs and MW loading, while the pump is started.

Proposed Answer:

- A. Direct the BOPCO to reduce MVARs, then allow the pump to be started.

Explanation:

A correct. Paralleled to the grid, starting a large load can affect the VAR loading of the diesel. The procedure precaution and limitation states MVAR loading should be reduced to less than 0.5 MVAR prior to starting large 12 KV or 4KV loads. STP M-9A appendix 8.1 shows that for a 0.8 pf, MVARS should be no more than 1.85. For this condition and because the procedure limit is 0.5 MVAR for starting large reactive loads, MVARS must be reduced.

B incorrect. Real load should not be affected by the start of the pump.

C incorrect. While not powered from the diesel, the reactive load of the pump will affect the diesel because it is currently paralleled with aux power.

D incorrect. Action is to reduce VARs prior to starting a large load to prevent exceeding limits.

Technical Reference(s): M-9A, Diesel Engine Generator Routine Surveillance Test

Proposed references to be provided to applicants during examination: None

Learning Objective: 6408 - Describe the precautions and limitations for the Diesel Engine Generator system.

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Modified distractor D. Raised MVAR load in setup to be closer to what is allowed.

K/A: 064 A2.19 - Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Consequences of high VARS on ED/G integrity (2.7)

Question 90.

SRO Question 90

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>2</u>
	Group:	_____	<u>1</u>
	K/A:	_____	103 G2.1.33
	Importance Rating:		4.0

Proposed Question:

The plant is operating at 100% power.

GIVEN:

- Narrow range containment pressure recorder (YR 26) and PPC Point (P0706A) are reading 0.9 psig
- Containment temperature:
 - TE – 85, 120°F
 - TE – 87, 122°F
 - TE – 89, 117°F
 - TE – 91, 118°F

What Technical Specification LCO, if any, has been exceeded?

- A. None. Containment parameters are within Technical Specifications.
- B. Only Technical Specification 3.6.4, Containment Pressure.
- C. Only Technical Specification 3.6.5, Containment Temperature.
- D. Technical Specifications 3.6.4, Containment Pressure and 3.6.5, Containment Temperature.

Proposed Answer:

- A. None. Containment parameters are within Technical Specifications.

Explanation:

A is correct. Pressure limit (upper) is 1.2 psig. Temperature limit is 120F, average. Current temperature average is just over 119F.

B incorrect. Limit is 1.2 psig (note lower limit is -1.0 psig)

C incorrect. Temperature limit is an *average* of the temperature readings.

D incorrect. Both are within specs.

Technical Reference(s): Tech Spec 3.6.4 and 3.6.5

Proposed references to be provided to applicants during examination: None

Learning Objective: 9697F - Identify 3.6 Technical Specification LCO

Question Source:

New

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments: No changes made to initial submittal.

K/A: 103 G2.1.33 - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (4.0)

Question 91.

SRO Question 91 r1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>2</u>
	Group:	_____	<u>2</u>
	K/A:		017 A2.02
	Importance Rating:		4.1

Proposed Question:

GIVEN:

- The crew has transitioned from E-0, Reactor Trip or Safety Injection, to E-1, Loss of Reactor or Secondary Coolant.
- A loss of offsite power has occurred.
- Emergency Diesel Generators are providing power to all vital buses.
- RCS pressure is 1800 psig.
- Core Exit Thermocouples are 1125°F
- RVLIS full range level is 29% and increasing slowly

Which of the following actions should be taken by the Shift Foreman?

- A. Core cooling is adequate, remain in E-1.
- B. The core is uncovered and an inadequate core cooling condition exists, entry into FR-C.1, Response to Inadequate Core Cooling is required.
- C. A degraded core cooling condition exists, entry into FR-C.2, Response to Degraded Core Cooling is required.
- D. A severe accident and core damage has probably occurred, entry into SACRG-1, Severe Accident Control Room Guideline Initial Response is required.

Proposed Answer:

B. The core is uncovered and an inadequate core cooling condition exists, entry into FR-C.1, Response to Inadequate Core Cooling is required.

Explanation:

A incorrect. Conditions of inadequate core cooling exist. A transition is required.

B correct. Temperature greater than 700°F, no RCPs and RVLIS below 32% is entry to C.1

C incorrect. RVLIS level is too low.

D incorrect. This is required if thermocouples are greater than 1200°F and only from C.1 (or ECA-0.0).

Technical Reference(s): F-0.2

Proposed references to be provided to applicants during examination: none.

Learning Objective: 9703 - Identify exit conditions for the FRPs

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Question rewritten to align with KA.

K/A: 017 A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Core damage (017)

Question 92.

SRO Question 92

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>2</u>
	Group:	_____	<u>2</u>
	K/A:	_____	034 A1.02
	Importance Rating:		3.7

Proposed Question:

Refueling is in progress one week after the reactor was shutdown.

A spent fuel assembly is dropped while the operator is removing the assembly from the core.

For this accident, which of the following was verified prior to fuel movement to ensure that the 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 limits?

- A. One train of Containment Ventilation Isolation instrumentation OPERABLE.
- B. Two trains of Fuel Handling Building Ventilation System OPERABLE.
- C. Refueling Cavity Concentration greater than limit in the COLR.
- D. Refueling Cavity Level greater than 23 feet.

Proposed Answer:

D. Refueling Cavity Level greater than 23 feet.

Explanation:

A incorrect. CVI instrumentation is required during movement of “recently irradiated fuel” or fuel that has been in a critical reactor less than 100 hours ago.

B incorrect. Only applies during movement of recently irradiated fuel.

C incorrect. Ensures subcriticality.

D correct. per TS bases 3.9.7 - Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits

Technical Reference(s): TS bases 3.9.7, OP B-8

Proposed references to be provided to applicants during examination: None

Learning Objective: 6497 - State the responsibilities and duties of Refueling SRO

Question Source:

New

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.7 - Fuel handling facilities and procedures.

Comments: No changes from initial submittal.

K/A: 034 A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal (3.7)

NOTE: From ES-401 - A number of the generic K/As in Section 2 of the catalogs are specifically linked to one or more topics specified in 10 CFR 55.43(b), and all of the Category A2, AA2, and EA2 K/A statements are (or, in the case of NUREG-1123, should be) similarly linked. Consequently, the K/As for the SRO examination will be drawn from those K/A categories (denoted by Columns "A2" and "G" in the SRO-only section of the applicable examination outline) **and from all K/A categories related to the fuel handling facilities, which are specifically identified for sampling in 10 CFR 55.43(b)(7).**

Question 93.

SRO Question 93 rev 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>2</u>
	Group:	_____	<u>2</u>
	K/A:	_____	075 A2.02
	Importance Rating:		2.7

Proposed Question:

GIVEN:

- The plant is at 45% power.
- A High Swell Warning is in effect.
- Actions of O-28, Intake Management are being implemented.
- Both Circulating Water pumps are running.
- PK13-01 - BAR RACKS/SCREENS is in alarm.
- Both screens have 15 to 20 inches of ΔP .

The operator monitoring the screens reports ΔP has increased to 80 inches on the screens for both Circulating Water pumps. One Circulating Water pump trips on overcurrent. ΔP for the running pump has not decreased.

Which of the following actions should the Shift Foreman direct the crew to perform?

- A. Trip the reactor.
- B. Perform a rapid load reduction to less than 25%.
- C. Trip the turbine and the running Circulating Water pump.
- D. Verify Screens are running in FAST and monitor Condenser ΔP s.

Proposed Answer:

- A. Trip the reactor.

Explanation:

A correct. ΔP greater than 70", with only 1 CWP running above 25% requires a reactor trip.

B incorrect. This action would allow the crew to perform an orderly shutdown but is not allowed for the current conditions..

C incorrect. This action is taken below 25%.

D incorrect. This is an action if ΔP is high, per O-28 and another section of AP-7, this already done.

Technical Reference(s): OP AP-7, Degraded Condenser, rev 34.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3477 - Describe the major actions of abnormal operating procedures

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Question modified to match KA.

K/A: 075 A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of Circulating Water pumps (2.7)

Question 94.

SRO QUESTION 94

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u> 3 </u>
	Group:	_____	<u> 1 </u>
	K/A:		G2.1.20
	Imp. Rating:	_____	<u> 4.2 </u>

Proposed Question:

Given the following conditions:

- Unit 1 is at full power when Safety Injection actuates.
- The crew is performing E-0, Reactor Trip Or Safety Injection.
- The BOPCO has been directed to perform Appendix E, ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS.

The SFM determines that a transition to E-1, Loss Of Reactor Or Secondary Coolant, is necessary. The BOPCO reports only the first 6 steps of Appendix E are complete.

Which of the following actions is required?

- A. Wait for the completion of all actions in Appendix E prior to transitioning to E-1.
- B. Wait for the completion of actions through step 8 in Appendix E prior to transitioning to E-1.
- C. Transition to E-1 and direct the BOPCO to continue Appendix E actions.
- D. Transition to E-1 and direct the BOPCO to discontinue the actions in Appendix E.

Proposed Answer:

- C. Transition to E-1 and direct the BOPCO to continue Appendix E actions.

Explanation:

A incorrect. The transition to E-1 should be made when the SFM determines it is required.

B incorrect. No wait is required.

C is correct. From LPE0: • If App E is not completed by the transition out of E-0, the SFM shall be notified of progress and the SFM shall determine how to proceed

- In most cases all of App E is expected to be done prior to transition out of E-0, but ESF verifications through step 8 (as a minimum) should be done (not set aside for other tasks) at that point

D incorrect. Appendix E should be completed.

- *The expectation is to complete App E, even if some steps remain after transition from E-0 (SFM call if an extreme resource problem arises)*

Technical Reference(s): Lesson LPE0.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3705 - Explain the responsibilities of the control room staff positions during emergencies

Question Source:
Bank

Question History: Last NRC Exam Beaver Valley 5/2005

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No changes from initial submittal.

K/A: G2.1.20 – Ability to execute procedure steps.

Question 95.

SRO QUESTION 95

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>3</u>
	Group:	_____	<u>2</u>
	K/A:	_____	G2.2.19
	Importance Rating:		3.1

Proposed Question:

Which of the following describes the type of work classified as “Emergent Work”?

- A. Work added to the daily schedule after scope freeze.
- B. Work assigned to the operations support team (OST).
- C. Immediate maintenance to prevent or mitigate the release of radioactive material.
- D. Work performed to restore a plant SSC with identified potential degradation that does not affect the SSC's design function or performance.

Proposed Answer:

- A. Work added to the daily schedule after scope freeze.

Explanation:

A correct, per procedure definition on page 6. “Work added to the daily schedule after scope freeze. Emergent work does not include OST work.”

B incorrect, this is OST work.

C incorrect, this is Emergency Maintenance.

D incorrect, this is Elective Maintenance.

Technical Reference(s): AD7.DC8 – Work Control

Proposed references to be provided to applicants during examination: None

Learning Objective: 7945 - State the definition of emergent work

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.43.3 - Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments: No changes to initial submittal.

K/A: G2.2.19 – Knowledge of maintenance work order requirements. (3.1)

NOTE: knowledge of the work process is a function of the Work Control Lead (WCL), an SRO position.

Question 96.

SRO Question 96

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>3</u>
	Group:	_____	<u>2</u>
	K/A:	_____	G2.2.23
	Importance Rating:		3.8

Proposed Question:

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.

Proposed Answer:

- A. Tracking is not done.

Explanation:

Only A is correct. From OP1.DC17, step 2.1.2 - During refueling outages (Modes 5 or 6) or at other times as approved by the operations manager, tracking of Info TSs will not be done. Elimination of Info TSs reduces unnecessary work tracked by other methods such as Mode transition checklist and clarifies the important TSs for the plant status.

Technical Reference(s): OP1.DC17, step 2.1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 3503 - Discuss administration requirements of Tech Spec status sheets

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content:

55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments: No changes from initial submittal.

K/A: G2.2.23 - Ability to track limiting conditions for operations. (3.8)

Question 97.

SRO Question 97 rev1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>3</u>
	Group:	_____	<u>2</u>
	K/A:	_____	G2.3.4
	Importance Rating:		3.1

Proposed Question:

A Site Area Emergency has been declared due to a LOCA Outside Containment.

An operator volunteers to make an emergency entry into the penetration area to isolate the leak. Isolating the leak would result in a significant reduction in the offsite dose.

Which of the following is the MAXIMUM exposure the Site Emergency Coordinator (SEC) may authorize for this situation?

- A. 4,500 mrem TEDE
- B. 5,000 mrem TEDE
- C. 10,000 mrem TEDE
- D. 25,000 mrem TEDE

Proposed Answer:

- D. 25,000 mrem TEDE

Explanation:

A incorrect, this what is automatically assumed for events of Alert or higher

B incorrect, this is the federal limit and emergency exposure limit for radiological assessment sampling

C incorrect, this is the dose to protect valuable equipment

D correct, to save a life or for dose saving to population, 25 rem is the guideline. Note, if it is an extreme situation, there is no limit but only to volunteers who are fully aware of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and the numerical estimates of the risk of delayed effects.

Technical Reference(s): EP RB-2, DCPD Emergency Exposure Guidelines

Proposed references to be provided to applicants during examination: none

Learning Objective: 7954 - State the emergency dose limits

Question Source: Bank # NRC

Question History: Last NRC Exam DCPD 2005

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.43 43.4 - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments: Modified question to address action to be taken by the SEC.

K/A: G2.3.4 - Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Question 98.

SRO Question 98

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>3</u>
	Group:	_____	<u>3</u>
	K/A:	_____	G2.3.6
	Importance Rating:		3.1

Proposed Question:

The crew is preparing to discharge the Laundry and Hot Shower tank to the outfall.

What Chemistry and SFM signatures are required for the discharge to proceed?

- A. The SFM must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.
- B. The SFM must sign the Discharge Checklist. Only Chemistry must sign the Discharge Permit.
- C. Chemistry must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.
- D. Both the SFM and Chemistry must sign the Discharge Checklist and the Discharge Permit.

Proposed Answer:

A. The SFM must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.

Explanation:

A correct. The SFM must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.

B incorrect. Both sign the Discharge Permit.

C incorrect. The SFM signs the Discharge Checklist.

D incorrect. Only the SFM must sign the Discharge Checklist.

Technical Reference(s): OP G-1:II, Discharge of Liquid Radwaste, rev. 32, page 1 and attachment 9.1.

Proposed references to be provided to applicants during examination: None

Learning Objective: 8443 - State the administrative requirements of Liquid Rad Waste system

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.43.4 - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments: No changes to initial submittal.

K/A: 2.3.6 – Knowledge of the requirements for reviewing and approving release permits. (3.1)

Question 99.

SRO Question 99

SRO Question 99 SRO Question 99
Examination Outline Cross-Reference:

Level	RO	SRO
Tier:	_____	<u>3</u>
Group:	_____	<u>4</u>
K/A:	_____	G2.4.4
Importance Rating:		4.3

Proposed Question:

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

The operator reports the following:

- Pressurizer pressure is 2250 psig and slowly increasing
- Pressurizer spray valves are closed
- Letdown Orifice Valves are closed
- Charging flow is decreasing
- Pressurizer Level 54% and increasing
- 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

Proposed Answer:

- B. AP-9, Loss of Instrument Air

Explanation:

A incorrect. If a level channel had failed the spray valves would still operate. If a pressure channel failed, letdown would not be isolated.

B correct. Spray valves and letdown orifices would close on loss of air to containment. Loss of letdown causes pressurizer level to increase, which will cause charging to decrease.

C incorrect. System is responding to increasing pressurizer level.

D incorrect, conditions do not reflect those of a break in letdown.

Technical Reference(s): AP-9

Proposed references to be provided to applicants during examination: None

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures

Question Source:

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No changes from initial submittal.

K/A: G2.4.4 – Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (4.3)

Question 100.

SRO Question 100

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	<u>3</u>
	Group:	_____	<u>4</u>
	K/A:	_____	G2.4.5
	Importance Rating:		3.6

Proposed Question:

GIVEN:

- Reactor trip and safety injection from 100% power due to a Main Steamline break upstream of the 1-1 MSIV.
- The operator reports increasing radiation on steamline 1-3.
- PK11-18, MAIN STM LINE HI RAD is ON
- NO RED or MAGENTA paths exist on the CSFSTs.

The proper procedure flowpath for this event will be E-0 to....

NOTE:

- E-2, Faulted Steam Generator Isolation
- E-3, Steam Generator Tube Rupture
- ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired

- A. E-2 to E-3.
- B. E-2 to E-3 to ECA-3.1.
- C. E-3 to E-2.
- D. E-3, foldout page transition to E-2 and then back to E-3.

Proposed Answer:

- A. E-2 to E-3

Explanation:

A correct. E-0 diagnostic steps are E-2, E-3, E-1. Crew would first go to E-2.

B incorrect, although there is a faulted SG, it is not the ruptured SG, so a transition to ECA-3.1 is not appropriate.

C incorrect, E-3 is performed after the faulted SG is isolated in E-2.

D incorrect, although there is a transition to E-2 from the E-3 foldout page, this would mean an improper transition to E-3 was made from E-0.

Technical Reference(s): E-0, rev30, page 9.

Proposed references to be provided to applicants during examination: None

Learning Objective: 6827 - State the entry conditions for emergency procedures

Question Source:

Bank – DCPB B-0076 - modified

Question History: Last NRC Exam: N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: No changes to initial submittal.

K/A: G2.4.5 - Knowledge of the organization of the operating procedures network for normal, abnormal and emergency evolutions. (3.6)

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK04-12
Rev. 13
Page 1 of 2

**REACTOR
TRIP
INITIATE**

08/31/05

Effective Date

QUALITY RELATED
CONTAINS REACTOR TRIP CRITERIA

1. **ALARM INPUT DESCRIPTION**

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
16	Pzr Lo Press and P7 2/4 React Trip	K0202	< 1950 psig	2.1
17	Pzr Hi Press 2/4 React Trip	K0203	> 2385 psig	2.1
18	Pzr Hi Lvl and P7 2/3 React Trip	K0204	> 90% of span	2.1
24	Stm Gen 1-1 Lo-Lo Lvl 2/3 React Trip	K0210	< 15%	2.1
25	Stm Gen 1-2 Lo-Lo Lvl 2/3 React Trip	K0211	< 15%	2.1
26	Stm Gen 1-3 Lo-Lo Lvl 2/3 React Trip	K0212	< 15%	2.1
27	Stm Gen 1-4 Lo-Lo Lvl 2/3 React Trip	K0213	< 15%	2.1
29	RCP Busses UF and P7 1/2 React Trip	K0301	< 54 hz	2.1
490	Manual React Trip	1RT/CS 2RT/CS		2.1

2. **OPERATOR ACTIONS**

2.1 **General Actions (All Inputs)**

- 2.1.1 IF plant conditions require a Reactor trip AND an automatic Reactor trip has NOT occurred, THEN manually TRIP the Reactor. []
- 2.1.2 IF at any time a reactor trip or safety injection occurs OR has been initiated, THEN GO TO EOP E-0, "Reactor Trip or Safety Injection". []

2.1.3 Probable Causes

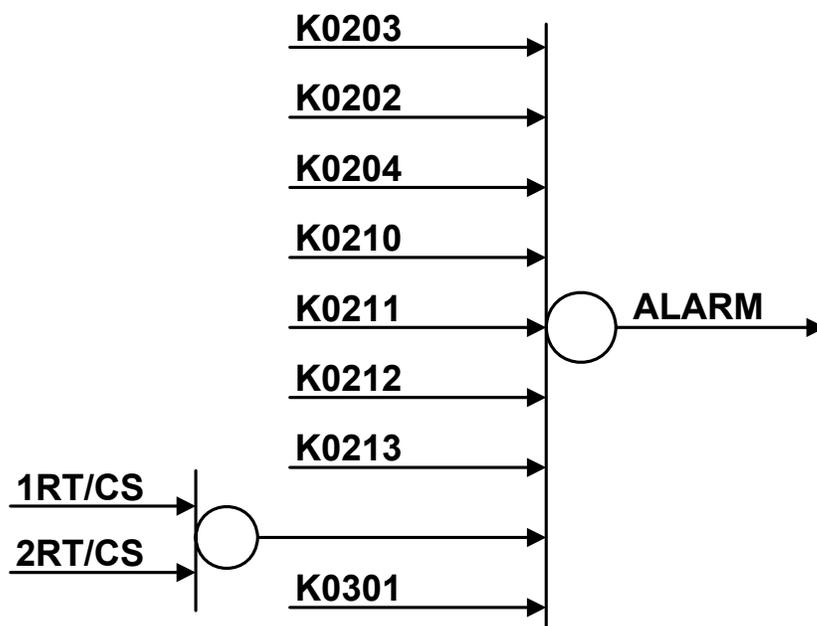
- The required number of reactor trip bistables are activated

3. AUTOMATIC ACTIONS

- 3.1 Reactor trip
- 3.2 Turbine - generator trip
- 3.3 Steam dump activation

4. REFERENCES

- 4.1 501125, "Electrical Schematic Diagram – Main Annunciator" (Electrical Drawing Section 8)

5. LOGIC DIAGRAM

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

15. CHECK If One CCP Should Be Stopped:

NOTE 1: If a steam or feed line break has occurred inside the aux building, then loop 1 WR T_h and/or T_c may fail low.

NOTE 2: After stopping any ECCS Pp, RCS Pressure should be allowed to stabilize or increase before stopping another ECCS Pp.

NOTE 3: The CCPs and SI Pps should be stopped on alternate ECCS Trains when possible.

a. Check both CCPs - RUNNING

a. GO TO step 16 (Page 13).

b. Determine required RCS Subcooling from table:

SI Pump Status	RCS SUBCOOLING (°F)	
	Any RCP Running	No RCP Running
None Running	74°F	87°F
One Running	53°F	53°F
Two Running	53°F	53°F

c. Check RCS Subcooling Based on core exit T/Cs GREATER THAN REQUIRED SUBCOOLING (Subcooled Margin Monitor, YI-31 or Steam Tables)

c. IF RCS Hot Leg WR Temperatures GREATER THAN 350°F,
THEN GO TO step 24 (Page 19).

IF RCS Hot Leg WR Temperatures LESS THAN 350°F,
THEN Start one RHR Pp in SI mode if none running.

IF At least one RHR Pp CANNOT be operated,

THEN GO TO step 24 (Page 19).

d. Check PZR Level - GREATER THAN 17% [50%]

d. RETURN TO step 13 (Page 9).

e. Depress Vital 4KV Auto Transfer Relay Resets: Blue Light - OFF

f. Stop one CCP

STEP DESCRIPTION TABLE FOR ES-1.2 Step 13 – NOTE 1

NOTE: After stopping any SI pump, RCS pressure should be allowed to stabilize or increase before stopping another SI pump.

PURPOSE: To remind the operator that RCS pressure should be allowed to stabilize before stopping another SI pump

BASIS:

After an SI pump is stopped, RCS pressure may decrease rapidly to a new equilibrium value where the reduced SI flow again matches leakage from the RCS. The criteria for stopping the next SI pump has been calculated assuming steady-state conditions. Hence, to ensure that these criteria are appropriate, RCS pressure and subcooling should be allowed to stabilize or increase before additional SI pumps are stopped.

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

RCS pressure may continue to decrease slowly as the reactor coolant temperature is reduced. However, if subcooling is increasing, the SI reduction criteria are appropriate and the SI flow can be further reduced when such criteria are satisfied.

PLANT-SPECIFIC INFORMATION:

N/A

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

16. CHECK If One SI Pp Should Be Stopped:

- a. Check any SI Pp - RUNNING

- b. Determine required RCS Subcooling from table

a. GO TO step 17 (Next Page).

SI Pump Status	RCS SUBCOOLING (°F)			
	One Cent. Charging Pump Running		No Cent. Charging Pump Running	
	Any RCP Running	No RCP Running	Any RCP Running	No RCP Running
One Running	153°F	171°F	Do not stop SI PP GO TO step 16c RNO	Do not stop SI PP GO TO step 16c RNO
Two Running	39°F	43°F	49°F	55°F

- c. Check RCS Subcooling based on core exit T/Cs - GREATER THAN REQUIRED SUBCOOLING (Subcooled Margin Monitor, YI-31 or Steam Tables)

c. IF RCS Hot Leg WR Temperatures GREATER THAN 350°F,
THEN GO TO step 24 (Page 19).

IF RCS Hot Leg WR Temperatures LESS THAN 350°F,
THEN Start one RHR Pp in SI mode if none running.

IF At least one RHR Pp CANNOT be operated,
THEN GO TO step 24 (Page 19).

- d. Check PZR Level - GREATER THAN 17% [50%]

- e. Stop one SI Pp

- f. RETURN TO step 16a

d. RETURN TO step 13 (Page 9).

2. INTRODUCTION

The function of the SI reduction tables in the guidelines is to decrease RCS pressure by reducing SI flow while still maintaining pressurizer pressure control and RCS subcooling. For a small break loss of coolant accident RCS pressure can equilibrate at a pressure that exceeds the RHR operational limits. If pressurizer pressure control is used to decrease RCS pressure to the RHR operational limits, safety injection flow will increase and ultimately fill the pressurizer rendering the pressurizer control system useless. An alternative to using pressurizer pressure control to decrease RCS pressure is to decrease safety injection flow. For the majority of plants safety injection flow cannot be throttled thus, to reduce SI flow one must stop an SI pump or realign its injection path. However, an SI pump cannot be stopped indiscriminately. If the break is large, reducing safety injection flow may result in decreasing RCS inventory and a loss of subcooling. Thus a conflict exists between maintaining adequate RCS inventory and maintaining pressurizer pressure control. The purpose of the SI reduction tables found in ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION, ECA-3.1 SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED, ECA-3.2 SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED and FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK (LP Version only) is to resolve this conflict. The SI reduction tables are constructed such that when the appropriate subcooling temperature is reached an SI pump can be stopped without losing pressurizer pressure control or RCS subcooling (except when in ECA-3.2).

The criteria (parameter) that is used to determine if and when a safety injection pump can be stopped is RCS subcooling. The subcooling criteria will depend on the number and type of safety injection pumps injecting into the RCS, whether the reactor coolant pumps are running and if the criteria for adverse containment conditions has been reached or not. For a High Pressure SI plant there are three means of injection considered when determining the subcooling criteria; charging/safety injection (CHG/SI) which injects through the BIT, high head safety injection (HHSI) which injects through the SI lines and injection through

During EOP E-1.2, "Post LOCA Cooldown and Depressurization," values for required subcooling vary greatly depending on running pump combinations.

What is the basis for these requirements?

- A. Ensures subcooling after an ECCS pump is shutdown, which decreases pressure and subcooling.
- B. Various inaccuracies involved with different combinations of pumps running.
- C. Allows for the reduced cooling effect when securing an ECCS pump.
- D. Takes into account the pump heat added by multiple ECCS pumps running.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7334	State the basis of SI termination criteria
3812	Explain the effect subcooling criteria has on ECCS pump operation
5430	Explain the conditions that affect SI Termination Criteria
5770	State the basis for required subcooling margin during accident conditions
6779	State the purpose of starting ECCS pumps as necessary to maintain RCS subcooling

Reference Id: P-33554
 Must appear: No
 Status: Active
 User Text: 5770.130494
 User Number 1: 0000003.00
 User Number 2: 0000003.30
 Difficulty: 1.00
 Time to complete: 2
 Topic: LPE1B - E-1.2, Subcooling requirements ECCS flow reduction
 Cross Reference: LPE-1B,OBJ5(5770) PG32

*** **UNCONTROLLED PROCEDURE - DO NOT USE TO PERFORM WORK or ISSUE FOR USE** ***

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP E-0.1
REVISION 29
PAGE 14 OF 25

TITLE: Reactor Trip Response

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

10. CHECK RCP 2 - RUNNING

Try to start RCP(s) to provide forced cooling and normal spray:

- a. IMPLEMENT APPENDIX B to start an RCP.
- b. Start RCP 2.

IF RCP 2 can **NOT** be started,
THEN Try to start other RCP(s) as necessary to provide forced cooling and normal spray.

IF No RCP can be started
THEN Verify Natural circulation based on:

- 1) RCS subcooling based on core exit T/Cs **GREATER THAN** 20°F.
- 2) S/G Pressures stable or decreasing.
- 3) RCS Hot Leg Temperatures stable or decreasing.
- 4) Core Exit T/Cs stable or decreasing.
- 5) RCS Cold Leg Temperatures at saturation temperature for S/G Pressure.

IF Natural Circulation **NOT** Verified,

THEN Increase Dumping Steam.

11. CHECK Secondary Systems Status:

- a. Check Mn Fdwtr Pps Tripped
 - o Mn Fdwtr Pps No. 1 and No. 2 Green Trip Light - ON
- b. Stop ALL but one Condensate and Booster Pump set

- a. **IF** **NOT** needed for control of feedwater,
THEN Trip the Mn Fdwtr Pps

THIS STEP CONTINUED ON NEXT PAGE

Question 4 Bank- INPO Question

015 AK1.01

5/12/2005

Beaver Valley Unit 1

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The Unit is operating at 100% power, NSA with the exception of [PT-447], Turbine First Stage Pressure Transmitter, which is in the process of being removed from service due to erratic operation. The condenser steam dumps are in the Tavg mode. A loss of reactor plant component cooling water then results in a manual reactor trip. Ten minutes after the trip, the following conditions exist during the performance of ES-0.1, Reactor Trip Response:

SG Pressure
1A 1000 psig and stable
1B 1005 psig and stable
1C 995 psig and stable

- " All RCP's are OFF.
- " RCS pressure is 2230 psig and stable.
- " Thot is approximately 575 degrees F in all three (3) loops and slowly lowering.
- " Core Exit TC's indicate approximately 580 F and stable.
- " Tcold is approximately 555 degrees F in all three (3) loops and stable.
- "Reactor Trip Breaker "A" failed to open.

Which ONE of the following describes the condition of the RCS and the preferred method of heat removal for the current plant conditions?

- A. Natural circulation exists. The condenser steam dumps are maintaining heat removal.
- B. Natural circulation does NOT exist. Heat removal will be established by opening the condenser steam dumps.
- C. Natural circulation exists. SG atmospheric steam dump valves are maintaining heat removal.
- D. Natural circulation does NOT exist. Heat removal will be established by opening the SG atmospheric steam dump valves.

B. Correct. Condenser steam dumps are the preferred method of heat removal in ES-0.2.

A. Incorrect. Natural circulation does not exist because Tcold temperatures are too high and stable.

C. Incorrect. Natural circulation does not exist because Tcold is too high for SG pressure, and not trending down. No indications that SG ADV's are open for heat removal.

D. Incorrect. Tcold is above saturation for each SG and not trending down indicating natural circulation does not exist. SG ADV's are not the preferred method of heat removal.

STEP DESCRIPTION TABLE FOR FR-S.1Step 4

STEP: Initiate Emergency Boration of RCS

PURPOSE: To add negative reactivity to bring the reactor core subcritical

BASIS:

After control rod trip and rod insertion functions, boration is the next most direct manner of adding negative reactivity to the core. The intended boration path here is the most direct one available, not requiring SI initiation, but using normal charging pump(s). Pump miniflow lines are assumed to be open to protect the pumps in the event of high RCS pressure.

Several plant specific means are usually available for rapid boration and should be specified here in order of preference. Methods of rapid boration include emergency boration, injecting the BIT, and safety injection actuation. It should be noted that SI actuation will trip the main feedwater pumps. If this is undesirable, the operator can manually align the system for safety injection. However, the RWST valves to the suction of the SI pumps should be opened first before opening up the BIT valves. If a safety injection is already in progress but is having no effect on nuclear flux, then the BIT and RWST are not performing their intended function, perhaps due to blockage or leakage. In this case some other alignment using the BATs and/or non-safeguards charging pump(s) is required.

The check on RCS pressure is intended to alert the operator to a condition which would reduce charging or SI pump injection into the RCS and, therefore, boration. The PRZR PORV pressure setpoint is chosen as that pressure at which flow into the RCS is insufficient. The contingent action is a rapid depressurization to a pressure which would allow increased injection flow. When primary pressure drops 200 psi below the PORV pressure setpoint, the PORVs should be closed. The operator must verify successful closure of the PORVs, closing the isolation valves, if necessary.

The following plant conditions exist:

The reactor has failed to trip both automatically and manually.
 Control rods are being inserted continuously.
 The turbine is tripped.
 The AFW pumps are running.
 Emergency boration is in progress.
 All PORVs are closed and their block valves open.
 RCS pressure is 2400 psig.
 The operating crew opens one PORV and reduces PZR pressure below 2135 psig.

Which one of the following describes why the crew opens the PORV?

- A. To allow sufficient borated injection flow into the RCS to ensure the addition of negative reactivity to the core.
- B. To prevent a challenge to the safety valves due to the rapid overpressurization transient associated with an ATWS.
- C. To minimize primary-to-secondary leakage in case of a SGTR, until other recovery actions can be taken.
- D. To ensure PTS limits will not be exceeded during the subsequent RCS cooldown.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7920	Explain basis of emergency procedure step
------	-------------------------------------------

Reference Id: P-0619
 Must appear: No
 Status: Active
 User Text: 7920.130494
 User Number 1: 0000003.00
 User Number 2: 0000003.40
 Difficulty: 3.00
 Time to complete: 2
 Topic: Why PORVs are opened in FR-S.1.
 Cross Reference: LPE-S

ACTION / EXPECTED RESPONSE**RESPONSE NOT OBTAINED****1. GO TO Applicable Steps of this Procedure for Existing Problem with CCW System:**

- HIGH CCW System Temp - steps 2, 3, 4, 5, 6 and 7
- LOW CCW System Flow - steps 7, 8, 9 and 10
- CCW System OUT-Leakage - steps 11, 12, 13 and 14
- CCW System IN-Leakage - steps 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26 and 27

2. PLACE a Second Train of ASW/CCW In Service per OP E-5:II Section 6.1^{T31387}**3. VERIFY ASW Flow to In Service CCW HX - ADEQUATE:**

- ASW Pp Amps - GREATER THAN 51 Amps

GO TO OP AP SD-3, LOSS OF AUXILIARY SALT WATER, unless this procedure was entered From OP AP SD-3 or it has been determined that ASW pumps are not capable of being placed in service, then implement Appendix E, Instructions for Loss of Ultimate Heat Sink and continue with step 4.

4. REDUCE CCW Heat Loads:^{T31387}

- SHUT DOWN the BA Evap
- STOP ALL RCPs
- ISOLATE Letdown
- SWAP ASDR to Other Unit

Pressurizer Level Channel Failures

Controlling Channel Failure	Indications	Alarms	Plant Response
Controlling PZR level fails HI	PZR level meter goes Hi PZR level recorder (if selected) goes High	PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06, PZR LEVEL HI/LO PK05-21 PZR LEVEL HI/LO CONTROL PK05-01 RCP Seal Alarms PK05-01	Backup heaters will initially energize (+5% deviation) Charging flow will decrease, causing actual level to drop, which eventually will cause letdown to isolate at 17%, and heaters to turn off. After letdown is isolated, actual Pressurizer level will start to increase and eventually cause a high level reactor trip, if the PDP is in service. If the CCP is in service FCV-128 can fully close stopping the increase in PZR level. This response of FCV-128 would also cut off flow to the RCP seals.
Controlling PZR level fails Low (or loses power)	PZR level meter goes Low PZR level recorder (if selected) goes Low	PZR LEVEL HI/LO PK05-21 PZR LEVEL HI/LO CONTROL PK05-22	Charging flow will increase. Letdown will isolate and the heaters will turn off. Actual level will increase, which will eventually cause a high level reactor trip.
Backup PZR level fails low	PZR level recorder (if selected) goes Low	PZR LEVEL HI/LO CONTROL PK05-22	Letdown will isolate and the heaters will turn off
Backup or Non-selected PZR level fails high	PZR level recorder (if selected) goes High	PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06	None

Note: No effects other than indication if LT-462 fails or loses power.

Question 8 Bank- INPO Question

..029.EK2.06

12/1/2002

Beaver Valley Unit 1

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Given the following conditions:

- "Reactor Trip testing is in progress on Train "A".
- "Reactor Trip Breaker "A" is open.
- "Reactor Trip Bypass Breaker "A" is closed.
- "A transient occurs requiring a reactor trip.
- "The RO attempts to manually trip the reactor but the reactor does NOT trip.

Which one of the following describes a failure that has contributed to the reactor trip failure?

Reactor Trip Bypass Breaker "A" Undervoltage Trip coil failed to deenergize.

Reactor Trip Breaker "B" Undervoltage Trip coil failed to energize.

Reactor Trip Breaker "B" Undervoltage Trip coil failed to energize.

Reactor Trip Bypass Breaker "B" Shunt Trip coil failed to energize.

Correct. RTB "B" trip relays act the same as RTB trip coils.

Incorrect. RTB "B" trip coils are normally energized. Deenergizes on trip signal.

Incorrect. Shunt trip energizes to trip the RTB.

Incorrect. RTB "B" not equipped with a shunt trip.

1.0 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

2.0 RESTART SAFEGUARDS EQUIPMENT AFTER LOSS OF OFFSITE POWER

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

3.0 SI REINITIATION CRITERIA

NOTE: THIS FOLDOUT PAGE ITEM IS NOT APPLICABLE UNTIL EOP E-3, STEP 25.
IF EITHER condition listed below occurs:
• RCS subcooling based on Core Exit T/Cs – LESS THAN 20°F (Subcooled Margin Monitor, YI-31 or Appendix C)
• PZR level – CANNOT BE MAINTAINED GREATER THAN 12% [36%]
THEN Manually start ECCS pumps as necessary to restore desired condition
AND
GO TO EOP ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT, SUBCOOLED RECOVERY DESIRED

4.0 MULTIPLE TUBE RUPTURE CRITERIA

IF Any Intact S/G Level increases in an Uncontrolled manner,
OR
Any Intact S/G has Abnormal Radiation
THEN Stabilize the plant and RETURN TO EOP E-3, STEAM GENERATOR TUBE RUPTURE, Step 1

5.0 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 cooldown,
THEN GO TO EOP E-2, FAULTED STEAM GENERATOR ISOLATION.

6.0 AFW SUPPLY SWITCHOVER CRITERION

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

Question 9 Reference

ON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK05-07
Rev. 11
Page 1 of 3

**SUBCOOLING
MARGIN
LO/LO-LO**

06/14/05

Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
1151	Subcooled Mon Lo-Lo Margin From RVLIS Train A	K4 (from SCMM A)	< 20°F	2.1
1152	Subcooled Mon Lo Margin From RVLIS A or B	K3 (from SCMM A)	< 30°F <u>AND</u> below P-10	2.1
		K3 (from SCMM B)	< 30°F <u>AND</u> below P-10	2.1
1600	Subcooled Mon Lo-Lo Margin From RVLIS Train B	K4 (from SCMM B)	< 20°F	2.1

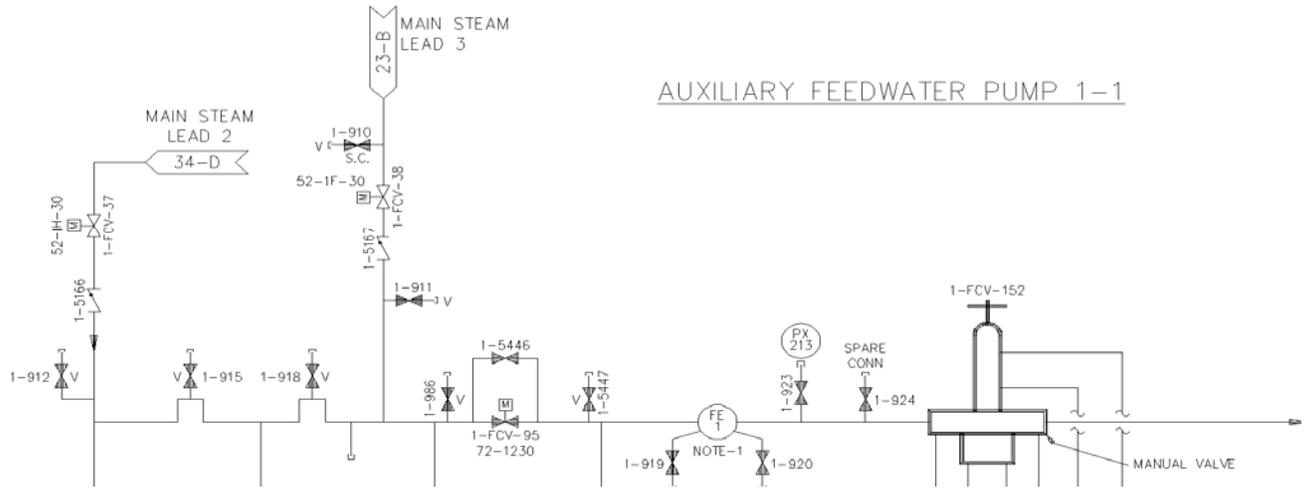
2. OPERATOR ACTIONS

NOTE: Three of four power range channels must be below P-10 to satisfy the alarm condition for Input 1152.

2.1 General Actions (All Inputs)

- 2.1.1 Observe RCS pressures and temperatures to confirm low subcooling margin. []
- 2.1.2 IF RCS is approaching a saturated condition, THEN perform at least one of the following, using appropriate means, to restore RCS subcooling to normal: []
 - Raise RCS pressure
 - Lower RCS temperature

OVID 106704 sheet 4



Design Information, Continued

Importance to safety
Obj 4

The MFW system provides a number of safety-related functions as described below.

- The safety-related portion of MFW system consists of the four MFW Isolation Valves, their associated check valves, and the piping downstream to the S/Gs. In addition, four Design Class I Control Valves and Bypass Valves are located upstream in non-safety-related piping.
- The non safety related portion that contains the control and bypass valves has been seismically analyzed and supported to assure the integrity and operability of the Control and Bypass Valves.

Function	The MFW system must be able to ...
Feedwater Isolation	stop feed flow in the event of a feedline or steam line rupture, to ensure that the minimum amount of high energy fluid is dumped into the Containment or to the environment. <ul style="list-style-type: none"> • The control valves and bypass valves, backed up by the main feedwater isolation valves (MFIVs), are all closed.
AFW flowpath	provide a <i>flow path for AFW</i> system from the MFW check valves to the S/Gs, to minimize the number of lines running into containment.

Question 12 Reference

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP ECA-0.0
REVISION 20
PAGE 14 OF 26

TITLE: Loss of All Vital AC Power

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

<p>16. <u>CHECK CST Level - GREATER THAN 10%</u></p>	<p>IMPLEMENT OP D-1:V, ALTERNATE AUXILIARY FEEDWATER SUPPLIES</p>
<p>a. Continue to monitor CST Level - GREATER THAN 10%</p>	<p>-----</p>

<p>17. <u>DEPRESSURIZE Intact Steam Generators To Reduce RCS Pressure To Inject Accumulators:</u></p>	
<p>*****</p>	
<p><u>CAUTION 1:</u> Accumulator Nitrogen injection into the RCS may occur if S/Gs are Depressurized to LESS THAN 140 PSIG.</p>	
<p><u>CAUTION 2:</u> S/G NR Level should be maintained GREATER THAN 6% [16%] in AT LEAST ONE intact S/G. <u>IF</u> S/G Level CAN NOT be maintained, <u>THEN</u> S/G depressurization should be stopped until S/G Level is restored in AT LEAST ONE intact S/G.</p>	
<p>*****</p>	
<p><u>NOTE:</u> The S/Gs are being depressurized at maximum rate to minimize RCS inventory loss due to anticipated RCP Seal failures.</p>	
<p>a. Check S/G NR Levels - GREATER THAN 6% [16%] IN AT LEAST ONE INTACT S/G</p>	<p>a. Perform the following:</p> <ol style="list-style-type: none"> 1) Maintain Maximum AFW flow until S/G NR Level GREATER THAN 6% [16%] in at least one intact S/G. 2) <u>WHEN</u> S/G NR Level is GREATER THAN 6% [16%] in at least one intact S/G, <u>THEN</u> Perform Steps 17b (Next Page), 17c, 17d and 17e. 3) Continue with Step 18 (Page 16) <p>-----</p>

THIS STEP CONTINUED ON NEXT PAGE

Question 12 Reference

PACIFIC GAS AND ELECTRIC COMPANY
 DIABLO CANYON POWER PLANT

NUMBER EOP ECA-0.0
 REVISION 20
 PAGE 15 OF 26

TITLE: Loss of All Vital AC Power

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

17. DEPRESSURIZE Intact Steam Generators To Reduce RCS Pressure To Inject Accumulators:

(Continued)

NOTE 1: PZR Level may be lost and Reactor Vessel Upper Head VOIDING may occur due to depressurization of S/Gs. Depressurization SHOULD NOT be stopped to prevent these occurrences.

NOTE 2: DO NOT start or continue S/G depressurization per step 17b unless at least one intact S/G Level is GREATER THAN 6% [16%].

b. Manually Dump Steam at MAXIMUM rate using S/G 10% steam dumps

b. Locally dump steam using 10% steam dumps.

c. Check RCS Cold Leg temperatures - GREATER THAN 270°F

c. Perform the following:
 1) Control S/G 10% Steam Dumps to Stop S/G Depressurization.
 2) GO TO Step 18 (Next Page).

d. Check S/G Pressures - LESS THAN 240 PSIG

d. Perform the following:
 1) WHEN S/G Pressures decrease to LESS THAN 240 PSIG,
THEN Perform Step 17e.
 2) GO TO Step 18 (Next Page)

e. Control S/G Pressure at 240 PSIG using 10% Steam Dumps

e. Locally control S/G 10% steam dumps to maintain S/G Pressures at 240 PSIG.

STEP DESCRIPTION TABLE FOR ECA-0.0Step 16

During SG depressurization, AFW flow may have to be increased to maintain the required SG narrow range level. Control of AFW flow will have to be performed from the control room or locally depending on plant specific design. Full AFW flow should be established to any SG in which level drops out of the narrow range.

RCS cold leg temperatures should be monitored during SG depressurization to ensure that the depressurization does not impose a challenge to the Integrity Critical Safety Function. This check is included in Step 16c since guideline ECA-0.0 has priority over the Function Restoration Guidelines and the operator is instructed to not implement a Function Restoration Guideline even if a Critical Safety Function challenge is detected by the Critical Safety Function Status Trees. Consequently, Step 16c implicitly protects the Integrity Critical Safety Function. The SG depressurization should not result in a challenge to the Integrity Critical Safety Function since the resultant RCS cold leg temperatures should not approach the temperature limit (i.e., T2 temperature) at which a challenge will exist.

Once the target SG pressure is reached, the SG PORVs and AFW flow should be controlled to maintain SG pressure at the target value until ac power is restored.

The target SG pressure for Step 16 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS. The target SG pressure should be based on the nominal SG pressure to preclude nitrogen addition, plus margin for controllability (e.g., 100 psi). To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tanks pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume.

The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1 V_1^\gamma = P_2 V_2^\gamma$$

where $\gamma = 1.25$ for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta p from P_2 and adding the margin to controllability. The RCS to SG delta p should be calculated as described in the RCP TRIP/RESTART section in the Generic Issues of the Executive Volume. Instrument uncertainties are not included in the determination of the steam generator pressure limit to preclude a bias toward either having more accumulator water injected into the RCS or having less nitrogen injected into the

Hagan Controller Response to Loss of Power

U1&2 Attachment 1: Page 1 of 1

NOTE 1: Manual power supplies power to the Hagan controller lights. Therefore, if manual power is lost, the lights on the controller will go out and the controller will go to the AUTO-HOLD setpoint, indicated by the controller demand meter indicating a value above zero. The controller will remain in AUTO-HOLD until manual power is restored. Once manual power is restored, the Hagan controller will transfer to Manual in 5-30 seconds. During the transfer, the controller Auto light will be lit with the controller in AUTO-HOLD. It should be noted that a "bump" could occur as the auto amplifier attempts to return the parameter to the setpoint if the controlled variable deviates from the Auto setpoint by a large amount.

NOTE 2: When both the Manual and Auto power are lost, the controller output will go to zero. One way to quickly recognize this is that the demand meter reads zero and the lights on the controller are out.

The following is a summary of Hagan controller response to a temporary loss of power (with the exception of LCV-8 and 12):

INITIAL CONDITION	TYPE OF POWER FAILURE	RESULTS	
		*INTERIM	FINAL
Auto	Loss of Auto Power momentarily	Manual	Manual
Auto	Loss of Manual Power momentarily	Auto Hold	Manual
Manual	Loss of Manual Power momentarily	Auto Hold	Manual
Manual	Loss of Auto Power momentarily	Manual	Manual

* The Hagan controller will be in this condition until power is restored.

7970

Identify failure modes associated with Hagan controllers

P-73313	Points:	1.00	Multiple Choice
---------	---------	------	-----------------

The Pressurizer Pressure controller is operating in AUTOMATIC.

A loss of PY-11 occurs which causes a loss of MANUAL power to the controller.

How are the indications and operation of the controller affected?

- A. Lights on the controller go out; the controller fails to Auto Hold.
- B. Lights on the controller remain lit; the controller still controls in automatic.
- C. Lights on the controller go out; the controller fails to Manual.
- D. Lights on the controller remain lit; the controller fails to Auto Hold.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7970	Identify failure modes associated with Hagan controllers
------	----------------------------------------------------------

Reference Id: P-73313
Must appear: No
Status: Active
User Text: 4303.030853
User Number 1:
User Number 2:
Difficulty: 3.00
Time to complete: 2
Topic: LPA-4, loss of manual power to pZR press controller
Cross Reference: LPA4,OBJ3(4303) PG 6&7
Comment: loss of manual power to a Hagan controller causes all lights to go out. in addition, the controller shifts to auto-hold. when power is restored, the lights will re-energize and the controller will control in manual.

reference: OP O-2, attachment 9.1

STG E5

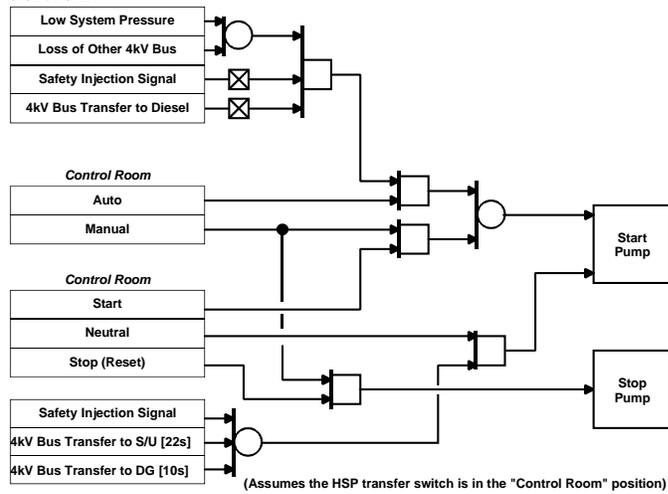
Power supplies
Obj 4

The power supplies for the pump motors are:

ASW Pump	4 kV vital bus
1-1	F
1-2	G
2-1	F
2-2	G

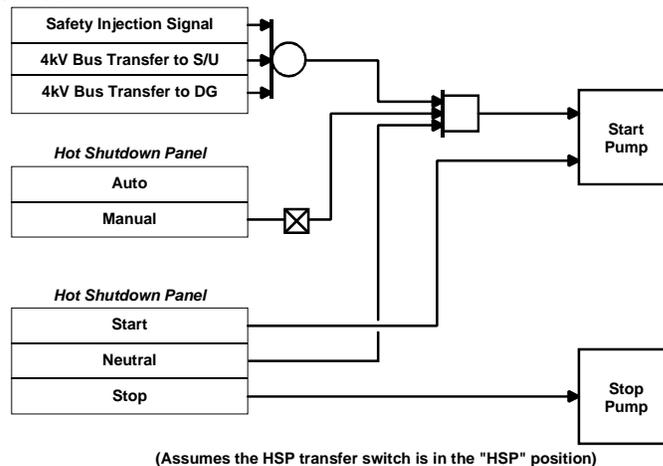
Logic diagrams for pump starts and stops from the control schemes
Obj 9

The following logic diagram assumes the transfer switch at the breaker in the CONT RM position.



ASW-05

The following logic diagram assumes the transfer switch at the breaker in the HSD PNL position.



(Assumes the HSP transfer switch is in the "HSP" position)

ASW-06

Trips
Obj 9, 27

The ASW pumps have these automatic trips:

- overcurrent
- 4 kV bus undervoltage (load shed)

STG E5

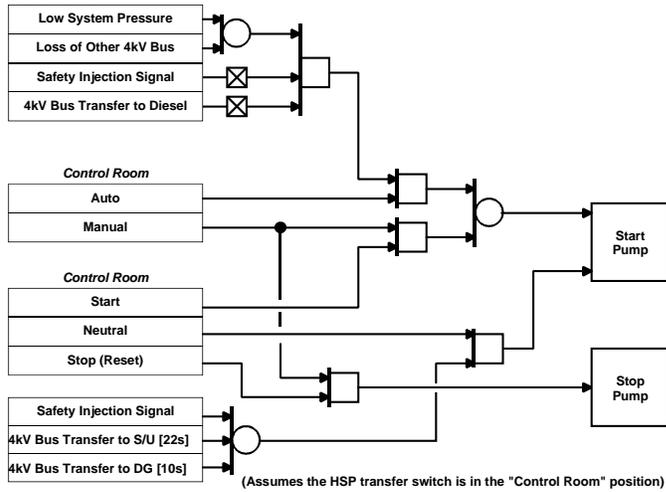
Power supplies
Obj 4

The power supplies for the pump motors are:

ASW Pump	4 kV vital bus
1-1	F
1-2	G
2-1	F
2-2	G

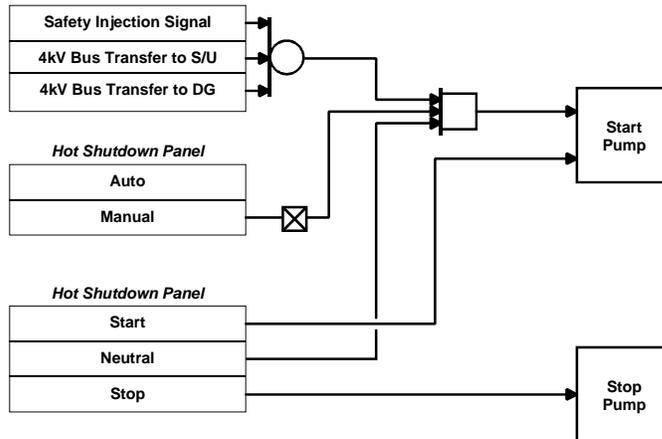
Logic diagrams for pump starts and stops from the control schemes
Obj 9

The following logic diagram assumes the transfer switch at the breaker in the CONT RM position.



ASW-05

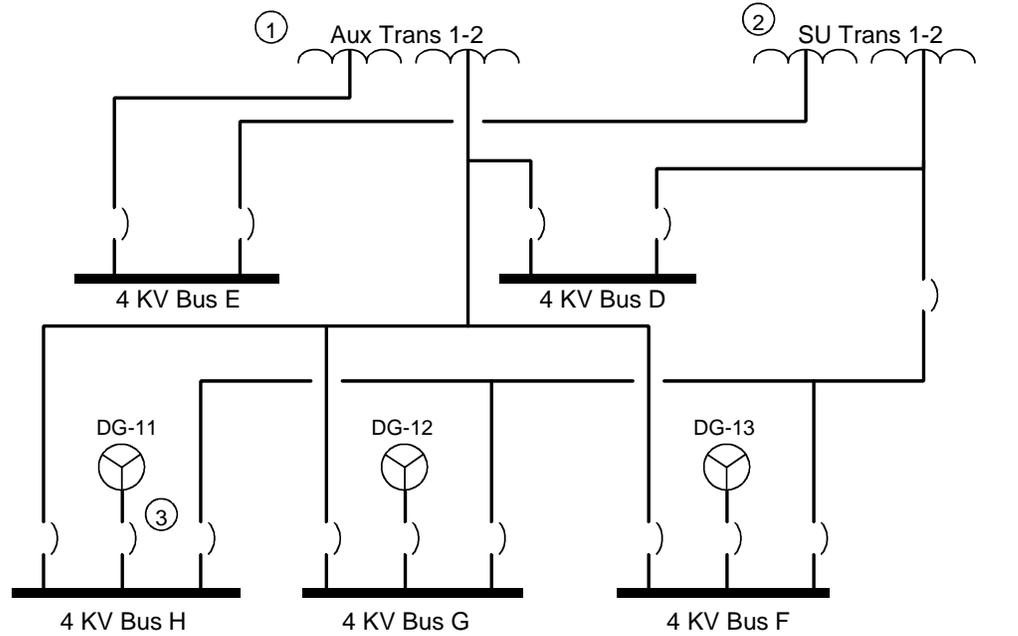
The following logic diagram assumes the transfer switch at the breaker in the HSD PNL position.



ASW-06

J6A
Description
4 kV system

The 4 kV system is shown here.

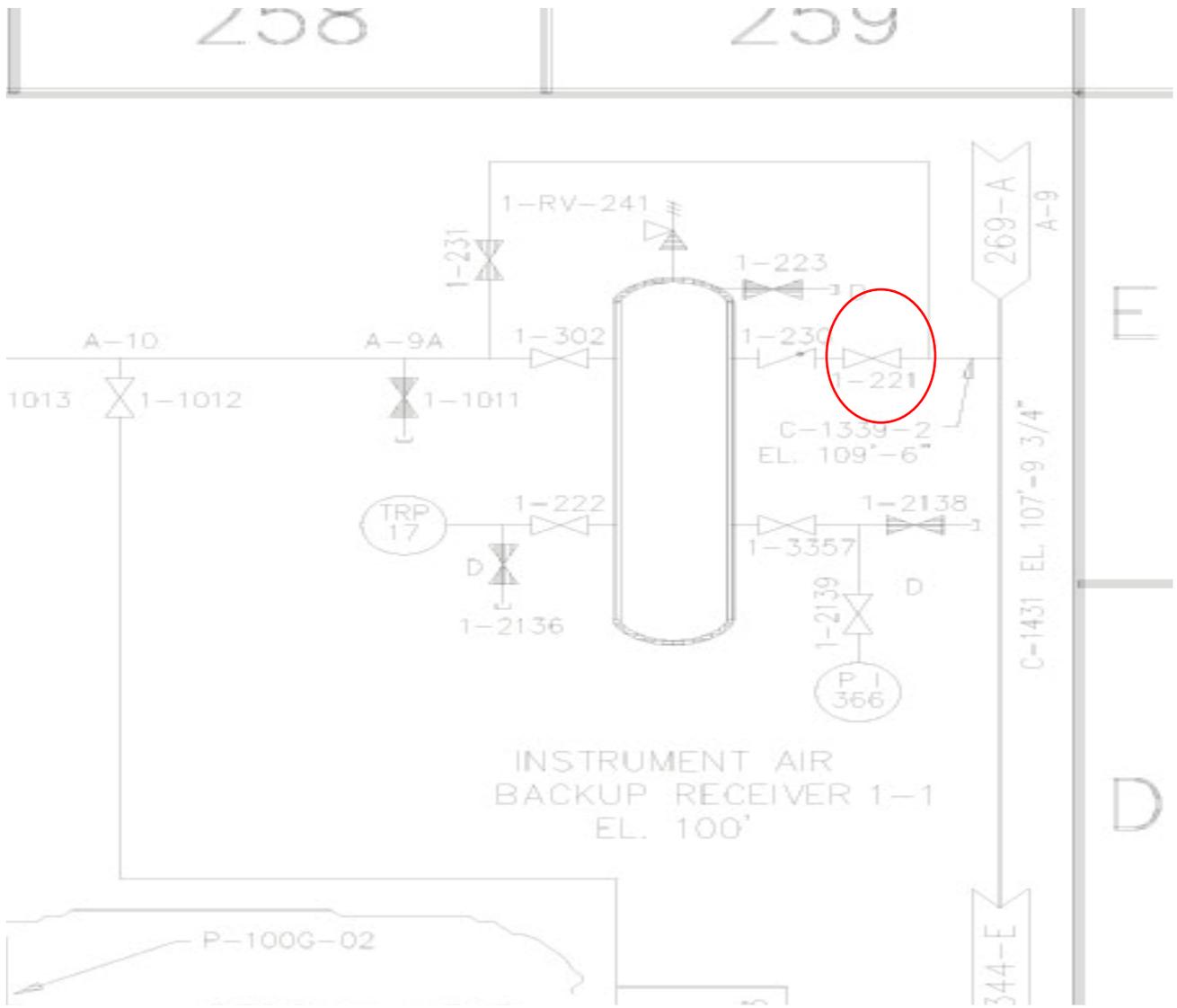


Each unit has five 4 kV buses divided into two groups designated as:

- Nonvital
 - Buses D and E.
- Vital
 - Buses F, G, and H.

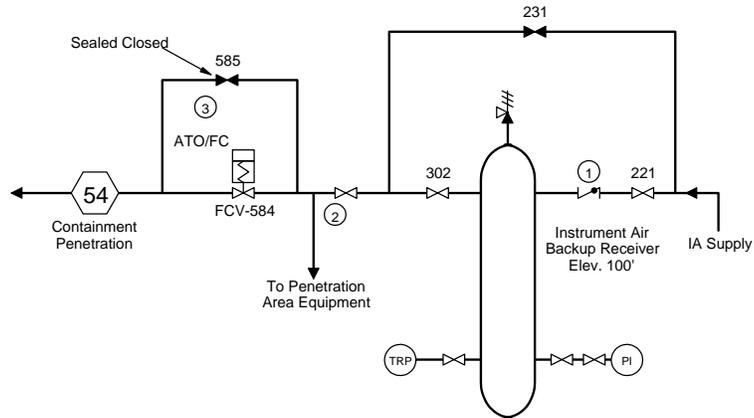
The nonvital group supplies power to the nonsafety-related and secondary plant 4 kV loads.

The vital group supplies power to the reactor protection and engineered safeguard loads.



Air Valves for Containment IA Isolation, FCV-584, Continued

Description The IA Supply system for the Containment is described below.



CAS-27

Stage	Description
1	IA for the Containment and the penetration area equipment is supply through a backup receiver. <ul style="list-style-type: none"> • A check valve prevents backflow to the IA header if IA pressure is lost. • A bypass valve (231) is provided to allow isolation and maintenance of the receiver.
2	Header branches into 2 lines downstream of the receiver: <ul style="list-style-type: none"> • One header provides air to loads in the penetration area. • One header goes through isolation FCV-584 to the containment.
3	FCV-584 is the automatic Containment Isolation valve for the IA system. <ul style="list-style-type: none"> • Bypass valve provided to allow maintenance and testing of FCV-584.

Controls
Obj 5, 9, 12

The following controls are available for FCV-584.

In the Control Room on VB-4

Control	Operation
CLOSE/OPEN	3 position, spring return to neutral.

Continued on next page

STEP DESCRIPTION TABLE FOR ECA-1.2Step 3STEP: Check If Break Is IsolatedPURPOSE: To determine if the LOCA outside containment has been isolated from previous actionsBASIS:

This step instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions. If the break is isolated in Step 2, a significant RCS pressure increase will occur due to the SI flow filling up the RCS with break flow stopped.

The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump.

ACTIONS:

- o Determine if RCS pressure is increasing
- o Transfer to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1
- o Transfer to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1

INSTRUMENTATION:

RCS pressure indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

It should be noted that for some breaks SI flow may cause an RCS pressure increase independent of break isolation. It should also be noted that for larger breaks, RCS repressurization may be delayed following break isolation. Additionally, if the RCS is saturated or a cooldown is in progress, RCS repressurization will proceed more slowly. Other means of verifying break isolation should be checked. For example, increasing RVLIS trend due to injection flow, decreasing trends in local abnormal conditions and local observation (if practical) may be useful.

Step description:

STEP DESCRIPTION TABLE FOR FR-H.1 Step 15

STEP: Establish RCS Bleed Path

PURPOSE: To open all PRZR PORVs to establish an RCS bleed path

BASIS:

The operator ensures that the pressurizer block valves are open and opens both pressurizer PORVs to establish an RCS bleed path. These valves must be maintained in the open position until secondary heat sink is restored.

Once the pressurizer PORVs are open, the RCS will depressurize and the charging/SI pumps and/or high-head SI pumps will deliver subcooled flow to the RCS. This will provide adequate RCS heat removal until flow can be established to the steam generators to restore secondary heat sink.

ACTIONS:

- o Determine if power is available to PRZR PORV block valves
- o Determine if PRZR PORV block valves are open
- o Restore power to block valves
- o Open block valves
- o Open all PRZR PORVs

INSTRUMENTATION:

- o PRZR PORV and block valve position indication
- o Indication of power available to PRZR PORV block valves

CONTROL/EQUIPMENT:

- o PRZR PORV and block valve switches
- o Block valve power supply controls

*** **UNCONTROLLED PROCEDURE – DO NOT USE TO PERFORM WORK or ISSUE FOR USE** ***

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP ECA-1.1
REVISION 18
PAGE 7 OF 31

TITLE: Loss of Emergency Coolant Recirculation

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

11. Check if ECCS Is In Service

GO TO Step 22 (Page 11)

- SI Pps - ANY RUNNING

OR

- Charging Injection - NOT
ISOLATED

OR

- RHR Pps - ANY RUNNING

NOTE: The CCP and SI Pps should be stopped in alternate trains when possible.

12. ESTABLISH One Train Of SI Flow:

- a. Depress Vital 4KV Auto Transfer
Relay Resets: Blue Light - OFF

- b. Verify CCPs - ONLY ONE
RUNNING

- c. Verify SI Pps - ONLY ONE
RUNNING

- d. RCS Pressure - LESS THAN
300 PSIG

- d. Stop Both RHR Pps

AND

GO TO Step 13 (Next Page).

- e. Verify RHR Pps - ONLY ONE
RUNNING

STG J15, page 2.1-3

Bus Auto Xfer to S/U Cutout Switch

Purpose
Obj 4, 5

The purpose of the Bus Auto Xfer to S/U Cutout Switch is to allow the operators to block the actuation of the transfer to S/U relay. The reasons this may be done are:

- Allow for the controlled restoration from a loss of power.
- Allow for the disabling of one 12kv bus auto transfer during degraded 230 kv grid conditions.

Location

All of the Bus Auto Xfer to S/U Cutout Switches are located in the Control Room on VB4/VB5 next to the bus that they control.

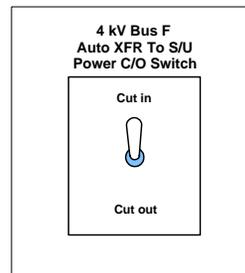
Description

A Bus Auto Xfer to S/U Cutout Switch is available for the following buses.

- 12 kV Bus D
- 12 kV Bus E
- 4 kV Bus D
- 4 kV Bus E
- 4 kV Bus F
- 4 kV Bus G
- 4 kV Bus H

Controls

A Bus Auto Xfer to S/U Cutout Switch shown below is available in the Control Room.



EPT-02

Control	Operation
CUT IN/CUT OUT	2 position maintained toggle

Pages 2.1-5 and 2.1-6

Bus Auto Xfer Reset Pushbutton

Purpose
Obj 4, 5

The purpose of the Bus Auto Xfer Reset Pushbutton is to reset the associated Auto Transfer Relays and remove the associated control functions.
Note: Refer to Section 2.2 for a description of the functions associated with each of the relays.

Location

All of the Bus Auto Xfer Reset Pushbuttons are located in the Control Room on VB4/VB5 next to the bus that they control.

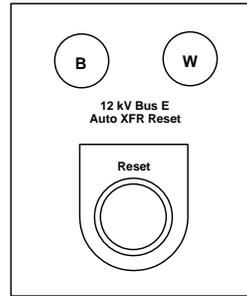
Description

A Bus Auto Xfer Reset Pushbutton is available for the following buses.

- 12 kV Bus D
- 12 kV Bus E
- 4 kV Bus D
- 4 kV Bus E
- 4 kV Bus F
- 4 kV Bus G
- 4 kV Bus H

Controls

A Bus Auto Xfer Reset Pushbutton shown below is available in the Control Room.



EPT-03

Control	Operation
RESET	pushbutton

Continued on next page

Bus Auto Xfer Reset Pushbutton, Continued

Logic
Obj 3

The logic associated with the Bus Auto Xfer Reset Pushbutton is described in the table below.

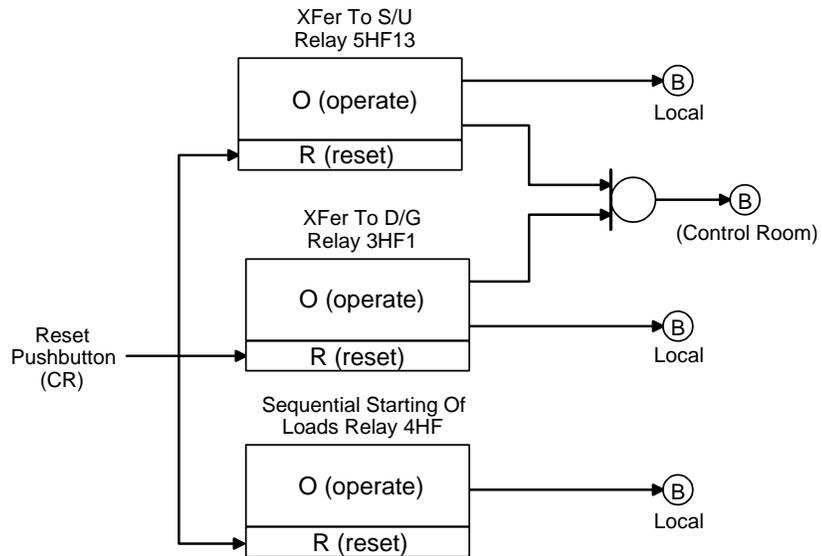
If the pushbutton is ...	And ...	Then the following relays and associated actuating signals will be reset
Depressed	The initiating logic to the relay has been cleared.	<ul style="list-style-type: none"> • 12 kV fast auto Xfer • 12 kV slow auto Xfer • 4 kV Xfer to S/U • 4 kV Xfer to D/G (vital only) • 4 kV sequential starting of loads (vital only)

Note: The initiating logic is automatically cleared when the S/U feeder or D/G output breaker supplying the bus is closed.

Note: Refer to Section 2.2 for a description of the initiating logic and associated functions for each of the relays.

Logic diagram
Obj 3

The logic diagram for the Bus Auto Xfer Reset Pushbutton is shown below.



Bus Auto Xfer Reset Pushbutton, Continued

Logic
Obj 3

The logic associated with the 4 kV Vital Bus Auto Transfer Trip Relay Reset Device is described in the table below:

If the auto transfer to S/U lock-out relay reset switch is taken to	And ...	Then...
RESET	The initiating logic to the relay has been cleared.	The lockout relay will be returned to its RESET position.
	The relay still has its initiating logic made-up.	The lockout relay will spring return to the TRIP position.

Note: The reset device is procedurally never manually taken to the TRIP position. Doing so could cause damage to the lock-out relay.

Note: The initiating logic is automatically cleared when the S/U feeder or D/G output breaker supplying the bus is closed.

Note: Refer to Section 2.2 for a description of the initiating logic and associated functions for each of the relays.

AP-12A

1. SCOPE

- 1.1 This procedure provides instructions for unwarranted continuous withdrawal or insertion of a control rod bank while at power or during startup.

2. SYMPTOMS

This condition may be indicated by one or more of the following:

- 2.1 Unwarranted rod motion is indicated by DRPI and rod step counters.
- 2.2 Changing TAVG Indication with no change in TREF.
- 2.3 Increasing source/intermediate range flux level and/or startup rate during reactor startup (continuous rod withdrawal).
- 2.4 Possible Main Annunciator Alarms:
 - 2.4.1 For continuous rod withdrawal.
 - a. ROD BANK D STOP C-11 (PK03-15).
 - b. **TAVG DEVIATION FROM REF (PK04-03).**
 - c. OT DELTA-T C-3 CHANNEL ACTIVATED (PK04-04).
 - d. OP DELTA-T C-4 CHANNEL ACTIVATED (PK04-05).
 - e. AUCTIONED TAVG HIGH (PK04-10).
 - f. **PZR PRESSURE HIGH (PK05-16).**
 - g. **PZR LEVEL HI LO CONTROL (PK05-22).**
 - h. OT DELTA-T ROD STOP & TURBINE RUNBACK C-3 (PK08-09).
 - i. OP DELTA-T ROD STOP & TURBINE RUNBACK C-4 (PK08-10).

#

1.00

Given the following:

Unit 1 is at 100% power
All control systems are in their normal alignment
Pressurizer Level Control is selected to 459/461

Which of the following Pressurizer level failures would result in the fastest increase in reactor coolant inventory?

- A. LT-459 fails LOW.
- B. LT-461 fails HIGH.
- C. LT-461 fails LOW.
- D. LT-459 fails HIGH.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

36926	Explain the effects of Pressurizer system failures, including: ? Transmitters ? Valves <ul style="list-style-type: none">o PORVs and Safeties (including leakage or water relief)o Spray Valves (Normal or Aux)
4621	Explain the effects of Pressurizer Level Transmitter failures

Reference Id: P-1509
Must appear: No
Status: Active
User Text: 4621.020045
User Number 1: 0000003.10
User Number 2: 0000003.90
Difficulty: 4.00

Pressurizer Level Channel Failures

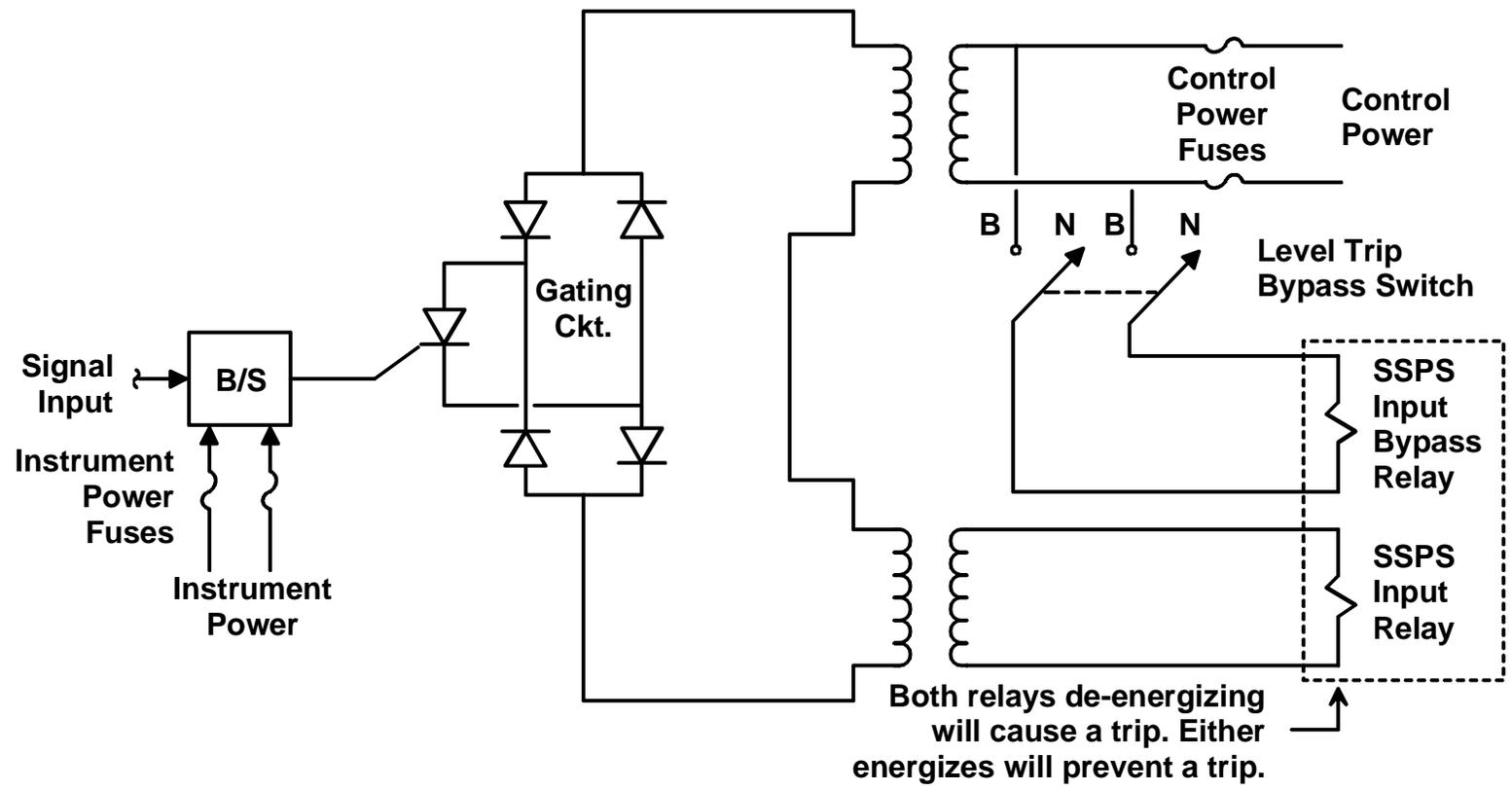
Controlling Channel Failure	Indications	Alarms	Plant Response
Controlling PZR level fails HI	<p>PZR level meter goes Hi</p> <p>PZR level recorder (if selected) goes High</p>	<p>PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06,</p> <p>PZR LEVEL HI/LO PK05-21</p> <p>PZR LEVEL HI/LO CONTROL PK05-01</p> <p>RCP Seal Alarms PK05-01</p>	<p>Backup heaters will initially energize (+5% deviation)</p> <p>Charging flow will decrease, causing actual level to drop, which eventually will cause letdown to isolate at 17%, and heaters to turn off.</p> <p>After letdown is isolated, actual Pressurizer level will start to increase and eventually cause a high level reactor trip, if the PDP is in service. If the CCP is in service FCV-128 can fully close stopping the increase in PZR level. This response of FCV-128 would also cut off flow to the RCP seals.</p>
Controlling PZR level fails Low (or loses power)	<p>PZR level meter goes Low</p> <p>PZR level recorder (if selected) goes Low</p>	<p>PZR LEVEL HI/LO PK05-21</p> <p>PZR LEVEL HI/LO CONTROL PK05-22</p>	<p>Charging flow will increase.</p> <p>Letdown will isolate and the heaters will turn off.</p> <p>Actual level will increase, which will eventually cause a high level reactor trip.</p>
Backup PZR level fails low	PZR level recorder (if selected) goes Low	PZR LEVEL HI/LO CONTROL PK05-22	Letdown will isolate and the heaters will turn off
Backup or Non-selected PZR level fails high	PZR level recorder (if selected) goes High	PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06	None

Note: No effects other than indication if LT-462 fails or loses power.

TS 3.3.1

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. Two Intermediate Range Neutron Flux channels inoperable.</p>	<p>G.1 -----NOTE----- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. ----- Suspend operations involving positive reactivity additions. ----- <u>AND</u> G.2 Reduce THERMAL POWER to < P-6.</p>	<p>Immediately</p> <p>2 hours</p>
<p>H. Not used</p>		
<p>I. One Source Range Neutron Flux channel inoperable.</p>	<p>I.1 -----NOTE----- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. ----- Suspend operations involving positive reactivity additions.</p>	<p>Immediately</p>
<p>J. Two Source Range Neutron Flux channels inoperable.</p>	<p>J.1 Open reactor trip breakers (RTBs).</p>	<p>Immediately</p>

Exc core Nuclear Instrumentation



Loss of power effects on source and intermediate range reactor trip signals are dependant upon the position of "Level Trip Bypass Switch." as follows:

	Level Trip Bypass Switch Position	
	Normal (N)	Bypass (B)
Loss of instrument power	Trip	Trip
Loss of control power	Trip	Trip

When the trip signals are blocked are blocked then no trip will occur on a loss of either power supply.

B-4-2

Rev 22

Question 22 Bank- INPO Question

..036.AA2.03

10/29/2001

Braidwood Unit 1

R

Mark
Question



Print
Recor

New
Sear

Limiting the magnitude of a potential release during a fuel handling accident is accomplished by limiting the maximum load traveling over the fuel assemblies in the Spent fuel Pool to LESS THAN OR EQUAL TO _____.

2000 lbs.

1000 lbs.

1500 lbs.

2500 lbs.

42.0 FUEL HANDLING SYSTEM

42.4 Refueling Operations - Crane Travel - Fuel Handling Building

ECG 42.4 No load > 2500 pounds shall travel over fuel assemblies in the spent fuel pool.

-----NOTE-----
The movable fuel handling building walls may travel over fuel assemblies in the spent fuel pool.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTIONS

-----NOTE-----
Prior to exceeding the Completion Time of any Required Action, a 10 CFR 50.59 evaluation must be approved by the PSRC justifying the acceptability of exceeding the Completion Time.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Load > 2500 pounds traveling over fuel assemblies in spent fuel pool.	A.1 Place crane load in a safe condition.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 42.4.1 Verify loads ≤ 2500 pounds.	Prior to movement over fuel assemblies in the spent fuel pool.

#

1.00

A HIGH RADIATION alarm is received on RE-18, the liquid radwaste rad monitor. Which of the following automatic valve action(s) occur?

- A. RCV-18 (liquid radwaste overboard) closes and FCV-477 (recirc to the EDR) opens.
- B. RCV-18 (liquid radwaste overboard) closes and FCV-477 (recirc to the EDR) closes.
- C. RCV-18 (liquid radwaste overboard) remains open, alarm is used as a warning only.
- D. RCV-18 (liquid radwaste overboard) remains open and FCV-477 (recirc to the EDR) opens.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8436	Analyze Liquid Rad Waste system control logic ? Containment Liquid Radwaste Isolation ? LRW Discharge High Radiation
69251	Explain the automatic actions associated with the Liquid Radwaste system.

Reference Id: P-0179
Must appear: No
Status: Active
User Text: 8442.130494
User Number 1: 0000002.90
User Number 2: 0000003.40
Difficulty: 0.00
Time to complete: 2

Radiation Monitors

Number	Use	Detec for Type	Location	Auto Actions	C/R Ind.
0-RM-1	Control Room	GM	By Rad Mon Rack	None	Y
RM-2 *	Containment	GM	By Pers. Hatch	None	Y
0-RM-3	Oily Wtr Sep	Scin.	85' OWS Room	None	Y
RM-4	Recip Charging	GM	73' Pump Room	None	Y
RM-6	NSSS Sample	GM	100' Aux Bldg	None	Y
RM-7	Incore Seal Tbl	GM	119' Containment	None	Y
0-RM-10	Aux Cont Bd	GM	85' Aux Bldg	None	Y
RM-11	Cont Air Part	Scin.	100' GE	None	Y
RM-12	Cont Rad Gas	GM	100' GE	None	Y
RM-13	RHR Exh Duct	Scin.	113' Aux Bldg Hall	None	Y
RM-14, 14R, 87	Pint Vnt Rad (Gas)	Beta Scint	85' FHB White Room	RM-14 auto starts RM-87 when RM-14 reaches high and of range	Y
RM-15, 15R	Air Ejectors	Scin.	104' Turb Bldg	None	Y
RM-17A/B	CCW	Scin.	73' Aux Bldg	Shuts Vent (RCV-16)	Y
0-RM-18	Liquid Radwaste	Scin.	55' Pipe Tunnel	Shuts OVBD (RCV-18) Opens EDR (FCV-477)	Y
RM-19	S/G B/D Sample	Scin	100' GE (Dual Setpoint)	Shuts B/D OC Valves Shifts FCV-498/499 to EDR vice OVBD	Y
RM-22	Waste Gas Vent	GM	55' Pipe Tunnel	Shuts OVBD (RCV-17)	Y
RM-23 *	S/G B/D Effluent	Scin.	100' GW	Same Actions as RM-19 above	N
RM-24/24R	Plant Vent Iodine	Scin.	85' FHB White Room	None	N
RM-25, 26	CR Vent Intake	Scin.	150' by 10% Dumps	CR Vent to Mode 4	Y
RM-28/28R	Plant Vent Part.	Beta Scin.	85' FHB White Room	None	Y
RM-29	Plant Vent Gross	Ion Ch.	140' by Escape Hatch	Alarms at OES	PAM2
RM-30, 31	Cont Hi Range	Ion Ch.	140' Containment	None	PAM2
RM-34	Area Monitor	Ion Ch.	85' White Room	None	PAM2
RM-41, 42, 43	Gas Decay Tanks	Ion Ch.	64' by Decay Tanks	None	N
RM-44 A&B	Containment Exhaust	Beta	100' GE	Containment Vent Isolation (CVI)	Y
RM-48	Post-LOCA Sample	Ion Ch.	85' Penetration Area	None	N
RM-51, 52, 53, 54	CR Press Intake	GM	140' Turb (Both Ends)	Transfers to other Unit	Y
RM-58, 59	Spent/New Fuel Area	GM	104' Fuel Hdlg Bldg	FHB Vent to Iodine Removal Mode	PAM2
0-RM-60, 61, 62, 63, 64, 65	TSC Area Monitors	GM	100' TSC Areas	None	N
0-RM-66	TSC Air Partic.	Scin.	TSC HVAC Room	None	N
0-RM-67	TSC Noble Gas	Scin.	TSC HVAC Room	None	N
0-RM-68	TSC Lab Air Partic.	Scin.	TSC HVAC Room	None	N
0-RM-69	TSC Lab Noble Gas	Scin.	TSC HVAC Room	None	N
RM-71, 72, 73, 74	Main Stream Line	GM	OC, line before MSIV	None	Y
0-RM-82, 83	TSC Iodine	Scin.	TSC HVAC Room	None	N
0-RM-90, 92	Radwaste Bldg Area	GM	Truck Bay, Laundry Rm	None	N

* Dual setpoint selectable via toggle switch at local indication panel

STEP DESCRIPTION TABLE FOR FR-Z.1Step 4

STEP: Verify Containment Fan Coolers - RUNNING IN EMERGENCY MODE

PURPOSE: To ensure that the containment fan coolers are running in the emergency mode

BASIS:

This step instructs the operator to verify that all containment fan coolers are operating in the emergency mode. If the fan coolers are not running, the operator should manually start the fan coolers. The intent of this step is to provide containment heat removal and therefore reduce containment pressure. In the reference plant design, the containment fan coolers running in the emergency mode provide an alternate heat removal means to the containment spray pumps. The fan coolers operate at low speed under post accident conditions (emergency mode) to prevent the fan from stalling under the higher density accident atmospheric conditions.

The operator should also ensure that proper cooling is being provided to the fan coolers. Typically, cooling water is provided to the fan coolers from either the component cooling water or service water systems.

ACTIONS:

- o Determine if containment fan coolers are running in the emergency mode
- o Start fan coolers in emergency mode

INSTRUMENTATION:

Containment fan cooler indicator lights

CONTROL/EQUIPMENT:

Containment fan cooler switches

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

Emergency Operations

Alignments
Obj 2

The CFCU system no longer has a normal or emergency damper alignment since the filters have been discarded.

- Normal damper alignment is now the same for emergency damper alignment.
- Emergency operations call for CFCU fans to be in LOW speed to prevent the fans tripping on overcurrent due to the extremely dense atmosphere after a LOCA.

Related EOPs
Obj 21

The emergency operating procedures with actions related to the CFCU system include:

EOP Procedure	Title	This EOP provides direction to address ...
E-1	Loss of Reactor or Secondary Coolant	<ul style="list-style-type: none"> • Appendix A-Blackout Emergency Loading of Vital Buses : Ensuring that the vital busses re-loading does not challenge the capability of restored power.
E-1.3	Transfer to Cold Leg Recirculation	<ul style="list-style-type: none"> • verifies a max of 3 CFCUs running if < 2 ASW Pps flowing thru 2 CCW HXs
ECA-1.1	Loss of Emergency Coolant Recirculation	<ul style="list-style-type: none"> • ensures that containment cooling is established. • ensures that if containment cooling is not adequate the CS pumps are started.
FR-Z.1	Response to High Containment Pressure	<ul style="list-style-type: none"> • ensures that containment cooling is established.
FR-Z.3	Response to Containment High Radiation Level	<ul style="list-style-type: none"> • ensures that containment cooling is established.

Actions outside the Control Room may require breaker operations or cooling alignments be made to ensure CFCU operations during emergencies.

DCPP 10/02
ES-401

RO & SRO Written Examination
Question Worksheet

Form ES-401-6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	<u>1</u>	<u>1</u>
	Group #	<u>1</u>	<u>1</u>
	K/A #	<u>076.AK2.01</u>	<u>1</u>
	Importance Rating	<u>2.6</u>	<u>3.0</u>

APE: 076 High Reactor Coolant Activity
AK2 Knowledge of the interrelations between the High Reactor Coolant Activity and the following:
AK2.01 Process radiation monitors.

Proposed Question # 58 :

Unit 1 has been operating at 100% power for three weeks with a 5 gpd tube leak in S/G 1-2.

Which one of the following will occur if RCS activity increases due to a fuel defect?

- A RM-15 counts will increase.
- B S/G 1-2 tube leak flow will increase.
- C RCS Argon-40 will increase.
- D RM-72 counts will increase.

Proposed Answer: A

Explanation:

Technical Reference(s): OP O-4 page 2

Proposed references to be provided to applicants during examination: NONE _____

Learning Objective: 8480 Demonstrate the ability to determine plant implications of each RMS radiation monitor indication.

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

O-4

The Air Ejector Off-gas monitors, RM-15 and 15R, are the most sensitive indicators of small changes in Steam Generator tube leakage. The Plant Process Computer (PPC) receives an input from RM-15 and 15R. The Plot or Archive function of the PPC can be used to detect small changes in the Air Ejector Off-gas count rate.

The High Alarm set point of RM-15 and 15R is set to the activity level that should be seen by the detector if the previously calculated RCS activity still exists and a total of 30 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the High Alarm setpoint may correspond to a leak rate greater than 30 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

The Alert alarm set point of RM-15 and 15R is set to the activity level that would be seen by the detector if the previously calculated RCS activity still exists and a total of 20 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the Alert Alarm setpoint may correspond to a leak rate greater than 20 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

Primary to secondary leak rates can be inferred by comparing existing RM-15, 15R readings to the Alarm and Alert set points. The relationship of leak rate to activity level is linear as indicated activity increases.

Chemistry projects what the RM-15(R) count rate will be for a Steam Generator Tube leak of 30 gpd, 75 gpd and 150 gpd. This information is provided for reference in accordance with CY1.DC3. This projected value can be used by the Shift Foreman to roughly estimate Steam Generator leak rates.

Frequent analysis of RM-15(R) count rate and evaluation of the change in count rate with time is the best indicator of changes in the Steam Generator leak rate.

1.00

The following plant conditions exist after stabilizing from a small break LOCA:

RCS Pressure = 1750 psig
Pzr Level = 20% stable
Subcooling = 45°F
AFW flow = 500 gpm
1 CCP running
2 SI pumps running

How will Pzr level be affected if both SI pumps are shut down?

- A. No change
- B. Initial increase, then slow continuous decrease
- C. Increase
- D. Decrease

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

6743 Explain PZR response during ECCS reduction sequence

Reference Id: P-6157

Must appear: No

Status: Active

User Text: 6743.020304

User Number 1: 0000003.10

User Number 2: 0000004.00

Difficulty: 3.00

Time to complete: 2

Topic: E-1.1 - Pzr level response during ECCS flow reduction

Safety Injection Pumps, Continued

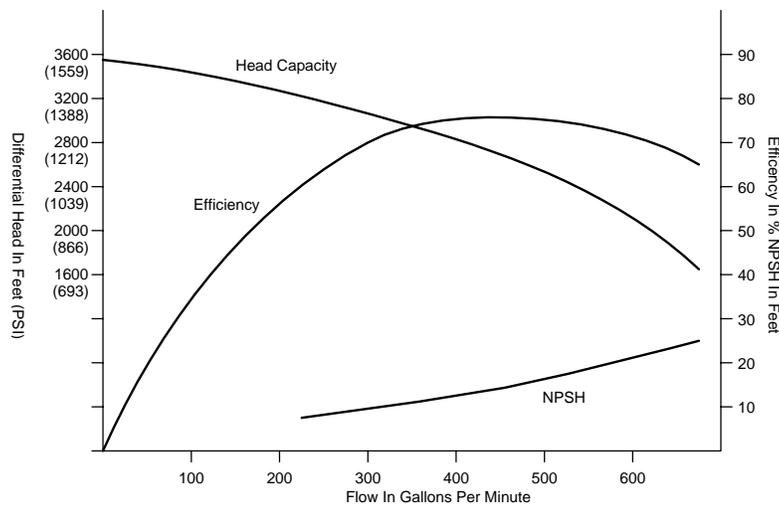
Physical description
Obj 10

Each unit has two 100% capacity pumps with the following characteristics.

Characteristic	Details
Type	10 stage, horizontal centrifugal pump.
Pump drive	4 kV 400 hp induction motor
Design Pressure	1700 psig
Design Temperature	300°F
Shutoff head	1559 psig
Design flow/head	425 gpm @ 2500 ft/1150 psid
Runout flow/head	675 gpm @ 1500 ft/675 psid [GRH35]

SIP operating characteristics
Obj 10

The diagram below illustrates the flow characteristics of the SIPs.



ECC-08

Continued on next page

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
EMERGENCY OPERATING PROCEDURE

EOP FR-C.3
REVISION 9
PAGE 1 OF 7
UNIT

1

TITLE: Response to Saturated Core Conditions
EFFECTIVE DATE

10/26/01

PROCEDURE CLASSIFICATION: QUALITY RELATED

1.0 SCOPE

1.1 This procedure provides actions to restore subcooled core cooling.

1.2 The major actions in EOP FR-C.3 are:

- o Establish ECCS flow to maintain a minimum RCS subcooling.
- o Check for open RCS vent paths.

2.0 VERIFY ENTRY CONDITION FOR EOP FR-C.3

2.1 EOP F-0.2, YELLOW Condition

ID: A-0157 Points: 1.00

With the plant initially at 100% power, a loss of all offsite power results in a reactor trip.

When reactor power drops to about 8-9% (due to prompt drop after the trip), adequate core cooling exists because of:

- A. Continued forced flow
- B. Natural circulation flow
- C. ECCS flow
- D. Sufficiently low core thermal power

Answer: A

Associated objective(s):

State the purpose of RCP system and equipment

Question 1 Details

Question Type: Multiple Choice

Topic: Purpose of RCP flywheel

System ID: 6635

User ID: A-0157

Status: Active

Always select on test: No

Authorized for practice: No

Difficulty: 4.00

Time to Complete: 2

Point Value: 1.00

Cross Reference Number: 03

Motor, Continued

Part/Function table
Obj 6

The table below lists the functions of the parts of the RCP motor.

Part	Function
Flywheel	Increase pump coast down time, to extend the period of forced flow following a pump trip.
Upper/lower radial bearings	Babbitt-on-steel, pivoted-pad bearing assemblies providing maximum shaft stability by minimizing radial movement.
Thrust bearing	Double-acting Kingsbury thrust bearing minimizing axial movement. See Figure RCP-15 on page 2-26.
Upper brg oil cooler	Cools the oil in the upper oil reservoir. A lower bearing oil cooler cools the oil in the lower oil reservoir (not shown).
Oil lift pump and motor	Creates an initial film of oil between the thrust bearing shoes and thrust bearing runner on startup.
Anti-Reverse rotation device	Prevents reverse rotation when the pump is shut down and other RCPs are running, thus preventing excessive starting currents from starting the pump while it is rotating backwards
Space heaters (not-shown)	Prevents moisture condensation in the motor windings. A breaker B contact automatically energizes them when the pump is shut down.
Shaft coupling and spool piece	Connects the motor to the pump <ul style="list-style-type: none"> An 18" spool piece (not shown) is used below the coupling to allow access to and removal of the seal package without removing the motor.
Oil level indicators Upper (shown) Lower (not shown)	Provide local indication of the oil levels in the upper and lower oil reservoirs.
Stator	Insulated, rectangular, copper wire, impregnated with epoxy resin for insulation. Provides for the production of a rotating magnetic field.
Rotor	Copper alloy bars fitted into rotor slots, brazed to a copper connection ring at both ends. The rotating magnetic field in the stator induces an electrical charge in the rotor, developing a torque capable of driving the pump impeller.

Continued on next page

#

1.00

During the performance of OP AP-24, while preparing to re-align one charging pump through charging injection, RWST level decreases to 33%. Neither Safety Injection nor Containment Isolation Phase A has actuated.

What action should be taken?

- A. Align the SI system for cold leg recirculation using EOP E-1.3, Transfer to Cold Leg Recirculation.
- B. Manually stop all ECCS pumps and go to EOP ECA-1.2, LOCA Outside Containment.
- C. Manually initiate SI and go to EOP E-1.3, Transfer to Cold Leg Recirculation.
- D. Manually stop all ECCS pumps and go to EOP ECA-1.1, Loss of Emergency Coolant Circulation.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3477	Describe the major actions of abnormal operating procedures
------	-------------------------------------------------------------

Reference Id: P-65994
 Must appear: No
 Status: Active
 User Text:
 User Number 1:
 User Number 2:
 Difficulty: 3.00
 Time to complete: 2
 Topic: AP-24 - Action to take if RWST level decreases to 33%

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

CAUTION 1: IF RWST level decreases to less than 33%, the SI system should be aligned for cold leg recirculation using EOP E-1.3.

CAUTION 2: If RCS leakage is isolated, steps 32 (Page 17) through 36 should be immediately performed to terminate SI flow.

4. REALIGN One Charging Pump Through Charging Injection

a. Charging Pps – ONLY One Running

a. Stop charging pump to establish only one pump running.

b. OPEN 8805A AND 8805B

c. OPEN Charging Injection Valves:

- 8803A AND 8803B
- 8801A AND 8801B

d. CLOSE 8107 AND 8108

e. Verify FCV-128 – Fully OPEN

5. EVACUATE Non-essential Personnel In Containment:

a. Activate the containment evacuation alarm

b. Notify Plant Personnel of LOCA via the Public Address system

6. ACTUATE Containment Isolation Phase A:

a. Actuate Phase A

b. Check Phase A portion of Monitor Light Box B:

- Red Activated Light – ON
- White Status Lights – OFF

b. Manually Close the Phase A Isol vlvs with White Status Lights - ON

ACTION / EXPECTED RESPONSE**RESPONSE NOT OBTAINED****7. CHECK If RCPs Must Be Stopped:**

Stop affected RCP(s).

- RCP No. 1 Seal differential pressure – GREATER THAN 255 PSID

AND

- RCP No. 1 Seal Leakoff flow – INSIDE APPENDIX B guidance

8. DISPATCH Personnel To Locally Restore Power To Locked Out ECCS Equipment:

- Install UC fuses on SI Pump with its breaker racked in
- Make all other Charging/SI pumps available

9. CHECK If One Safety Injection Pump Should Be Started:

- Check the following
 - PZR Level – LESS THAN 12% [36%]

OR

 - RCS Subcooling based on core exit T/Cs – LESS THAN 20°F

GO TO step 13 (Page 6)

- Establish flow from one Safety Injection Pump:
 - Verify SI system aligned for Cold Leg injection
 - Start one SI Pp

STEP DESCRIPTION TABLE FOR ARG-2 Step 4 - Caution

CAUTION: If RWST level decreases to less than (6), the SI system should be aligned for cold leg recirculation using (7).

PURPOSE: To guarantee coolant flow to the core by switching to cold leg recirculation if the RWST level decreases below the switchover setpoint.

BASIS:

If the switchover level in the RWST is reached, which could happen at any time during the course of guideline ARG-2 depending upon the size of the LOCA, the operator should immediately align the SIS for cold leg recirculation using the appropriate plant specific procedure to maintain coolant flow to the core.

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

- o (6) Enter plant specific value corresponding to RWST switchover setpoint in plant specific units.
- o (7) Enter plant specific procedure for transferring the SI System to cold leg recirculation.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP L-5
REVISION 75
PAGE 41 OF 52

TITLE: Plant Cooldown From Minimum Load to Cold Shutdown

UNIT 1

- 9.2.12 The plant is particularly vulnerable to exceeding the PZR Heatup and Cooldown rates while in Mode 5, with the RCS at cooler temp., and a bubble in the PZR. The Heatup limit is 100°F per hour, and the cooldown rates are 200°F per ECG 7.5.
- The cooldown is self explanatory, however the Heatup can catch you off guard. If there is plant transient which causes the plant to cool greater than 100°F, caution should be used when trying to restore temp.
 - Cooldown RATE shall be assured by limiting the cooldown rate to 200°F per hour. The cooldown should be uniform over time thus avoiding large step changes.
 - If T_{COLD} is 200°F less than the PZR liquid space temperature, do not exceed a PZR fill rate (net charging rate) of 35 GPM. Energize all PZR heater banks during the fill process. Limit any PZR fill rate so as not to exceed a 200°F per hour cooldown.^{T31586, T32450}
 - Pressurizer cooldown and heatup rates should be monitored closely, during any evolution planned or unplanned affecting makeup or rejection.
- 9.2.13 When operating in a solid condition with RCS pressure being maintained by CVCS-1-PCV-135, changes to the flow rate in the RHR loops or starting and stopping the RHR pumps will result in changes in RCS pressure. CVCS-1-PCV-135 should be operated in MANUAL and adjust as necessary to minimize RCS pressure transient.
- 9.2.14 If the RCS is to be depressurized with the head on for an extended period, the RCS is susceptible to gas accumulation in the vessel head. Actions which may be taken to minimize/detect gas accumulation include.^{T35812}
- Verify at least one RVLIS train in service and monitor for decreasing vessel level.
 - Monitor Pressurizer level for increases that appear larger than can be explained by a charging/letdown mismatch.
 - Reduce VCT Nitrogen pressure, or vent to atmosphere.
 - If gas accumulation is suspected, contact system engineering and maintenance to setup a temporary vent path to remove any accumulated gas.
- 9.2.15 Use of N/A shall be in accordance with AD2.ID1, "Procedure Use and Adherence," with the following additions. Any N/A for non-conditional steps must be explained sufficiently in the Remarks Section that this procedure will stand alone, and must be initialed by the SFM to indicate concurrence with the action. As stated in AD2.ID1, conditional steps that are N/A'd do not require explanations or specific SFM approval.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP L-5
REVISION 75
PAGE 43 OF 52
UNIT 1

TITLE: Plant Cooldown From Minimum Load to Cold Shutdown

DATE / TIME / INITIALS

9.3.4 Place the RCS in a SOLID condition as follows:^{T31830}

NOTE: PCV-135 can be operated in manual or automatic.

- a. Exercise CVCS-1-PCV-135 and locally verify that it strokes smoothly through its entire valve travel. (A partial STP V-2I2 in conjunction with stroke of PCV-135 including TCV-149 and HCV-133 is recommended)
- b. Energize all pressurizer heater banks while adding water to the pressurizer.
- c. Increase net charging to fill the PZR at a rate less than 35 gpm.^{T31586, T32450}
- d. Use sprays as necessary, to control RCS pressure as PZR level increases.
- e. As the bubble collapses, as indicated by inability to control pressurizer pressure with spray, reduce charging and adjust the position of CVCS-1-PCV-135 as necessary to control RCS pressure.
- f. When PZR is solid, turn OFF all PZR heaters and CBC Tag to SFM using Attachment 12.3, Section V.
- g. Close the PZR steam space purge to VCT by closing sample line valves NSS-1-9354A and B.
- h. With CVCS-1-PCV-135 in MANUAL, isolate normal letdown one orifice valve at a time, (CVCS-1-8149C, 8149B, and 8149A), and minimize RCS pressure transients.

_____/_____/_____
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NOTE: RHR Letdown Flow is maintained via CVCS-1-HCV-133.^{T35757}

- i. Close LCV-459, LCV-460 and CVCS-1-8152 to prevent crud entrapment during forced oxygenation.

_____/_____/_____

Seabrook Unit

WEC 5/30/2003

1

ID: 24643

Cog level: 2

Plant conditions are as follows:

\$RCS temperature is 300 F

\$RCS pressure is 340 psig

\$Train A Residual Heat Removal (RHR) is in service with flow set at 3500 gpm; Train B RHR is in ECCS standby mode

\$The current cooldown rate is 12 F/hr.

\$The US orders the cooldown rate increased from 12 F/hr to 40 F/hr over 10 minutes while maintaining RHR flow relatively constant.

Which of the following describes how the operator will increase the cooldown rate?

- A. Throttles the RHR heat exchanger bypass valve, RHR-FCV-618, in the closed direction. This causes less water to flow through the RHR heat exchanger bypass line. Flow is automatically increased through the RHR heat exchanger to maintain the combined flow rate constant at 3500 gpm.
- B. Throttle open RHR-FCV-610 RHR pump A mini-flow. The RHR heat exchanger outlet and bypass valves require no throttling since RHR system flow rate remains constant at 3500 gpm.
- C. Throttles the RHR heat exchanger outlet valve, RHR-HCV-606, in the closed direction such that RHR system water will spend more time in the RHR heat exchanger to be cooled further by PCCW. Flow is automatically increased through RHR-FCV-618 to maintain the combined flow rate constant at 3500 gpm.
- D. Throttles the RHR heat exchanger outlet valve, RHR-HCV-606, in the open direction. This causes more RHR system water to flow through the RHR heat exchanger. Flow is automatically decreased by RHR-FCV-618 to maintain the combined flow rate constant at 3500 gpm.

D - correct - the operator manipulates the outlet valve to send more or less water through the Hx to be cooled. The bypass valve modulates to maintain 3500 gpm total coming from the Hx and bypass line.

A - incorrect - will reduce CD rate

B - incorrect - PCCW is not manipulated by procedure;

C - incorrect - bypass valve is modulated, the Hx outlet is not.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP B-2:V
REVISION 24
PAGE 11 OF 15
UNIT 1

TITLE: RHR - Place In Service During Plant Cooldown

- 6.2.32 Crack open the RHR heat exchanger bypass flow control valve HCV-670 while observing temperature recorder TR-636/TR-648.
- a. Verify that RHR temperature is slowly increasing to equal the RCS temperature.

- 6.2.33 When the RHR and RCS temperatures are approximately equal, slowly open HCV-670 until the flow to the RCS cold legs is approximately 3000 gpm.

NOTE: The Plant Cooldown may be performed using one RHR Pump with two RHR Heat Exchangers OR two RHR Pumps with two RHR Heat Exchangers. (See P&L 5.3 for flow limitations)

- 6.2.34 Start the second RHR Pump, if desired.

- a. Verify FCV-641A (FCV-641B), minimum flow recirculation valve, OPENS and then CLOSES.

- 6.2.35 When the system temperatures have stabilized, open HCV-670 until the total flow to the RCS cold legs is approximately 3000 to 5000 gpm as indicated on FI-970 and FI-971.

- 6.2.36 Commence the primary system cool down process per OP L-5.

NOTE: CCW temperatures are available for trending on the PPC as points T0611A and T0612A.

- a. During the cooldown, monitor CCW return line temperature from the RHR Heat Exchanger. Temperature shall not exceed the maximum qualified piping temperature of 145°F.
- b. Gradually open the RHR heat exchanger flow control valves HCV-637 and 638 while closing off the bypass flow control valve HCV-670 to maintain a cool down rate consistent with OP L-5.

NOTE: It may be necessary to place PCV-135 in MANUAL to achieve an adequate letdown flow rate due to the low RCS pressure.

- 6.2.37 Initiate letdown flow to the CVCS by slowly opening letdown flow control valve HCV-133.

- 6.2.38 During the cool down process, proportion the flow through the RHR heat exchangers to ensure compliance within the following limits:

- a. Total RHR flow to the RCS cold legs should be limited to approximately 5000 gpm.
- b. The temperature of the component cooling water leaving the CCW heat exchangers should not be allowed to exceed 120°F.

- 6.2.39 The RHR system can now be used to control the RCS temperature and letdown for operation in Mode 4 and 5.

#

1.00

A large-break LOCA has occurred. Two of the ECCS accumulators fail to discharge, as indicated by constant accumulator level and pressure readings. The MAJOR safety concern is that

- A. insufficient water will be available for reactor core refill which may result in excessive peak cladding temperatures.
- B. discharge of the accumulators at a later time will interfere with natural circulation core cooling.
- C. discharge of the accumulators at a later time could initiate a rapid RCS cooldown and re-initiate break flow.
- D. insufficient water will be available in the recirculation sump to provide long-term core cooling ability during a post-LOCA recovery.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

5459	Explain why ECCS Accumulators are isolated for various plant conditions.
------	--------------------------------------------------------------------------

Reference Id: P-0545
 Must appear: No
 Status: Active
 User Text: 5459.020065
 User Number 1: 0000003.30
 User Number 2: 0000004.00
 Difficulty: 3.00
 Time to complete: 2
 Topic: ECCS - Safety concern with accumulators not inj per design.

Overview, Continued

ICC, continued *Obj 15,51,52*

Smaller LOCAs, which slowly depressurize the RCS, rely on the HPSI system to refill the RCS. Without this system, a small LOCA would continue to remove RCS coolant, eventually uncovering the core.

The core remains uncovered until RCS pressure drops below the Accumulator setpoint. An ICC condition may develop during this time. Larger LOCAs, which depressurize the RCS quickly, rely on the Accumulators and low head SI to refill the RCS. Without these systems, the core would quickly uncover and remain uncovered until the HPSI system could refill the RCS. An ICC condition may develop during this refilling period. The probability of a larger LOCA is lower than the probability of a small LOCA. When combined with a probability of losing both Accumulators and the low head SI, the probability of a large LOCA inadequate core cooling event is very small indeed.

The time required for the symptoms of ICC to develop is dependent upon the RCS volume, break size and location, core heat generation rate and the type and amount of SI available. A small LOCA without high-head SI at a large 1000 MWe four loop plant near the beginning of life will take a considerably longer period of time to reach an ICC condition than a similar size break at a smaller two loop plant near the end of life.

In any case, either the failure of, or the intentional operator action to turn off the SI system is required to obtain an ICC condition. The SI system, designed to refill the RCS following a LOCA, provides both high pressure and low pressure pumped flow to the RCS. A total failure of this system is considered unrealistic since multiple failures would be required. Manual action in the Control Room to prematurely terminate SI is a more probable cause for loss of SI. This is what happened at TMI.

Continued on next page

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK05-25
Rev. 14
Page 1 of 6

**PRT
PRESS/LVL
TEMP**

02/14/06

Effective Date

QUALITY RELATED

1. **ALARM INPUT DESCRIPTION**

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
318	Pzr Relief Tk Temp Hi	TC471	> 130°F	2.1
367	Pzr Relief Tk Lvl Hi	LC470A	> 89%	2.1
545	Pzr Relief Tk Press Hi and Vent Hdr Isol	PC472X	> 10 psig	2.1
1394	Pzr Relief Tk Lvl Lo	LC470B	< 56%	2.1

2. **OPERATOR ACTIONS**

2.1 **General Actions (All Inputs)**

- 2.1.1 Check the following indications to determine alarm cause (VB2):
 - PI-472, Pzr Relief Tk Press []
 - LI-470, Pzr Relief Tk Level []
 - TI-471, Pzr Relief Tk Temp []
- 2.1.2 IF low PRT level exists,
THEN GO TO Section 2.2. []
- 2.1.3 IMPLEMENT OP AP-1, "Excessive RCS Leakage", to aid in identifying source and magnitude of leakage. []
- 2.1.4 IF source of leakage is a safety valve or PORV,
THEN REFER TO the following for operability concerns:
 - TS 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)" []
 - TS 3.4.13, "RCS Operational LEAKAGE" []

2.1.5 IF high PRT level exists,
THEN perform the following:

- a. Verify CLOSED RCS-1-8030, PRT Pri Wtr Sply (VB2). []

CAUTION: RCS-1-8031, PRT Drain to RCDT, is a slow-closing valve. If closing is not initiated at 60% RCDT level, the RCDT may overflow before the valve is fully closed. Overfilling the RCDT will fill the LI reference legs, which will then require draining by maintenance.

- b. Lower PRT level to approximately 84% by draining to the RCDT as follows:

1. Establish communications with the Aux Senior watch. []

2. Direct the Aux Senior watch to perform the following:

- a) IF RCDT Level reaches 60% as read on LI-188, Reactor Clnt Drain Tank (Aux Control Board),
THEN immediately inform the Control Room. []

- b) Run BOTH RCDT pumps to keep up with the drain rate by holding BOTH pump start switches in the "ON" position. []

3. IF at any time RCDT level reaches 60%,
THEN perform the following:

- a) Immediately CLOSE RCS-1-8031, PRT Drain to RCDT (VB2). []

- b) Allow the RCDT level to lower to approximately to 34% as read on LI-188, Reactor Clnt Drain Tank (Aux Control Board). []

- c) IF desired to continue to lower PRT Level,
THEN OPEN RCS-1-8031. []

4. OPEN RCS-1-8031 to begin draining. []

5. WHEN PRT level reaches 84%,
THEN CLOSE RCS-1-8031. []

- 2.1.6 IF high PRT temperature exists,
THEN reduce PRT temperature to 120°F by performing the following:
- a. OPEN RCS-1-8030, PRT Pri Wtr Sply (VB2). []
 - b. WHEN TI-471, Pzr Relief Tk Temp, is 120°F or lower,
THEN CLOSE RCS-1-8030 (VB2). []
 - c. IF lowering PRT level is desired,
THEN GO TO step 2.1.5 to drain the PRT to the RCDT. []
- 2.1.7 IF high PRT pressure exists,
THEN perform the following:
- a. Verify CLOSED RCS-1-8045, PRT N₂ Supply Isol (VB2). []

NOTE: RCS-1-PCV-472, PRT Vent to Vent Hdr, is NOT available when PRT pressure is 10 psig or above.

- b. IF PRT pressure is 10 psig or above,
THEN lower PRT pressure to slightly less than 10 psig by performing ONE of the following:
 - Lower PRT level PER step 2.1.5. []
 - Direct an Operator to remove blank flange from RCS-1-8048, PRT Vent Vlv To Contmt Atmosphere, and bleed pressure off slowly (Ctmt 91 ft el, PRT). []
- c. WHEN PRT pressure is less than 10 psig,
THEN perform the following:
 - 1. Inform the Aux Building watch of the intention to vent the PRT to the Waste Gas Header. []
 - 2. Verify Waste Gas Header available. []
 - 3. OPEN PCV-472, PRT Vent to Vent Hdr, to continue to lower pressure to approximately 3 psig or until pressure otherwise stabilizes (VB2). []
 - 4. CLOSE PCV-472. []

NOTE: A PRT pressure increase without increases in temperature or level indicates a possible failure of RCS-1-PCV-8035, PRT N₂ Supply.

- d. Direct maintenance to troubleshoot. []

- e. IF there is a slow increase in PRT pressure,
THEN maintain PRT pressure between 3 and 10 psig by performing the following:
 - 1. Verify CLOSED RCS-1-8045, PRT N₂ Supply Isol (VB2). []
 - 2. OPEN PCV-472 to lower pressure to approximately 3 psig or until pressure otherwise stabilizes. []
 - 3. CLOSE PCV-472. []
- f. IF the nitrogen regulator is working properly,
THEN OPEN RCS-1-8045. []

2.1.8 Probable Causes

- a. High PRT pressure, temperature or level:
 - RCS-1-PCV-474, 455C or 456, Pressurizer power relief valves, leaking or lifting
 - RCS-1-8010A, B or C, Pressurizer safety valves, leaking or lifting
 - Malfunction of Pzr pressure control system
- b. High PRT pressure or level:
 - 1. Relief valve (from outside containment) lifting or leaking:
 - CS-1-RV-9007A, Containment Spray Hdr 260 psig
 - SI-1-RV-8858, SI Pump Suct Hdr 220 psig
 - SI-1-RV-8856B, RHR Ht Exch 1-2 600 psig
 - CVCS-1-RV-8125, Charging Pps Suct Hdr 220 psig
 - SI-1-RV-8853B, SI Pp 1-2 Disch Hdr 1750 psig
 - SI-1-RV-8853A, SI Pp 1-1 Disch Hdr 1750 psig
 - SI-1-RV-8851, SI Pps Disch Hdr 1750 psig
 - SI-1-RV-8856A, RHR Ht Exchg 1-1 600 psig
 - CS-1-RV-9007B, Contmt Spray Hdr 260 psig
 - 2. Relief Valve (from inside containment) lifting or leaking:
 - CVCS-1-RV-8121, Seal Wtr Ret Hdr 150 psig
 - CVCS-1-RV-8117, CVCS Letdown Hdr 600 psig
 - RHR-1-RV-8707, RHR Pp Suct Hdr 450 psig
 - SI-1-RV-8856A, B, RHR Pp Inj Loops 1 & 2 600 psig
 - RHR-1-RV-8708, RHR to RCS Hot Legs Hdr 600 psig

- c. High PRT temperature or level due to valve stem leak-off from:
 - RCS-1-8000A, B, C, Pzr PORV Iso
 - RCS-1-PCV-455A, B, Pzr Spray
 - RHR-1-8701 and 8702, Loop 4 to RHR system
 - RCS-1-8033A, B, C, D, Pzr Spray Iso
 - RCS-1-8076, CVCS Ltdn Iso
 - CVCS-1-8143, Excess Ltdn Divert
 - CVCS-1-HCV-123, Excess Ltdn Flow Control
- d. High PRT level:
 - Pressurizer safety valve loop seal drain header valve(s) leaking
 - RCS-1-8030, PRT Pri Wtr Sply, leaking
- e. High PRT pressure due to RCS-1-PCV-8035, PRT N2 Supply, malfunction

2.2 Low PRT Level Alarm

CAUTION: Primary Water fill rate to the PRT is quicker than the venting rate. Do NOT exceed 10 psig in the PRT. RCS-1-PCV-472, PRT Vent to Vent Hdr, is not available to depressurize the RCDT when PRT pressure is 10 psig or above.

- 2.2.1 Monitor PI-472, PRT Press, during filling (VB2). []
- 2.2.2 IF at any time PRT pressure approaches 10 psig, THEN perform the following:
 - a. CLOSE RCS-1-8030, PRT Pri Wtr Sply. []
 - b. WHEN PRT pressure lowers sufficiently, THEN OPEN RCS-1-8030. []
- 2.2.3 OPEN RCS-1-8030, PRT Pri Wtr Sply. []
- 2.2.4 WHEN PRT level is 84%, THEN CLOSE RCS-1-8030. []
- 2.2.5 Verify CLOSED RCS-1-8031, PRT Drain to RCDT (VB2). []
- 2.2.6 IF PRT is still draining to the RCDT, THEN cycle RCS-1-8031 OPEN and CLOSED. []
- 2.2.7 IF PRT level continues to drop, THEN direct an Operator to check PRT and instrument taps for leaks. []

2.2.8 Probable Causes

- RCS-1-8031, PRT Drain to RCDDT, leaking by
- PRT or level tap leaking below the water line

3. **AUTOMATIC ACTIONS**

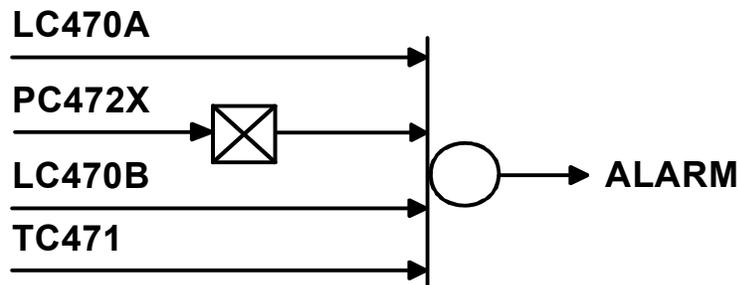
3.1 Vent header isolation at 10 psig (RCS-1-PCV-472, PRT Vent to Vent Hdr, closes)

4. **REFERENCES**

4.1 501127, "Electrical Schematic Diagram - Main Annunciator" (Electrical Drawing Section 8)

4.2 OP A-4B:IV, "Pressurizer Relief Tank - Normal Operation"

5. **LOGIC DIAGRAM**



PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

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UNITS 1 AND 2

Question 34 Reference

TITLE: Malfunction of Component Cooling Water System

Appendix B

CCW Heat Load Isolation

1. SCOPE

- a. This appendix provides guidance for isolation of various CCW heat loads.

2. DISCUSSION

- a. The heat loads are listed in the recommended order for isolation. In addition, the isolation steps may be used to isolate equipment for leaks into or out of the CCW system.
- b. The CFCUs and the SFP Heat Exchanger may be used as heat sinks and should not be isolated from the CCW system unless leaking or the temperature in the CCW system will cause an increase in containment temperature above that allowed by Tech Specs.
- c. If letdown is desired, Normal Letdown at 45 gpm is less of a heat load on the CCW system than Excess Letdown at 40 gpm.

3. INSTRUCTIONS

- a. Boric Acid Evaporator
 - 1) Shutdown the Boric Acid Evaporator using OP B-1B:I, Section 6.3 for guidance.
 - 2) Shut CCW Supply and Return to BA Evap
 - Close CCW-351
 - Close CCW-352
 - 3) If cooling still required, open BA Evap Unit Crosstie Valves CCW-2-355 and CCW-2-356.
- b. Auxiliary Steam Drain Receiver Vent Condenser
 - 1) Transfer CCW supply to Unit 2(1):
 - a) Open CCW-2(1)-355 and CCW-2(1)-356.
 - b) Shut CCW-1(2)-355 and CCW-1(2)-356.
- c. Letdown Heat Exchanger
 - 1) Isolate Letdown
 - a) Close CVCS-8149A, B, C
 - b) Close LCV-459/460
 - 2) Reduce charging flow to minimum while maintaining 8-13 GPM seal injection flow to each RCP.
 - 3) Isolate normal or alternate charging as applicable:
 - Close CVCS-8146
 - Close CVCS-8147

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Question 34 Reference

TITLE: Malfunction of Component Cooling Water System

Appendix B (Continued)

- 4) Verify closed HCV-133, Letdown from RHR system.
 - 5) Place in manual and close PCV-135, Letdown Back Pressure Control Valve.
 - 6) Isolate CCW flow to the Letdown Heat Exchanger
 - Close CCW-406
 - Close CCW-409
 - 7) Establish Excess Letdown per OP B-1A:IV.
 - 8) Verify PZR Pressure and Level are being maintained or trending toward the programmed band.
- d. Seal Water Heat Exchanger
- 1) Isolate Reactor Coolant flow to the HX.
 - a) Open HX Bypass valve CVCS-8381(SC)
 - b) Close HX inlet valve CVCS-8398A(SO)
 - c) Close HX outlet valve CVCS-8398B(SO)
 - 2) Isolate CCW flow to the HX.
 - Close supply valve CCW-401
 - Close return valve CCW-405
 - 3) If the HX is being isolated due to leakage from the CCW system into the CVCS system, perform the following additional steps:
 - a) Take manual control of HC-130B (TCV-130) to control VCT outlet temp to 90-100°F.
 - b) Consider increasing letdown flow to 120 GPM per OP B-1A:XII in order to provide more cool water into the VCT.
 - 4) Monitor RCP Seal/Bearing Temperatures. If any RCP No. 1 Seal outlet temp or RCP Radial Bearing outlet temp cannot be maintained below the alarm setpoint,
 - a) Trip the Reactor
 - b) Trip the RCPs
 - c) GO TO EOP E-O, REACTOR TRIP OR SAFETY INJECTION

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Question 34 Reference

TITLE: Malfunction of Component Cooling Water System

Appendix B (Continued)

CAUTION: Do not continue in this Appendix until the Reactor has been Tripped per Section A of this procedure.

- e. CCW flow to the RCPs
 - 1) Check Reactor Tripped and all RCPs secured.
 - 2) Isolate CCW Flow to the RCPs and Reactor Vessel Supports
 - Close FCV-749
 - Close FCV-750
 - Close FCV-356
- f. VERIFY Charging Pump 3 secured.
- g. ISOLATE Excess Letdown, if in service.
 - CLOSE CVCS-8166
 - OR-
 - CLOSE CVCS-8167
 - CLOSE HCV-123
- h. If CCW temperature continues to increase
 - 1) IMPLEMENT APPENDIX C of this procedure.
 - 2) VERIFY the following pumps secured:
 - RHR Pumps 1 and 2
 - SI Pumps 1 and 2
 - ECCS CCP not being supplied with fire water for cooling
 - 3) Place all CFCUs in fast speed and PURGE Containment as necessary to maintain CCW temperatures less than 120 deg F.
 - 4) Verify CCW flow to the Spent Fuel Pool HX is at least 2800 gpm (FI-197, 100' El Penetration area)
 - 5) Consider removing one train of ECCS pumps from service to limit CCW temperatures in the event of a Safety Injection.

CCW Pumps, Continued

Logic

Obj 8, 25, 26

The logic associated with CCW pump operation is described in the following tables:

With the breaker transfer switch selected to CONT RM:

If the VB1 Auto/Manual switch is in ...	Then the pump will ...
AUTO	Start on: <ul style="list-style-type: none"> • low vital header pressure 69.6 psig (with no SIS or Transfer to Diesel) NOTE: The start will be delayed until lube oil pressure reaches 6 psig.
MANUAL	Manual start and stop (overcurrent reset also permitted) NOTE: The start will be delayed until lube oil pressure reaches 6 psig.
Pump starts independent of Auto/Manual selection	
When the following occurs...	the pump will start...
4 kV Auto Transfer to Diesel (assumes no SI signal)	after a time delay of 5 sec.
SI signal	after a time delay of \cong 18 seconds (14 sec for 1-3). (If bus voltage is lost, the time delay will start upon voltage restoration)

With the breaker transfer switch selected to HSD PNL:

If the HSDP Auto/Manual switch is in ...	Then the pump will ...
AUTO	Start on: <ul style="list-style-type: none"> • 4 kV Auto Transfer to Diesel (no SI signal) • SI signal • Manual NOTE: The LOW header pressure auto-starts are disabled.
MANUAL	Manual start and stop. NOTE: All auto-starts are disabled. The start will be delayed until lube oil pressure reaches 6 psig.

Continued on next page

CCW Pumps, Continued

Logic
(continued)

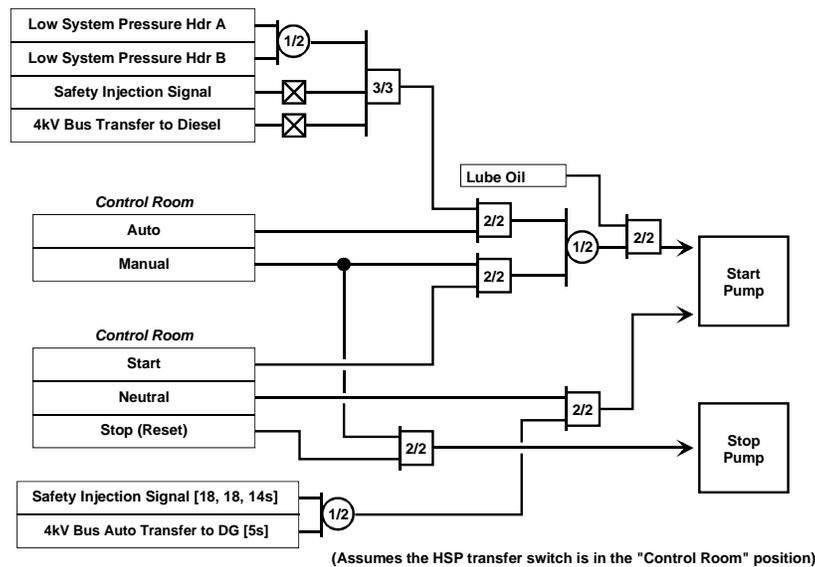
When controlling CCW pumps from the HSDP:

- all automatic trips continue to function normally,
- the manual trip at the HSDP and the switchgear still operate normally, and
- all CCW pump lights in the Control Room, with the exception of the white light, are extinguished.

Logic diagrams for pump starts and stops from the control schemes

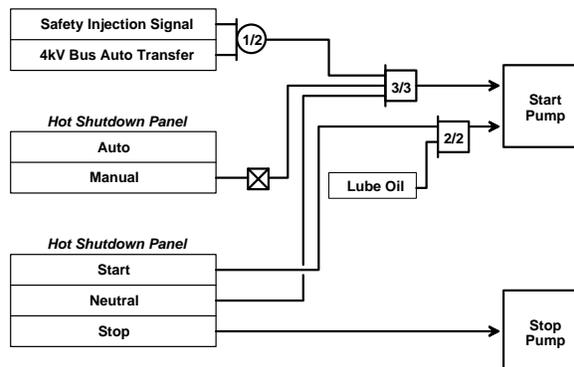
Obj 8, 9, 26

The following logic diagram assumes the transfer switch at the breaker in the CONT RM position:



CCW-03

The following logic diagram assumes the transfer switch at the breaker in the HSD PNL position:



CCW-04

Continued on next page

***** UNCONTROLLED PROCEDURE - DO NOT USE TO PERFORM WORK or ISSUE FOR USE *****

PACIFIC GAS AND ELECTRIC COMPANY
 DIABLO CANYON POWER PLANT

NUMBER EOP E-3
 REVISION 27
 PAGE 30 OF 47

TITLE: Steam Generator Tube Rupture

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

35. CONTROL RCS Pressure And Charging Flow To Minimize RCS-To-Secondary Leakage:
 (Continued)

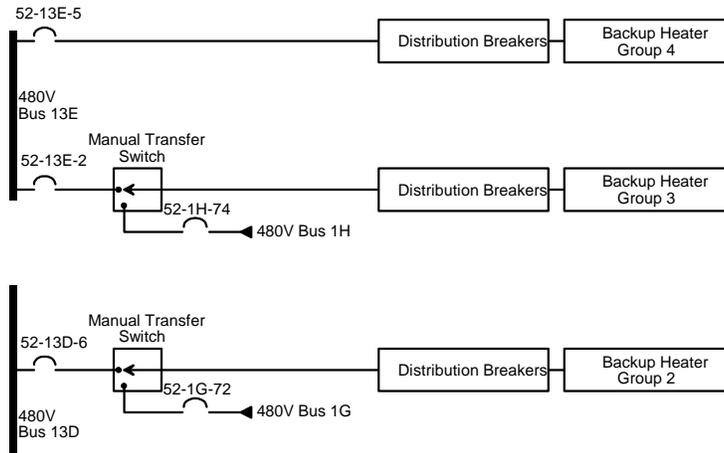
c. Decision table:

	RUPTURED S/G NR LVL INCREASING	RUPTURED S/G NR LVL DECREASING	RUPTURED S/G NR LVL OFF SCALE HIGH
PZR LEVEL LESS THAN 17%[47%]	<ul style="list-style-type: none"> Increase Charging Flow <p style="text-align: center;"><u>AND</u></p> <ul style="list-style-type: none"> Depressurize RCS 	Increase Charging Flow	<ul style="list-style-type: none"> Increase Charging Flow <p style="text-align: center;"><u>AND</u></p> <ul style="list-style-type: none"> Maintain RCS And Ruptured S/Gs Pressures Equal
PZR LEVEL BETWEEN 17%[47%] <u>AND</u> 50%	Depressurize RCS	Verify PZR Heaters - ON	Maintain RCS And Ruptured S/Gs Pressures Equal
PZR LEVEL BETWEEN 50% <u>AND</u> 74%[66%]	<ul style="list-style-type: none"> Depressurize RCS <p style="text-align: center;"><u>AND</u></p> <ul style="list-style-type: none"> Decrease Charging Flow 	Verify PZR Heaters - ON	Maintain RCS And Ruptured S/Gs Pressures Equal
PZR LEVEL GREATER THAN 74%[66%]	Decrease Charging Flow	Verify PZR Heaters - ON	Maintain RCS And Ruptured S/Gs Pressures Equal

Backup Heaters, Continued

Power distribution diagram

The diagram below illustrates the electrical power distribution to Backup Heater groups.



PZR-24

Each distribution breaker supplies three heaters, one per phase.

Power supplies Obj 14

The table below lists the power supplies to the Backup Heaters.

Heater group	Power Supply	
	Group 2	Normal
Emergency		480V bus G
Group 3	Normal	480V bus E
	Emergency	480V bus H
Group 4	Normal	480V bus E

Continued on next page

Backup Heater HSDP Controls, Continued

Backup heater automatic actions

Obj 12

The table below describes the conditions under which the Backup Heaters are energized.

Switch Positions	Condition
<ul style="list-style-type: none"> • Transfer Switch in CONT RM • Normal Power Aligned 	<ul style="list-style-type: none"> • Backup Heater Control switch on CC-1 is taken to ON and released to AUTO • Pressure deviation-low from setpoint, MPC output less than or equal to 9.4% (-25 psi), when the Backup Heater Control switch is in AUTO-after-OFF • Level deviation-high from setpoint, +5% of span (LC-459E), when the Backup Heater Control switch is in AUTO-after-OFF
<p><u>Group 2</u></p> <ul style="list-style-type: none"> • Transfer switch in CONT RM position • Vital power aligned 	Backup Heater Control switch on CC-1 to ON
<p><u>Group 3</u></p> <p>Vital power aligned</p>	Backup Heater Control switch on CC-1 to ON
<p><u>Group 2 & 4</u></p> <ul style="list-style-type: none"> • Control Transfer Relay Cutout Switch in CUTIN • Transfer switch in LOCAL (HSDP) 	Backup Heater Local Control switch (HSDP) to ON (<u>note</u> : for group 2 only, normal power must be aligned)

Continued on next page

Backup Heater HSDP Controls, Continued

Backup heater automatic actions (continued)

The table below describes the conditions under which the Backup Heaters are deenergized.

Switch Positions	Condition
<ul style="list-style-type: none"> • Transfer switch in CONT RM • Normal power aligned 	<ul style="list-style-type: none"> • Backup Heater Control switch on CC-1 is placed in OFF. • MPC output $\geq 15.6\%$ when the Backup Heater Control switch is in AUTO-after-OFF. • Level deviation-high reduced to less than +5% of span (LC-459E) when the Backup Heater Control switch is in AUTO-after-OFF. • Low Pressurizer level, 17% of span (LC-459C or 460C), when the Backup Heater Control switch is in AUTO-after-OFF or AUTO after-ON. • Overcurrent.
<ul style="list-style-type: none"> • Transfer switch in CONT RM • Vital power aligned 	<ul style="list-style-type: none"> • Backup Heater Control switch on CC-1 to OFF. • Safety Injection signal trips the breaker to prevent overloading of the vital bus; the breaker cannot be reclosed until the SI signal is reset.
<u>Groups 2 and 4</u> <ul style="list-style-type: none"> • Control Transfer Relay Cutout Switch is CUTIN • Transfer switch in LOCAL (HSDP) • Normal power aligned (Gp 2) 	<ul style="list-style-type: none"> • Backup Heater Local Control switch on HSDP to OFF • Low Pressurizer level, 17% of span (LC-459C or 460C)

Continued on next page

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP E-0.3
REVISION 15A
PAGE 7 OF 16

TITLE: Natural Circulation Cooldown With Steam Void in Vessel
(With RVLIS)

UNIT 1

ACTION EXPECTED RESPONSE

RESPONSE NOT OBTAINED

3. **CONTINUE RCS Cooldown And
Initiate Depressurization Toward
RHR Entry Conditions (Continued):**

- | | |
|---------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| <p>f. Use Auxiliary Spray and Depressurize as follows:</p> <ol style="list-style-type: none"> 1) Open both PCV-455A and B 2) Open 8145 or 8148, Auxiliary Spray Vlvs 3) Close 8146 and 8147, Normal and Alt Chg to LP4 and LP3 Cold Leg Vlvs 4) Control RCS Pressure by: <ul style="list-style-type: none"> o Adjusting Charging flow o Bypassing Auxiliary Spray with PCV-455A or B o Controlling PZR Heaters as necessary g. Maintain Delta-T between Aux Spray flow and PZR Steam Space LESS THAN 320°F (TI-454 and TI-126, or PPC address U0498) | <p>11) <u>IF</u> Letdown <u>CANNOT</u> be established,
<u>THEN</u></p> <ol style="list-style-type: none"> (a) Use one PZR PORV to Depressurize the RCS. (b) GO TO Step 4 (Next Page). <p>-----</p> <p>f. <u>IF</u> Auxiliary Spray is <u>NOT</u> desired,
<u>THEN</u> Use one PZR PORV

<u>AND</u>
GO TO Step 4 (Next Page)</p> <p>-----</p> <p>g. Perform the following:</p> <ol style="list-style-type: none"> 1) Stop Auxiliary Spray flow. 2) Continue Depressurization using one PZR PORV. 3) GO TO Step 4 (Next Page). <p>-----</p> |
|---------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|

Control Interlocks and Permissives

Signal	Setpoint	Logic	Interlock	Function
C-1 Intermediate Range Rod Stop (PK03-08)	> 20% power (current equiv.)	1/2	Blocked when trip is blocked (P10)	Stops outward rod motion.
C-2 Power Range Rod Stop (PK03-09)	> 103% nuclear power	1/4	Can be bypassed at NI cabinets	Stops outward rod motion.
C-3 OTdT Rod Stop & Turbine Runback (PK04-04/08-09)	1% below calculated trip setpoint	2/4		Stops outward rod motion. Runs back turbine at 10%/min net rate
C-4 OPdT Rod Stop & Turbine Runback (PK04-05/08-10)	1% below calculated trip setpoint	2/4		Stops outward rod motion. Runs back turbine at 10%/min net rate
C-5 Turbine Low Power (PK-8-11)	< 15% turbine load (PT-505)	1/1		Stops auto outward rod motion.
C-7A Turbine Load Rejection (PK08-12)	10% loss of load within 140 seconds (PT-506)	1/1	Manually reset	Arms 40% steam dump valves in Tavag mode.
C-7B Turbine Load Rejection (PK08-13)	50% loss of load within 140 seconds (PT-506)	1/1	Manually reset	Arms 35% & 10% steam dump valves in Tavag mode.
C-9 Condenser Available (PK08-14) (A)	> 20" Hg Vacuum -and- Circ water pump running (breaker closed)	2/2		Arming signal for 40% steam dump valves.
C-11 Bank D Rod Stop (PK03-15) (A)	Control Bank D > 220 steps			Stops auto outward rod motion.
C-20 AMSAC Armed (PK08-15) (A)	PT-505/506 equiv. to > 40% Core Rated Thermal Power (PT-505/506)	2/2		Arms AMSAC

(A) Indicates AMBER lens (a permissive which is normally lit during high power operation)

A "permissive" is a condition which permits certain actions to occur.
An "interlock" is a condition which prevents certain actions from occurring.

#

1.00

What is the main concern during a loss of secondary heat sink accident, assuming no operator-initiated recovery actions?

- A. Core uncover and overheating.
- B. S/G dryout and tube damage from overheating.
- C. Excessive differential pressure across the S/Gs tubes.
- D. Uncontrolled RCS cooldown and reactor restart.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

11317	Discuss the analyzed events associated with a Loss of Heat Sink and be able to describe the identified operator actions used to mitigate this event.
11319	Describe the loss of feedwater event leading to core damage.

Reference Id: M-0068
Must appear: No
Status: Active
User Text: 11317.
User Number 1:
User Number 2:
Difficulty: 1.00

1. INTRODUCTION

Guideline FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, provides guidance to address an extreme challenge (i.e., RED priority) to the Heat Sink Critical Safety Function that results if total feed flow is below a minimum value and level is below the narrow range in all SGs at any time. Less severe challenges to the Heat Sink Critical Safety Function that result from secondary inventory concerns on individual steam generators are discussed in the background documents for guidelines FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL, and FR-H.5, RESPONSE TO STEAM GENERATOR LOW LEVEL.

An early indication that secondary heat transfer capability may be challenged is that auxiliary feedwater (AFW) flow is not available to any steam generator. Following a reactor trip and safety injection (SI), main feedwater isolation is automatically initiated. Auxiliary feedwater flow to the steam generators must be automatically or manually initiated in order to maintain adequate secondary inventory for decay heat removal. Consequently, a failure of the AFW system results in a challenge to the Heat Sink Critical Safety Function. If reactor trip and SI occur, the operation of the AFW system is verified in guideline E-0, REACTOR TRIP OR SAFETY INJECTION, prior to Status Tree monitoring. If minimum AFW flow is not being provided, the operator is directed to implement guideline FR-H.1.

The objective of guideline FR-H.1 is to maintain reactor coolant system (RCS) heat removal capability by establishing feed flow to an SG or through establishing RCS bleed and feed heat removal. Guideline FR-H.1 is entered at the first indication that secondary heat removal capability may be challenged.

This permits maximum time for operator action to restore feedwater flow to at least one steam generator before secondary inventory is depleted and secondary heat removal capability is lost. Once secondary heat removal capability is lost, RCS bleed and feed must be established to minimize core uncover and prevent an inadequate core cooling condition.

During Period 5, flow out of the pressurizer PORVs is a subcooled liquid. Even the minimum capacity pressurizer PORVs are expected to be large enough to maintain RCS pressure at the PORV set pressure. Thus, during Period 5, the RCS pressure remains constant at the PORV setpoint. If operator action is not taken, the system will continue in this mode until decay heat addition causes the reactor fluid to increase to the saturation temperature for water at the pressure of the PORV setpoint (e.g., T_{SAT} at 2335 psig is 659°F). Therefore, Period 5 (without operator action) is a time period where core decay heat energy is absorbed by the RCS fluid and the subcooling of the RCS liquid inventory is reduced. Furthermore, a great deal of RCS inventory is lost out of the PORVs in order to maintain pressure at the PORV setpoint.

Period 6

When the RCS fluid temperature reaches saturation, boiling will begin. The generation of steam within the RCS results in a large volumetric increase in the RCS inventory and the pressurizer PORVs may not be able to compensate for this. The result is that system pressure will rise until either the PORVs can handle the volumetric increase or the pressurizer code safety valves open relieving the excessive volumetric buildup in the RCS. In the example case (Figure 1) the PORVs are able to compensate for the steam generation at a system pressure of around 2405 psig and, thus, safety valves did not open. However, the PORVs are now being held open continuously by the steam generation rate in the core, and this condition will persist until the system boils off all the liquid inventory above the core and partial core uncover occurs. Significant core uncover and potential core damage will follow unless operator action is taken to initiate bleed and feed or feedwater capability is restored.

During the time interval covered by Periods 1 through 6, an automatic SI signal will not have been generated. If SI were manually actuated, it would not be effective in preventing core uncover due to the limited capability of the SI pumps to inject when RCS pressure is at or above the pressurizer PORV setpoint. The mass flow rate out of the pressurizer PORVs is anywhere from 50 to 100 lbm/sec. The charging/SI pump system on the reference plant can inject

Safety Injection Signals

Signals

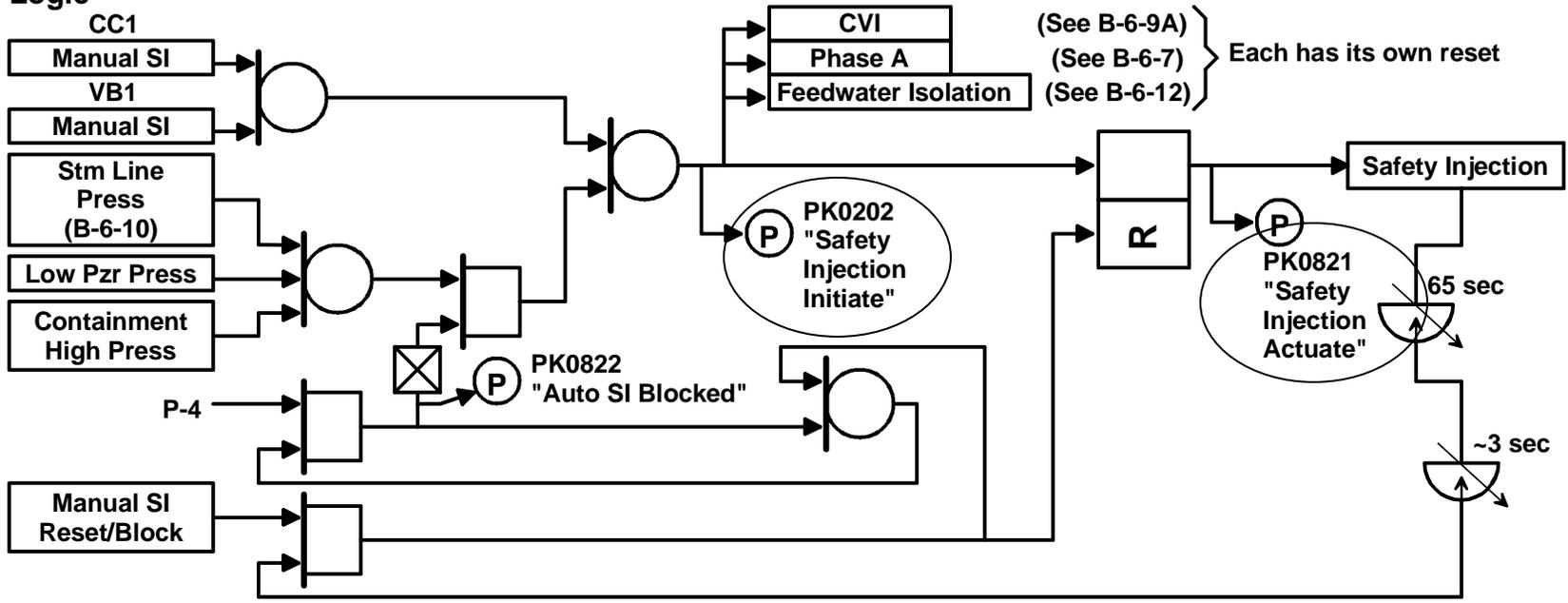
("S" Signal)

Signal	Setpoint	Coincidence	Interlocks	Protection Afforded
Manual	(NA)	1/2	None	Operator judgement
Containment High Pressure	3 psig	2/3	None	High energy line break inside containment
Pressurizer Low Pressure	1850 psig	2/4	P-11	LOCA
Low Steam Line Pressure	600 psig lead-lag	(See B-6-10)	P-11	Steam line break

Question 40 Reference

Logic

B-6-5



Rev 26

Reference Drawing 495848

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK01-21
Rev. 13
Page 1 of 6

**CONTMT
FAN CLRS**

03/07/06

Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
149	Contmt Fan Clr 1-1 OC Trip	49X-1F-2	Hi 447 amps Lo 267 amps	2.2
150	Contmt Fan Clr 1-2 OC Trip	49X-1F-1	Hi 447 amps Lo 267 amps	2.2
228	Contmt Fan Clr 1-3 OC Trip	49X-1G-1	Hi 447 amps Lo 267 amps	2.2
229	Contmt Fan Clr 1-5 OC Trip	49X-1G-2	Hi 447 amps Lo 267 amps	2.2
300	Contmt Fan Clr 1-4 OC Trip	49X-1H-1	Hi 447 amps Lo 267 amps	2.2
433	Contmt Fan Clr 1-2 Vib Hi	YS10 YS16	> 3 in/sec G > 150% NORM	2.3
434	Contmt Fan Clr 1-3 Vib Hi	YS11 YS17	> 3 in/sec G > 150% NORM	2.3
435	Contmt Fan Clr 1-4 Vib Hi	YS12 YS18	> 3 in/sec G > 150% NORM	2.3
436	Contmt Fan Clr 1-5 Vib Hi	YS13 YS19	> 3 in/sec G > 150% NORM	2.3

(Continued)

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
441	Contmt Fans Temp PPC	T4007C		2.1
	<u>INPUTS TO T4007C</u>	<u>PPC ID</u>		
	Contmt Fan Mtr 1-1 Inbd Brg Temp	T1065A		
	Contmt Fan Mtr 1-1 0-Bd Brg Temp	T1066A		
	Contmt Fan 1-1 Stator Temp	T1067A	≥ 219°F (see NOTE)	
	Contmt Fan 1-1 Inbd Brg Temp	T1068A		
	Contmt Fan 1-1 Outbd Brg Temp	T1069A		
	Contmt Fan Mtr 1-2 Inbd Brg Temp	T1070A		
	Contmt Fan Mtr 1-2 0-Bd Brg Temp	T1071A		
	Contmt Fan 1-2 Stator Temp	T1072A	≥ 219°F (see NOTE)	
	Contmt Fan 1-2 Inbd Brg Temp	T1073A		
	Contmt Fan 1-2 Outbd Brg Temp	T1074A		
	Contmt Fan Mtr 1-3 Inbd Brg Temp	T1075A		
	Contmt Fan Mtr 1-3 0-Bd Brg Temp	T1076A		
	Contmt Fan 1-3 Stator Temp	T1077A	≥ 219°F (see NOTE)	
	Contmt Fan 1-3 Inbd Brg Temp	T1078A		
	Contmt Fan 1-3 Outbd Brg Temp	T1079A		
	Contmt Fan Mtr 1-4 Inbd Brg Temp	T1080A		
	Contmt Fan Mtr 1-4 0-Bd Brg Temp	T1081A		
	Contmt Fan 1-4 Stator Temp	T1082A	≥ 219°F (see NOTE)	
	Contmt Fan 1-4 Inbd Brg Temp	T1083A		
	Contmt Fan 1-4 Outbd Brg Temp	T1084A		
	Contmt Fan Mtr 1-5 Inbd Brg Temp	T1085A		
	Contmt Fan Mtr 1-5 0-Bd Brg Temp	T1086A		
	Contmt Fan 1-5 Stator Temp	T1087A	≥ 219°F (see NOTE)	
	Contmt Fan 1-5 Inbd Brg Temp	T1088A		
	Contmt Fan 1-5 Outbd Brg Temp	T1089A		

(Continued)

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
442	Contmt Fan Clr 1-1 Vib Hi	YS9	> 3 in/sec G	2.3
		YS15	> 150% NORM	

NOTE: On the PPC only, an additional alarm is received when this parameter reaches 293°F.

2. OPERATOR ACTIONS

2.1 Fan High Temperature (Input 441)

2.1.1 Check annunciator and PPC printouts to determine affected CFCU. []

2.1.2 REFER TO STP M-51, "Routine Surveillance Test of Containment Fan Cooler Units," for acceptable CFCU CCW flows, and check that CCW flow to the affected CFCU is adequate: []

- CFCU 1-1: FI-34
- CFCU 1-2: FI-35
- CFCU 1-3: FI-36
- CFCU 1-4: FI-37
- CFCU 1-5: FI-38

2.1.3 IF Containment ambient air temperature is 120°F or above, THEN REFER TO TS 3.6.5, "Containment Air Temperature". []

NOTE: A CFCU can be run up to the temperatures listed below without affecting component lifetime:

- CFCU motor stator: 219°F
- CFCU motor inboard bearing: slow speed 180°F, fast speed 206°F
- CFCU motor outboard bearing: slow speed 180°F, fast speed 206°F
- CFCU fan inboard bearing: 180°F
- CFCU fan outboard bearing: 180°F

2.1.4 IF any CFCU is left in service with actual high temperature alarms, THEN document the following in an Action Request: []

- The time period alarm limits were exceeded
- The maximum temperature reached

2.1.5 IF CFCU stator temperature is 293°F or above, THEN contact engineering to evaluate securing the CFCU. []

2.1.6 IF necessary to stop or start CFCUs, THEN REFER TO the appropriate OP H-2 series procedure(s). []

2.1.7 Probable Causes

- High Containment ambient temperature
- Low CCW flow to CFCU fan motor
- CFCU fan motor stator temperature out-of-limit
- CFCU fan or fan motor bearing temperature out-of-limit

2.2 Fan Cooler Overcurrent Trip (Inputs 149, 150, 228, 229, 300)

2.2.1 Check annunciator and PPC printouts to determine affected fan cooler. []

NOTE: CFCUs are normally in service sufficient to maintain Containment environment below 120°F.

2.2.2 IF needed to maintain acceptable Containment environment, THEN place another available CFCU in service PER OP H-2:I, "Containment Fan Cooler Units-Make Available and System Operation." []

CAUTION: To preclude further damage or creation of an electrical safety hazard, do NOT restart the tripped CFCU until the trouble has been investigated by maintenance.

2.2.3 Before attempting a restart of the tripped CFCU, direct maintenance to investigate. []

2.2.4 REFER TO TS 3.6.6, "Containment Spray and Cooling Systems". []

2.2.5 Probable Causes

- Motor overload sufficient to heat thermal device to opening temperature
- Fault in motor or on feeder cable causing instantaneous overload trip

2.3 Fan Cooler High Vibration (Inputs 433, 434, 435, 436, 442)

2.3.1 Check annunciator printout to determine affected fan cooler. []

NOTE: A CFCU vibration alarm can be expected when starting a CFCU or shifting a running CFCU to high speed.

2.3.2 Reset the vibration alarm on the Main Control Board. []

- 2.3.3 IF the annunciator comes ON again due to high vibration,
THEN perform the following:
- a. Press the "STOP" button on the Main Control Board to shut down the affected CFCU. []
 - b. Start another available CFCU PER OP H-2:I, Section 6.2, "Starting a CFCU." []
 - c. Direct maintenance to investigate. []
 - d. REFER TO TS 3.6.6, "Containment Spray and Cooling Systems". []

- 2.3.4 Probable Causes
- Imbalance in CFCU motor or fan
 - Starting a CFCU
 - Shifting a running CFCU to high speed

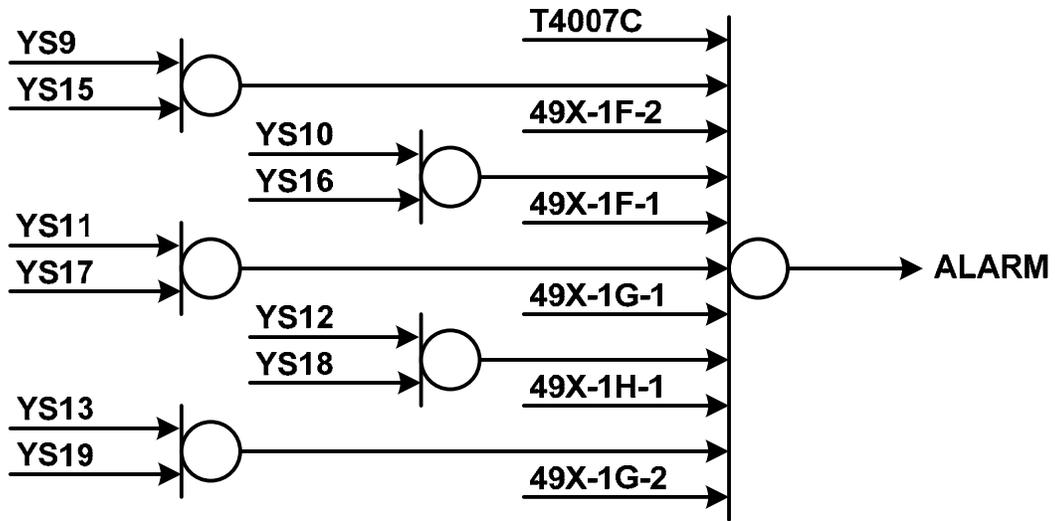
3. AUTOMATIC ACTIONS

NONE

4. REFERENCES

- 4.1 OP H-2:II, "Containment Fan Coolers Shutdown and Clearing"
- 4.2 106714, "OVID - Component Cooling Water System"
- 4.3 437600, "Schematic Diagram Containment Fan Coolers"
- 4.4 501121, "Electrical Schematic Diagram - Main Annunciator" (Inputs 228, 229, 300, 442) (Electrical Drawing Section 8)
- 4.5 501122, "Electrical Schematic Diagram - Main Annunciator" (Inputs 149, 150, 433, 434, 435, 436, 441) (Electrical Drawing Section 8)

5. LOGIC DIAGRAM



PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER AR PK05-23
REVISION 17
PAGE 3 OF 3
UNIT 1

TITLE: PZR SAFETY OR RELIEF LINE TEMP

5.8 Safety Valve Loop Seal Temp HI/LO Alarm Input 0008

5.8.1 Display PPC group "PK05-23" to determine which loop seal temperature is in alarm and that the RTD's in each loop channel check appropriately.

- a. If a loop seal temperature exceeds 500°F, declare the corresponding safety valve INOPERABLE per TS 3.4.10.
- b. If a safety valve is declared inoperable due to exceeding 500°F, the valve can be declared operable when both of the following criteria are met (Reference AR# A0543295):
 1. Loop seal temperature has decreased to less than 500°F.
 2. There is a clear trend that loop seal temperature is continuing to decrease.

NOTE: The 320°F alarm setpoint is a nominal value, and may range between 320°F and 350°F for an individual loop, depending on specific circumstances.

- c. If a loop seal temperature exceeds 320°F check tailpipe temperature for indication of seat leakage, and contact system engineer.

NOTE: The 235°F alarm setpoint is a nominal value, and may range between 225°F and 235°F for an individual loop, depending on specific circumstances.

- d. If a loop seal temperature is less than the 235°F low alert setpoint while the plant is at NOP/NOT, contact the system engineer to evaluate. Otherwise, during a plant heatup or cooldown perform OP A-4A:IV to restore loop seal temperatures by draining.
- e. If a loop seal temperature is less than the 221°F low alarm setpoint, declare all safety valves INOPERABLE per TS 3.4.10.

5.9 Refer to ECG 7.8 for instrumentation failures.

#

1.00

Which valve from the list below is interlocked with the residual heat removal (RHR) to containment spray rings isolation valve (9003A)?

- A. Containment recirculation sump suction valve (8982A).
- B. RHR cold leg injection isolation valve (8809A).
- C. Containment spray pump discharge valve (9001A).
- D. RHR to containment spray rings isolation valve (9003B).

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

68489	Describe Containment Spray System automatic features.
37578	Analyze the control logic for Containment Spray system equipment, including: ? Pumps ? Valves

Reference Id: P-0308
Must appear: No
Status: Active
User Text: 6043.060264
User Number 1: 0000003.60
User Number 2: 0000003.40
Difficulty: 1.00

RHR to Containment Spray Valves (CS-9003A/B), Continued

Controls

Valves CS-9003A/B have control capability from the

- Control Room on VB1 and
- local handwheel.

Control at VB1	Operation
CLOSE/STOP/OPEN	3 position, maintain position.

Logic

Obj 9

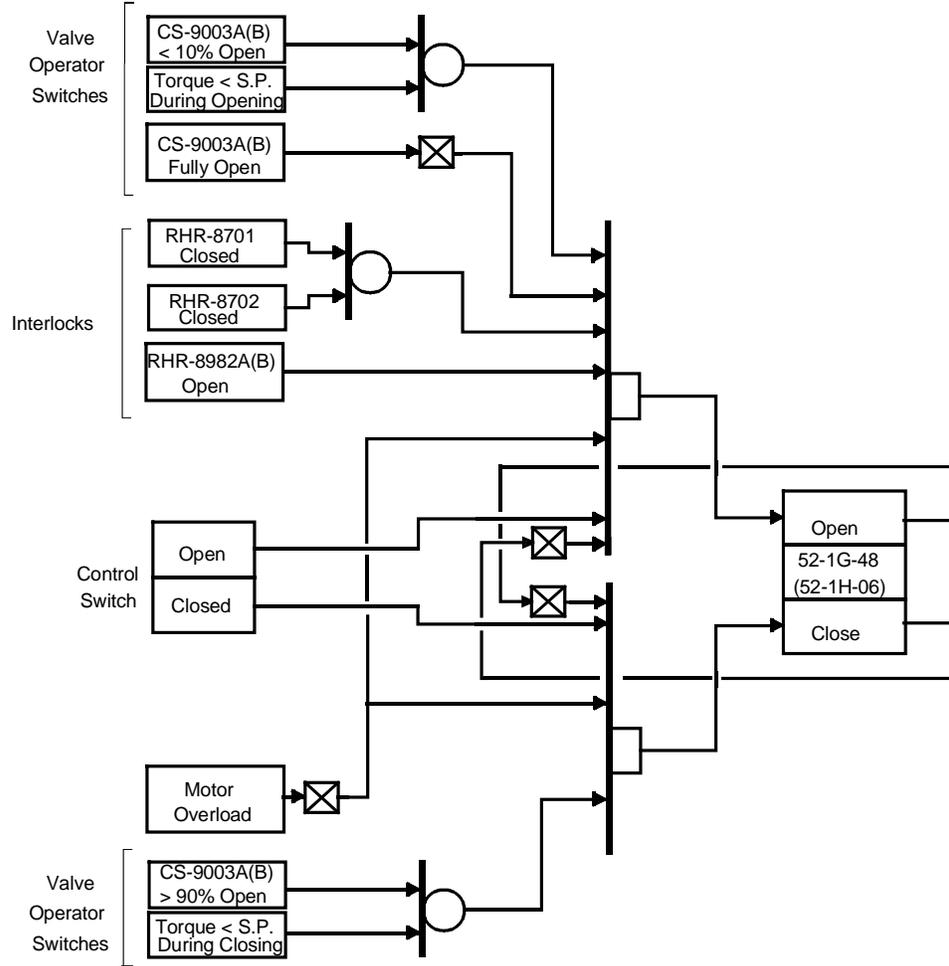
The logic associated with valves CS-9003A/B is described in the following tables

If the VB1 control switch is in ...	Then valve CS-9003A will ...
CLOSE	Close
OPEN	Open IF both <ul style="list-style-type: none"> • RHR-8982A, RHR pump 1 suction from Containment sump is open AND • RHR-8701 OR -8702, RHR suction from RCS loop 4 is closed
If the VB1 control switch is in ...	Then valve CS-9003B will ...
CLOSE	Close
OPEN	Open IF both <ul style="list-style-type: none"> • RHR-8982B, RHR pump 2 suction from Containment sump is open AND • RHR-8701 OR -8702, RHR suction from RCS loop 4 is closed

Continued on next page

RHR to Containment Spray Valves (CS-9003A/B), Continued

Logic Diagram This is a logic diagram of the controls for CS-9003A, CS-9003B is similar.
Obj 9



CSS-13

Continued on next page

Phase B CI and Containment Spray Actuation, Continued

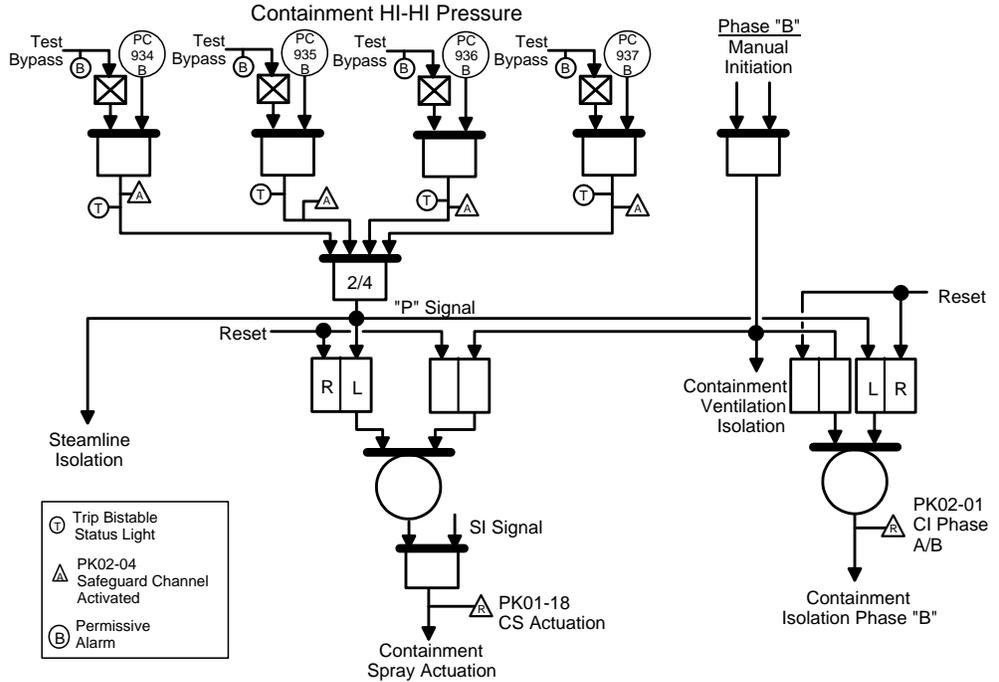
Logic (continued)

IF	AND	Then
Phase B CI Train A (B) signal is RESET		the Train A (B) Phase B CI signal is reset. <ul style="list-style-type: none"> • the manual actuation signal goes through a retentive memory unit and will seal in until reset, once reset, manual actuation of Phase B CI is still available. • the auto actuation signal goes through a retentive memory with manual reset unit and will also seal in until reset. <ul style="list-style-type: none"> • the signal can be reset even though the initiating signal is still present. • automatic actuation is blocked until the Containment High High Pressure signal is cleared. • PK02-01, CI Phase A/B will clear. • Red light for Train A (B) above Monitor Light Box D goes out.
CS Train A (B) signal is RESET		the Train A (B) CS signal is reset. <ul style="list-style-type: none"> • the manual actuation signal goes through a retentive memory unit and will seal in until reset, once reset, manual actuation of CS is still available. • the auto actuation signal goes through a retentive memory with manual reset unit and will also seal in until reset. <ul style="list-style-type: none"> • the signal can be reset even though the initiating signal is still present. • automatic actuation is blocked until the Containment High High Pressure signal is cleared. • PK01-18, CS Actuation will clear.

Continued on next page

Phase B CI and Containment Spray Actuation, Continued

Logic diagram The logic diagram for the Phase B CI and CS Actuation signal is shown below. Only one SSPS Train is shown, the other is identical.
Obj 11



RPS-33

Indications The following indications are available for the Phase B CI CS Actuation signal.
Obj 5

VB1 Bistable Status Panel		
Indicating Light	Meaning	Normal Status
Red	Hi Cntnmt Press bistable PC934B tripped	OFF
White	Hi Cntnmt Press bistable PC935B tripped	
Blue	Hi Cntnmt Press bistable PC936B tripped	
Yellow	Hi Cntnmt Press bistable PC937B tripped	

Continued on next page

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
OPERATING PROCEDURE

NUMBER OP O-4
REVISION 17A
PAGE 1 OF 11
UNITS

TITLE: Primary to Secondary Steam Generator Tube Leak
Detection

1 AND 2

12/07/06

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

- 1.1 This Operating Order sets forth the policies and procedures that are to be followed to provide early detection of a leaking Steam Generator tube.
- 1.2 This Operating Order provides direction for the on-shift crew on the required actions to be taken if the indicated Steam Generator tube leakage increases.

2. DISCUSSION

- 2.1 The response to Steam Generator Tube Leakage within this procedure is divided into four major categories. The section used is dependent upon the amount of primary to secondary leakage per Steam Generator, estimated from radiation monitor readings. Each category gives specific guidance on operator actions to mitigate the effects of the primary to secondary leakage. The six major categories are:
- 2.1.1 Normal Operation - Less than detectable OR < 5 gallons per day (gpd).
- 2.1.2 Increased Monitoring - ≥ 5 gpd and < 30 gpd.
- 2.1.3 Action Level 1 - ≥ 30 gpd and < 75 gpd (0.05 gpm).
- 2.1.4 Action Level 2 - ≥ 75 gpd - sustained ≥ 1 hr. **WITH** a leak rate increase < 30 gpd/hr.
- 2.1.5 Action Level 3a - ≥ 75 gpd - **WITH** a leak rate increase ≥ 30 gpd/hr.
- 2.1.6 Action Level 3b - ≥ 150 gpd. (0.1 gpm)
- 2.2 The rate of increase associated with Action Levels 2 and 3a (see step 2.1) can be evaluated as leak rate changes over time intervals not exceeding 30 minutes (e.g., leak rate has not increased ≥ 15 gpd in a 30 minute period). Alternately, the PPC rate of increase calculation may be used

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TITLE: Primary to Secondary Steam Generator Tube Leak
 Detection

2.3 The Air Ejector Off-gas monitors, RM-15 and 15R, are the most sensitive indicators of small changes in Steam Generator tube leakage. The Plant Process Computer (PPC) receives an input from RM-15 and 15R. The Plot or Archive function of the PPC can be used to detect small changes in the Air Ejector Off-gas count rate.

2.3.1 The High Alarm set point of RM-15 and 15R is set to the activity level that should be seen by the detector if the previously calculated RCS activity still exists and a total of 30 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the High Alarm setpoint may correspond to a leak rate greater than 30 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

2.3.2 The Alert alarm set point of RM-15 and 15R is set to the activity level that would be seen by the detector if the previously calculated RCS activity still exists and a total of 20 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the Alert Alarm setpoint may correspond to a leak rate greater than 20 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

2.3.3 Because the established monitor High Alarm and Alert Alarm setpoints are fixed values, they may correspond to a leak rate value that is different than 30 gpd or 20 gpd, depending upon changes in assumptions used to calculate the setpoints. The table below describes how some of these changes affect the setpoint:

Parameter	Relative Parameter Change	Alarm Setpoint Affect
RCS Noble Gas Concentration	↑	More Conservative
	↓	Less Conservative
SJAЕ Exhaust Flowrate	↑	Less Conservative
	↓	More Conservative

2.3.4 Primary to secondary leak rates can be inferred by comparing existing RM-15, 15R readings to the Alarm and Alert set points. The relationship of leak rate to activity level is linear as indicated activity increases.

2.3.5 Chemistry projects what the RM-15(R) count rate will be for a Steam Generator Tube leak of 30 gpd, 75 gpd and 150 gpd. This information is provided for reference in accordance with CY1.DC3. This projected value can be used by the Shift Foreman to roughly estimate Steam Generator leak rates.

2.3.6 Frequent analysis of RM-15(R) count rate and evaluation of the change in count rate with time is the best indicator of changes in the Steam Generator leak rate.

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 Detection

- 2.4 The PPC monitoring of RM-15(R) also provides an alarm and several warnings if Air Ejector Off-gas count rate increases. These warnings/alarms are:
 - 2.4.1 HALM - Provides an alarm at the same time the Main Annunciator PK11-06, alarm input 423 does. Refer to I&C Rad Monitor Data Book for set point.
 - 2.4.2 HWRN - Provides the first warning if RM count rate increases from the normal. Refer to I&C Rad Monitor Data Book for set point.
 - 2.4.3 Significant Change Alarming - The HWRN alarm will reflash each time the value increases by a third of the difference between the HWRN and HALM alarm set points. (i.e., if the HWRN and HALM set points are 250 cpm and 1000 cpm, the HWRN will reflash at 500 cpm and again at 750 cpm.)
 - 2.4.4 PPC points available for monitoring (GRPDIS OP O-4):
 - a. RM-15 R0015A
 - b. RM-15R R0115A
 - c. RM-19 R0019A
 - d. FIT-81 F0704A or U5515
 - e. Leak rate based on RM-15 U5503
 - f. Leak rate based on RM-15R U5504
 - g. Rate of increase based on RM-15 U5507
 - h. Rate of increase based on RM-15R U5508
 - 2.4.5 Wide-range SJAE FIT-81 live data is used in the PPC S/G tube leakage calculation when FIT-81 is OPERABLE (KPSLK is set to zero, and F0704A is used). However, PPC monitoring may still be credited when FIT-81 is out-of-service. In this case, KPSLK is set to 1, and the PPC substitutes an existing SJAE off gas flow constant (KSJAEFLW, based on previous chemistry data) into the calculation (U5515). This is the "pre-planned alternate method of determining SJAE flow rate" described in ECG 2.1, action A.1.1.2 bases.
- 2.5 Confirmed spiking on RM-15 and 15R that is increasing in frequency and/or amplitude is a precursor to SG tube failure. These symptoms were observed at Comanche Peak Unit One prior to their tube leak/failure (LER 445-02002).

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TITLE: Primary to Secondary Steam Generator Tube Leak
 Detection

3. RESPONSIBILITIES

- 3.1 Shift Foreman (SFM) is responsible for evaluating the RM-15(R) count rate increases and grab sample results for Steam Generator leakage changes.
- 3.2 Chemistry is responsible to provide RCS activity levels to the SFM so that RM-15(R) readings can be equated to leak rates between the primary and secondary systems. Immediate notification of the Control Room should be made if indications by grab samples indicate a leak rate ≥ 5 gpd.
- 3.3 Normally, the PPC will monitor and alarm steam generator tube leakage. If PPC monitoring is out of service, the Control Operator or the BOP Control Operator shall compare the changes in RM-15(R) count rate at least every 4 hours per step 6.1.3. As manpower needs dictate, utilize the Shift Engineer to perform manual monitoring, especially when 15-minute intervals are required.

4. PREREQUISITES

- 4.1 Air Ejector Off-gas monitors, RM-15 and RM-15R shall have a High Warning alarm set to a value high enough above normal reading, but low enough to give an early warning of changes in Air Ejector Off-gas count rate.
- 4.2 PPC address U5503 and U5504, Primary-to-Secondary leak rate (based on RM-15(R) and Air Ejector Flowrate) should have a High Warning alarm set to a value that gives an early warning of a significant change in leak rate.
- 4.3 PPC address U5507 and U5508, Primary-to-Secondary leak rate of increase (based upon an average of recent leak rate calculations) should have a High Warning alarm set to a value that gives an early warning of a significant indication of leak rate change (gpd/hr).
- 4.4 The Significant Change Alarming function for Modes 1, 2 and 3 shall be enabled.

5. PRECAUTIONS

None

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TITLE: Primary to Secondary Steam Generator Tube Leak
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6. INSTRUCTIONS

NOTE: RM-15 and RM-15R are identical and redundant monitors. Either one can be used to meet the intent of this procedure. RM-15 is the preferred data point for consistent log data.

6.1 NORMAL OPERATION - Less than detectable OR < 5 gpd per S/G

6.1.1 PPC monitoring of S/G tube leakage:

- a. Initiate PPC monitoring (GRPDIS OP O-4).
- b. Verify RM-15 or RM-15(R), Condenser Air Ejector Radiation Monitors, and Plant Process Computer, PPC, are in service and OPERABLE, and PPC quality codes are NOT light blue (cyan).
- c. Verify FIT-81, SJAE Off-gas Flow monitor status (ECG 2.1).
 - 1. If FIT-81 is OPERABLE and in service, verify PPC constant KPSLK = 0.
 - 2. If FIT-81 is NOT OPERABLE, verify PPC constant KPSLK = 1 in accordance with OP O-15, "Control of PPC Addressable Points".
- d. Check R0015A (R0115A) HWRN or HALM not in alarm status.
- e. Check leak rate U5503 (U5504) HWRN or HALM not in alarm status.
- f. Check rate of increase, U5507 (U5508) HWRN or HALM not in alarm status.
- g. Inform the SFM if any S/G tube leakage monitoring points are in alarm.

6.1.2 Perform a qualitative evaluation of Air Ejector radiation monitor count rate by viewing a continuous trend of the monitor output. Look for any upward trends. The evaluation of the count rate may be made using any one of the following methods:

- a. Maintaining a continuous plot of R0015A or R0115A on any Control Room PPC CRT with a time scale of 270 minutes.
- b. Maintaining a continuous trace of R0015A or R0115A on Control Room analog trend recorder.
- c. Running an Archival Plot of R0015A or R0115A and another associated PPC radiation monitor input (R0019A or R0023R is suggested) every four hours.

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- 6.1.3 Manual monitoring of S/G tube leakage (PPC monitoring out of service).
 - a. Record RM-15(R) count rate, Air Ejector Off-gas flow rate, and the results of the evaluation on Attachment 9.1 every four hours.
 - b. Evaluate the RM-15(R) count rate at least once every four hours for an increase in the instantaneous value and/or a slow increase of count or leak rates over the previous four hours.
 - c. Enter UNSAT for the evaluation if any of the following conditions exist:
 - 1. RM-15(R) or Air Ejector grab sample count rates increase by a factor of 2 or greater over the previous 4 hour period.
 - 2. RM-15(R) count rates indicate that primary to secondary leakrate has increased to 30 gpd, or has increased more than 20 gpd in the last 4 hours.
 - d. Inform the SFM if the results of the evaluation show an increasing count rate or rate of increase.
- 6.1.4 Normal Operation with RM-15 and RM-15R Out Of Service
 - a. Inform the shift C&RP technician to increase Air Ejector grab sampling per OM12.DC1.
 - b. IF the leak rate was unstable or on an increasing trend when RM-15 and RM-15R were both removed from service, THEN perform the following:
 - 1. Enter Action Level 2, paragraph 6.4
 - 2. Inform the shift C&RP technician to increase Air Ejector grab sampling to every 4 hours.
 - c. IF in Action Level 2 when RM-15 and RM-15R are both removed from service, THEN enter Action Level 3b, paragraph 6.6.
 - d. Elevate restoration of RM-15(R) to the highest priority.

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- 6.2 Increased Monitoring - ≥ 5 gpd and < 30 gpd
- 6.2.1 Normally, this condition would be identified by chemistry's routine grab samples. Any indication of leakage ≥ 5 gpd should be immediately reported to the Control Room.
- 6.2.2 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to FIRST identify the source and THEN quantify and amount of primary to secondary leakage per chemistry procedure CAP D-15.
- a. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.
- 6.2.3 If any Radiation Monitors for monitoring the secondary system per step 6.1 are out of service, elevate their restoration to the highest priority. Manual monitoring per steps 6.1.3 or 6.1.4 may be required.
- NOTE:** The following will provide a warning of additional increases in leakrate.
- 6.2.4 If any secondary system radiation monitors alarm, request that chemistry initiate a re-calculation and resetting of the setpoints above their current levels, but no higher than 30 gpd per CYC2.DC1, "Radiation Monitoring High Alarm Setpoint Control Program."
- 6.2.5 Tailboard OP AP-3.
- 6.3 Action Level 1 - ≥ 30 gpd and < 75 gpd.
- 6.3.1 If a HWRN alarm is received and evaluated as NOT having been caused by a transient noise spike on the radiation monitor, commence increased monitoring per step 6.3.3.
- 6.3.2 If RM-15(R) or Air Ejector grab sample indicate a substantial increase (greater than a factor of 2 over the previous 4 hour average) **OR** RM-15(R) is above the Alert set point, commence increased surveillance per step 6.3.3.
- 6.3.3 With indications of increasing primary to secondary leak rate as in step 6.3.1 or 6.3.2 above, begin increased surveillance as follows:
- a. Increase monitoring of the RM-15(R) count rate to continuous by dedicating a Control Room PPC CRT to the trending of the RM-15 (R0015A) and/or RM-15R (R0115A) count rate(s), S/G leak rate calculations (U5503 or U5504) and rate of increase (U5507 or U5508).
1. If PPC is in service, visual monitoring of these points and associated alarms should be made approximately once every 15 minutes.
 2. If PPC is out of service, manual monitoring, per step 6.1.3, should be completed every 15 minutes.
 3. If the PPC, RM15 and RM-15R are all out of service, comply with step 6.1.4.

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TITLE: Primary to Secondary Steam Generator Tube Leak
 Detection

- b. Direct the shift C&RP technician to attempt to FIRST identify the source and THEN quantify and amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.
 - 1. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.
 - c. Direct the shift C&RP technician to increase SJAE sampling in accordance with OM12.DC1 until otherwise directed by the Shift Foreman.
- 6.3.4 Contact chemistry for an evaluation of the potential for contamination of the secondary system.
- 6.3.5 Once the leak rate is stable for 1 hour ($\leq 10\%$ increase during a one-hour period), monitoring of RMS responses can be reduced to once every 2 hours. Once leak rates are stable for 24 hours ($\leq 10\%$ increase during 24 hours), monitoring frequencies can return to those required by step 6.2.
- a. Evaluate resetting RM-15(R) HWRN and HALM setpoints to above their existing baseline reading, but not over equivalent 75 gpd.

NOTE: The following will provide a warning of additional increases in leakrate.

- 6.3.6 Once leak rate is stable for 1 hour, request that chemistry initiate a re-calculation and reset RM-15(R) HWRN and HALM setpoints above their current levels, but no higher than 75 gpd per CYC2.DC1, "Radiation Monitoring High Alarm Setpoint Control Program."
- 6.3.7 Tailboard OP AP-3 if not done previously.

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- 6.4 Action Level 2 - ≥ 75 gpd (0.05 gpm) sustained ≥ 1 hr. **WITH** a leak rate increase < 30 gpd/hr
- 6.4.1 Verify PPC is monitoring leak rate (U5503 or U5504) and rate of increase (U5507 or U5508) every 15 minutes. See step 6.1.1 for details.
- a. If rate of increase is < 30 gpd/hr, continue monitoring.
 - b. If rate of increase is ≥ 30 gpd/hr, go to Action Level 3a.
- 6.4.2 If PPC monitoring is out of service and either RM-15 or RM-15R are available, determine leak rate and rate of increase every 15 minutes on Attachment 9.1
- a. (Leak rate increase over a 30 minute period) $\times 2 =$ gpd/hr.
 - b. If < 30 gpd/hr, continue monitoring.
 - c. If ≥ 30 gpd/hr, go to Action Level 3a.
- 6.4.3 If the PPC, RM15 and RM-15R are all out of service, comply with step 6.1.4.
- 6.4.4 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to **FIRST** identify the source, and **THEN** quantify the amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.
- a. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.
- 6.4.5 IF confirmed leak rate is ≥ 75 gpd,
AND rate of increase is < 30 gpd/hr,
THEN reduce power to take the unit off-line and be in Mode 3 within 24 hours.
- a. A qualitative confirmation of the leak rate can be made by channel checking the RM-15(R) count rates against an upscale indication on either blowdown or steamline radiation monitors.
- 6.4.6 Contact chemistry and RP for an evaluation of the potential for contamination of the secondary system.
- 6.4.7 Tailboard OP AP-3 if not done previously.

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NOTE: The 30 gpd/hr rate of increase should be sustained for a minimum of 30 minutes before taking action to initiate shutdown per Action Level 3a.

- 6.5 Action Level 3a - ≥ 75 gpd (0.05 gpm) **AND** ≥ 30 gpd/hr
- 6.5.1 IF confirmed leak rate is ≥ 75 gpd,
AND a continued rate of increase of ≥ 30 gpd/hr measured over a minimum of 30 minutes, (leak rate increase over a 30 minute period $\times 2 =$ gpd/hr),
THEN reduce to $\leq 50\%$ power within 1 hour and be in Mode 3 within the next 2 hours.
- A qualitative confirmation of the leak rate can be made by channel checking the RM-15(R) count rates against an upscale indication on either blowdown or steamline radiation monitors.
 - See step 6.1.1 for PPC monitoring.
 - See step 6.4.2 for manual monitoring.
- 6.5.2 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to FIRST identify the source, and THEN quantify the amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.
- Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.
- 6.5.3 Contact chemistry and RP for an evaluation of the potential for contamination of the secondary system.
- 6.6 Action Level 3b - ≥ 150 gpd (0.1 gpm). This limit includes leak rate spikes.
- 6.6.1 IF confirmed leak rate is ≥ 150 gpd,
THEN be in Mode 3 within 6 hours.
- A qualitative confirmation of the leak rate can be made by channel checking the RM-15(R) count rates against an upscale indication on either blowdown or steamline radiation monitors.
 - If shutting down per Action Level 3a, do not relax the Mode 3 time limit if leak rate is ≥ 150 gpd.
 - See step 6.1.1 for PPC monitoring.
 - See step 6.4.2 for manual monitoring.
- 6.6.2 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to FIRST identify the source, and THEN quantify the amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.
- Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.
- 6.6.3 Contact chemistry and RP for an evaluation of the potential for contamination of the secondary system.
- 6.6.4 Tailboard OP AP-3 if not done previously.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP O-4
REVISION 17A
PAGE 11 OF 11
UNITS 1 AND 2

TITLE: Primary to Secondary Steam Generator Tube Leak
Detection

7. REFERENCES

- 7.1 Enclosure to PG&E Letter No. DCL-88-064
- 7.2 NRC Bulletin 88-02
- 7.3 NRC Information Notice 88-99
- 7.4 NRC Information Notice 91-43
- 7.5 EPRI Guideline 1008219 Final Report, December 2004, "PWR Primary-to-Secondary Leak Guidelines - Revision 3"
- 7.6 OM12.DC1, "Relieving the Watch"

8. RECORDS

- 8.1 Route the Data Sheet to the Shift Foreman for evaluation
- 8.2 Route completed Data Sheets to chemistry

9. ATTACHMENTS

- 9.1 "RM-15(R) Count Rate Evaluation Data Sheet," 05/08/04

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK06-06
Rev. 1
Page 1 of 2

**RCS - S/G
PRESS
MISMATCH**

02/01/06

Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
863	RCS S/G Pressure Mismatch Pre S.I. Warning	Y1000C	RCS pressure > 1850 psig <u>AND</u> SG pressure < 630 psig	2.1

2. OPERATOR ACTIONS

2.1 General Actions

NOTE: This alarm indicates that Low Steam Line Pressure Safety Injection setpoint is being approached.

2.1.1 IF pressurizing the RCS,
THEN perform the following:

- a. STOP the RCS pressurization. []
- b. Heat up the RCS until SG pressures are above 630 psig PER OP L-1, "Plant Heatup From Cold Shutdown to Hot Standby". []
- c. RETURN TO procedure and step in affect. []

CAUTION: Prior to reducing pressure below P-11 (1915 psig) and blocking the Low Streamline Pressure SI signal, the RCS shall be borated per STP R-19 to cold shutdown conditions to avoid an unanalyzed condition during a main steam line rupture.^{Ref 4.2}

- 2.1.2 IF cooling down the RCS,
THEN perform the following:
- a. STOP the RCS cooldown. []
 - b. Before continuing, verify RCS borated to cold shutdown conditions per STP R-19, "Shutdown Margin Determination".^{Ref 4.2} []
 - c. BLOCK Low Steam Line Pressure Safety Injection signal (CC2). []
 - d. Before continuing, depressurize the RCS to less than 1850 psig PER OP L-5, "Plant Cooldown From Minimum Load to Cold Shutdown". []
 - e. RETURN TO procedure and step in affect. []

- 2.1.3 Probable Causes
- Pressurizing the RCS without heating up the RCS
 - Cooling down the RCS without depressurizing the RCS

3. AUTOMATIC ACTIONS

3.1 Possible Safety Injection if main steam line pressure is 600 psig or below, and RCS pressure is 1915 psig or above

4. REFERENCES

- 4.1 501135, "Electrical Schematic Diagram - Main Annunciator" (Electrical Drawing Section 8)
- 4.2 A0564034, "Assessment Of Callaway Plant Event Report 39142"

5. LOGIC DIAGRAM



Lesson

Main Feedwater System

Normal operation

During normal plant operation at 100% power, both Main Feedwater pumps are in operation.

- Each MFP is capable of approximately 60% flow.

Loss of MFP transient
Obj 1

The plant transient initially seen on a loss of a MFP is shown in the following table.

Component/ Parameter	Reaction
Reactor power	Reactor power decreases slightly due to decreased secondary heat transfer as seen by the following RCS indications: <ul style="list-style-type: none"> • T_{ave} increases • Pzr level increases • Loop ΔT decreases
Turbine load	Turbine load increases due to increased steam flow through LP turbines.
S/G	S/G inventory density decreases due to heatup. <ul style="list-style-type: none"> • The swell masks the inventory loss due the remaining feedwater pump not being able to supply enough flow for present steam flow.
Plant trip	With no operator action the plant will trip on low S/G level in approximate 45 seconds from 100% power.

Special considerations

While performing AP-15 to recover from the loss of a feedwater pump transient, some of the special considerations the operator should be aware of are:

- Rapidly reducing turbine load causes S/G level to shrink.
- Maintaining S/G pressure below the 10% atmospheric dump valve setpoint is desired.
- Very rapid operator action is required to prevent a reactor trip.

STG C8B, page 1-3

Basic Description

Purpose of the DFWCS
Obj 1

The purpose of the Digital Feedwater Control System (DFWCS) is to automatically maintain a programmed steam generator water level during steady-state operation and to restore and maintain the water level within acceptable levels during normal unit transients

Basic description

The DFWCS:

- uses a state of the art, microprocessor based system design,
- houses hardware in two cabinets, located behind the vertical boards in the control room,
 - circuitry dedicated to loops one and two and main feedwater pump speed control is housed in cabinet one,
 - circuitry dedicated to loops three and four is housed in cabinet two.

The DFWCS controls main feedwater flow by sending positioning signals to:

- main feed regulating (reg) valves,
- main feed reg bypass valves, and
- main feed pump speed control,
 - to maintain a programmed differential pressure between the feed and steam pressures, and
 - maintain the programmed level in the steam generators.

The control system derives an error signal for the controlling elements based on inputs from key parameters of the Steam Generator System, Main Steam Supply System, Main Turbine and Main Feedwater System.

Continued on next page

N-73157

The Digital Feedwater Control System controls feedwater flow to maintain:

- A. programmed level in the Steam Generator.
- B. constant mass in the Steam Generator.
- C. constant level in the Steam Generator.
- D. programmed pressure in the Steam Generator.

Answer: A

ASSOCIATED INFORMATION:

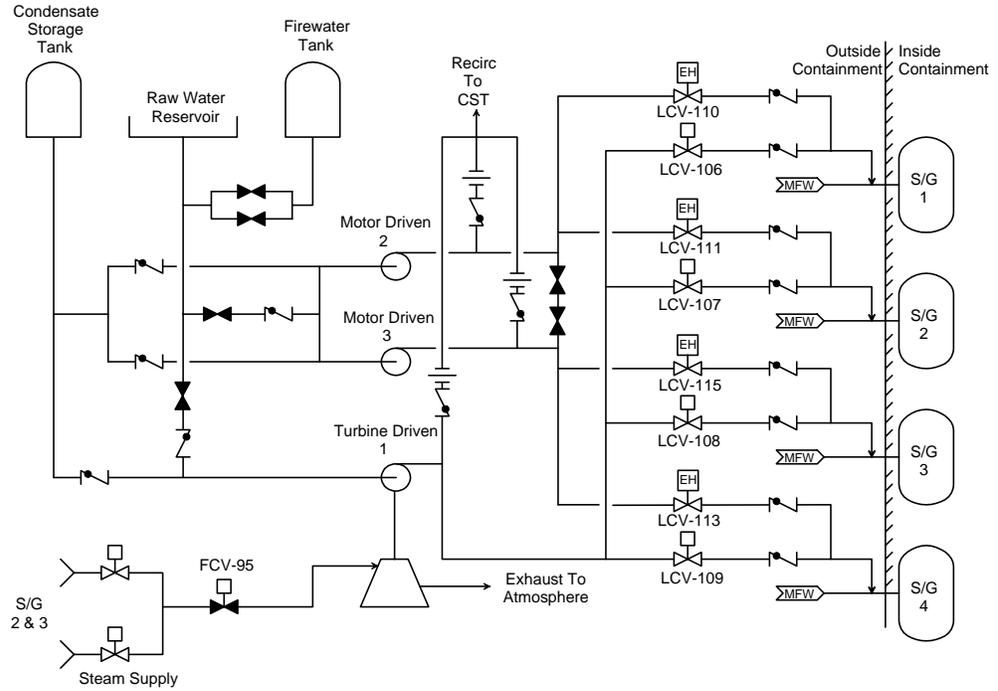
Associated objective(s):

70366	State the purpose of the Main Feedwater Control System
-------	--------------------------------------------------------

Reference Id: N-73157
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 1.00

Basic Description, Continued

Basic flowpath The basic block and flow diagram of the AFW system is shown here.
Obj 2



D-1-1

TH11

System Coefficient Effects (Open System)

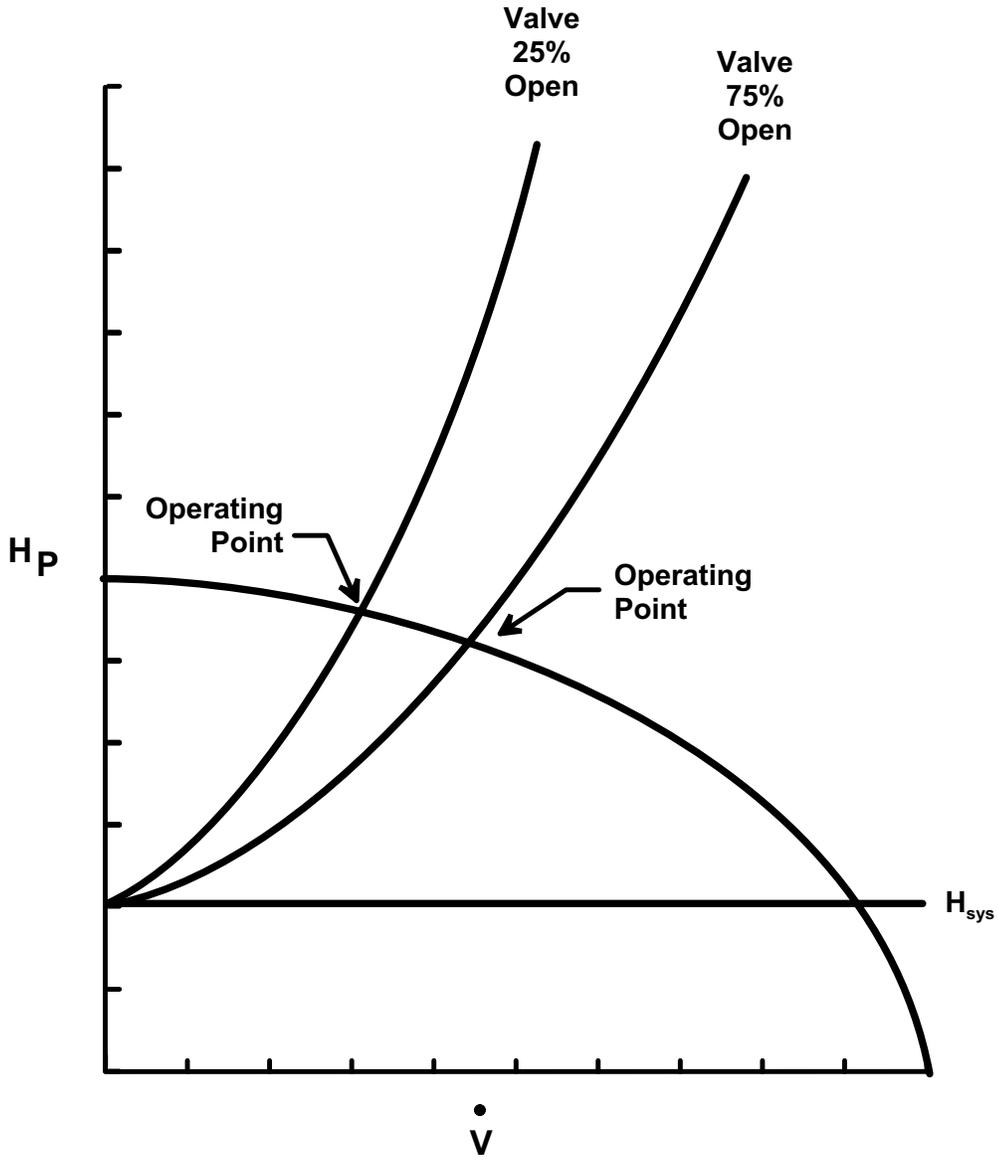


Figure TH11-18

#

1.00

Which ONE of the following statements is correct concerning the POWER SUPPLY (F,G, or H), and ECCS TRAIN (A or B) for the 1-1 RHR pump?

- A. Bus G, Train B
- B. Bus G, Train A
- C. Bus F, Train B
- D. Bus F, Train A

Answer: A

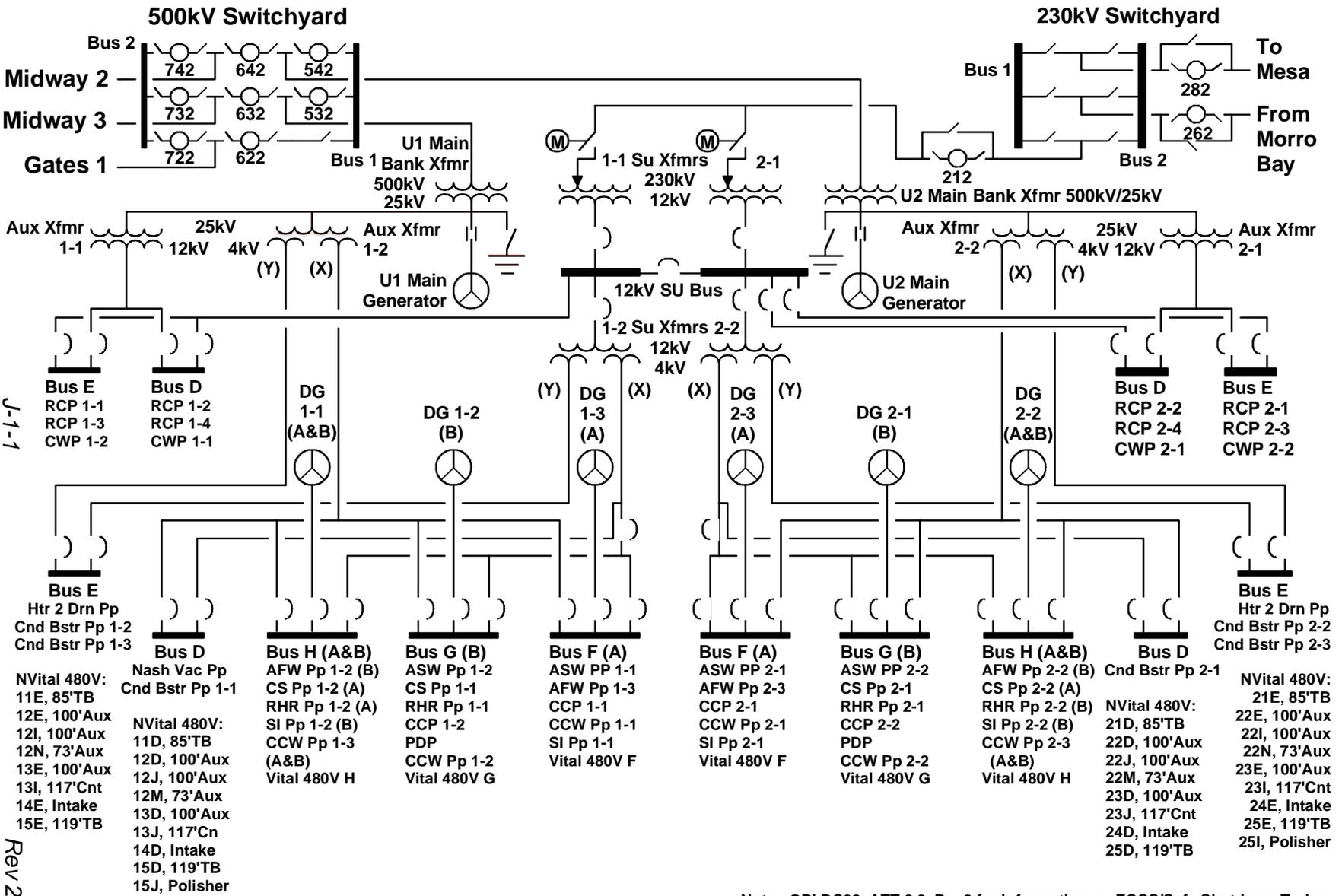
ASSOCIATED INFORMATION:

Associated objective(s):

7011	State the power supply to the RHR pumps
------	-----------------------------------------

Reference Id: P-0648
Must appear: No
Status: Active
User Text: 7356.020122
User Number 1: 0000002.90
User Number 2: 0000003.30
Difficulty: 1.00
Time to complete: 3
Topic: Correct power supply for RHR pump.
Cross Reference: STG B-2

Electrical Distribution Overview



J-1-1

Rev 22

Question 49 Reference

Note: OPI.DC38, ATT 8.2, Pg. 2 for information on ECCS/Safe Shutdown Trains

#

1.00

WHICH ONE (1) of the following describes 12KV and 4KV breaker operation if DC control power is lost to these breakers?

- A. Breakers will remain in their "as is" condition and operation would only be possible by manual means
- B. Remote operation of the breakers would be available but automatic trip functions would not operate
- C. Automatic breaker trips would remain operational but remote operation of breakers would not be possible
- D. Breakers would trip open and operation would only be possible by manual means

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

34876	State the function of DC Control Power.
-------	-----------------------------------------

Reference Id:	P-1514
Must appear:	No
Status:	Active
User Text:	4260.070623
User Number 1:	0000003.00
User Number 2:	0000003.40
Difficulty:	3.00
Time to complete:	3
Topic:	J1 Loss of DC control power to 4KV & 12KV breakers

STG J9, DC Power

Abnormal Operations

Alternate system lineups
Obj 12

Refer to OP J-9:II Operating the Battery Chargers for the following operations.

Section	Title
6.1 to 6.5	Placing Vital Battery Charger ... In Service
6.6 to 6.10	Removing Vital Battery Charger ... From Service
6.11	Swapping Vital Battery Chargers
6.12 to 6.14	Placing Non-Vital Battery Charger ... In Service
6.15 to 6.18	Removing Non-Vital Battery Charger ... From Service

Malfunction effects
Obj 17

The following tables describe effects of malfunctions associated with DC system/component operation.

Effects of ...	Consequence
loss of AC power to the battery chargers on continued operation DC system.	<ul style="list-style-type: none"> • DC systems operation is limited to the capacity of the associated battery. • Batteries are sized to provide necessary DC to respond to a design basis accident. • Unnecessary loads should be secured to prolong battery operation.
loss of DC control power	<p>Breakers used for 4 kV and higher voltage systems use 125 VDC for control power. The load is controlled by opening or closing the breaker using closing (CC) and trip coils (TC).</p> <ul style="list-style-type: none"> • The control power is used in both control and protection schemes. It energizes the CC to CLOSE the breaker (control), and energizes the TC to OPEN the breaker (control and protection). • If the 125 VDC control power is lost, the breaker fails 'as is', that is, it cannot be opened or closed automatically or remotely.

#23185

064.k6.07 Cognitive Level: 1
Salem Unit 1

WEC 11/4/2002

Unit 1 is at 100% power. A relief valve has failed open on 1A Starting Air Receiver. The leakage exceeds the capacity of the starting air compressor. The Primary NEO is responding to the associated alarm.

Which one of the following correctly describes the response of 1A Diesel Generator if a start signal occurs before any operator action?

It will start in the normal time via the two starting air solenoids associated with 1B Starting Air Receiver.

It will take longer to start or may NOT start because the starting air system will be depressurizing/depressurized.

It will take longer to start or may NOT start because the starting air system will be depressurizing/depressurized or the fuel rack may not reset

It will start in the normal time from 1B Starting Air Receiver via all four starting air solenoids.

- (A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids
- (A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids
- (A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids
- (A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids

STG J6B, Diesel Generator page 2.6-9

Starting Air Receivers

Purpose
Obj 11

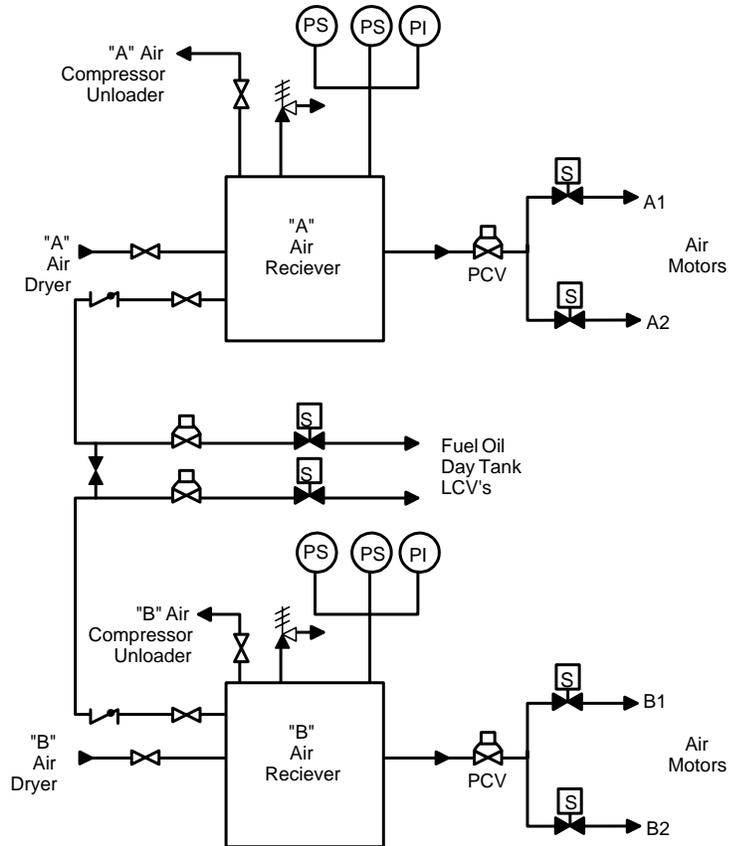
The purpose of the starting air receivers is to hold sufficient air for 45 seconds of continuous diesel cranking. Each receiver has the ability to perform 12 consecutive starts without recharging. Each Starting Air Receiver also provides operating air for a Fuel Oil Day Tank Level Control Valve.

Location
Obj 7

The starting air receivers are located on the north and south sides of the diesel engine room as illustrated on the starting air system location drawing

Physical description

Each starting air receiver has 53 cubic feet capacity. The flowpath through the starting air receivers is shown below:



DEG-49

Continued on next page

Given the following:

- A liquid radwaste discharge permit has been authorized.
- RE-18, Liquid Radwaste Monitor, was determined to be INOPERABLE when adjusting the setpoint.

WHICH ONE (1) of the following actions is correct with respect to liquid radwaste discharges?

- A. Discharges are allowed if two independent samples and release rate calculations are done.
- B. No discharges are allowed until RE-18 is OPERABLE.
- C. A temporary monitor must be installed to perform discharges.
- D. The SFM and the Chemistry and Rad Protection Foreman must jointly authorize the original discharge permit.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8443	State the administrative requirements of Liquid Rad Waste system
69249	Explain the basic principles of operation for the Liquid Radwaste System and the major components and equipment.

Reference Id: P-1245
 Must appear: No
 Status: Active
 User Text: 8443.130743
 User Number 1: 0000001.80
 User Number 2: 0000003.30
 Difficulty: **2.00**
 Time to complete: 2
 Topic: Discharge actions when RE-18 det. INOP

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP G-1:II
REVISION 33
PAGE 4 OF 10
UNITS 1 AND 2

TITLE: Liquid Radwaste System - Discharge of Liquid Radwaste

6.5.2 If RE-18 is found to be inoperable proceed as follows:

CAUTION: Liquid Radwaste discharges may continue for up to 14 days after RE-18 is declared inoperable if the applicable ACTION statements in ECG 39.3 are followed.

- a. Note the date and time that RE-18 is declared inoperable in the Shift Log.
- b. Review the applicable portions of ECG 39.3.
- c. Attach a Caution tag (Admin tagout) to the key switch for the radwaste discharge valve 0-FCV-647, made out to the SFM. Indicate on the tag the date and time of the expiration of the 14 day period when termination of batch releases are required.
- d. Ensure chemistry analyzes a second independent liquid radwaste sample.
- e. Ensure that the release rate calculations are verified by at least two qualified staff members.
- f. If the tank activity is not below the limits stated in CAP A-5, process the liquid radwaste per chemistry instructions to reduce the tank activity.
- g. Initiate an AR to initiate repair of RE-18.
- h. Fill out the appropriate sections of the Discharge Permit.

6.5.3 If RE-18 has been inoperable for less than or equal to 14 days:

CAUTION: Liquid radwaste discharges may continue for up to 14 days after RE-18 is declared inoperable if the ACTION statements in ECG 39.3 are followed.

- a. Review the applicable portions of ECG 39.3.
- b. If RE-18 will remain inoperable longer than 14 days, go to step 6.5.4.
- c. Ensure chemistry analyzes a second independent liquid radwaste sample.
- d. Ensure that the release rate calculations are verified by at least two qualified staff members.
- e. If the tank activity is not below the limits stated in CAP A-5, process the liquid radwaste, per chemistry instructions to reduce the tank activity.
- f. Fill out the appropriate sections of the Discharge Permit.

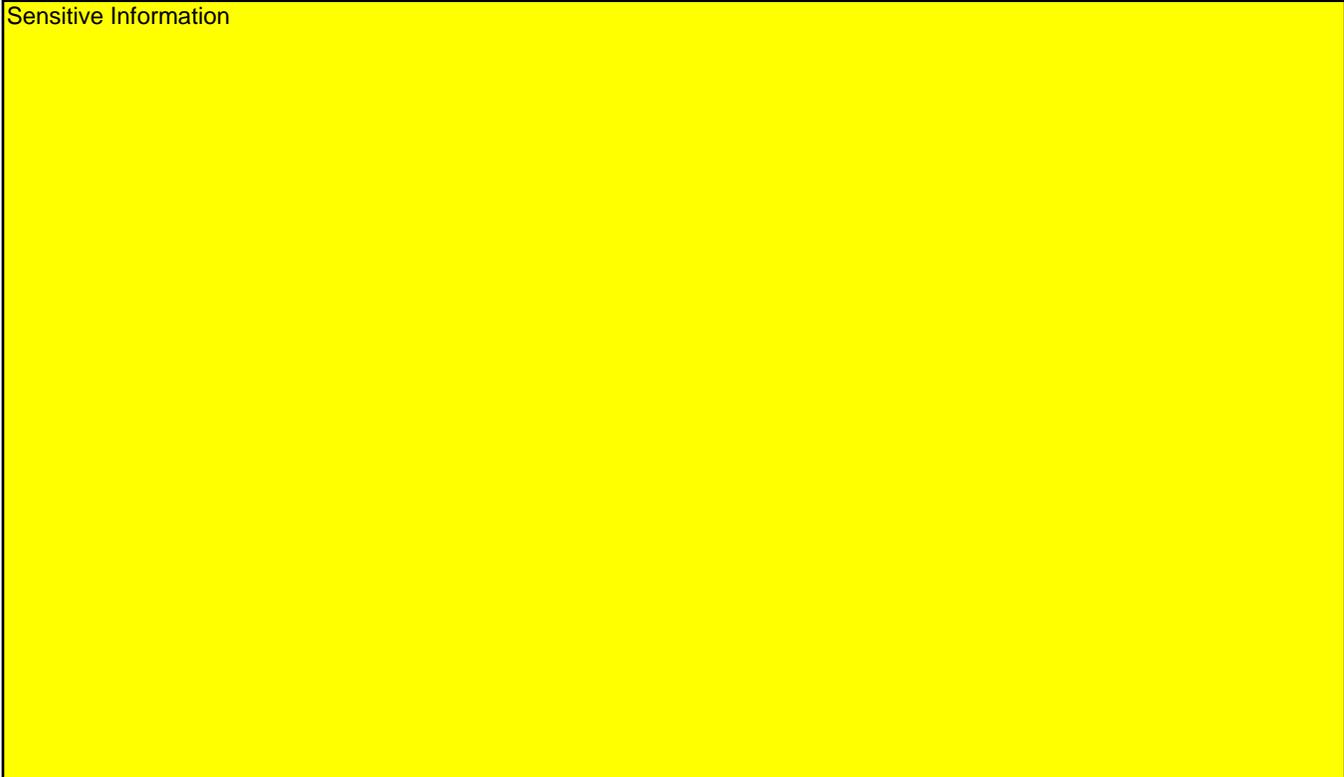
STG E-5, ASW – page 2-4

ASW Pumps

Purpose
Obj 2,3, 21

The purpose of the ASW pumps is to provide the driving head necessary to deliver saltwater to the CCW heat exchangers at a sufficient flow rate to remove the design heat load.

Sensitive Information



Each pump is in a separate room with a watertight door.

Power supplies
Obj 4

The power supplies for the pump motors are:

ASW Pump	4 kV vital bus
1-1	F
1-2	G
2-1	F
2-2	G

Continued on next page

Normal Operations, Continued

Normal system responses (continued)

Effect of ...	Consequence
Standby starts, IA compressors.	<p>Compressors 0-1 through 0-4 will start if their control switch is in AUTO and:</p> <ul style="list-style-type: none"> • PS-130 decreases to 93 psig, or • If any other running compressor trips. <p>Compressor 0-5 through 0-7 will start if they are in standby, green Auto Operation light on and:</p> <ul style="list-style-type: none"> • they have shutdown due to being unloaded for >20 minutes and a compressor load signal is received. <p>or</p> <ul style="list-style-type: none"> • if auto start switch is in ENABLE (ON for 0-7) and the power supply bus has lost power and power is restored

Effects of operation Obj 19

Equipment operating effects on system indications are summarized below. Although numerous possibilities exist for the initial conditions of operating a component, the information below assumes operation from a normal configuration.

Effect of operating...	Consequence
	The operator should expect to see...
IA Compressor 0-1 through 0-4	<p>compressor discharge pressure increase, oil and air temperatures increase. Instrument air header pressure should cycle from:</p> <ul style="list-style-type: none"> • 98 to 105 psig with master unloader ON. • 95 to 102 psig with master unloader OFF.
IA Compressor 0-5 and 0-6	<p>compressor discharge pressure increase, oil and air temperatures increase. Instrument air header pressure should cycle from:</p> <ul style="list-style-type: none"> • 103 to 108 psig on the lead compressor, and • 100 to 104 psig on the lag compressor.

Continued on next page

#

1.00

If instrument air header pressure indicates 89 psig, what would be the status of the instrument air compressors?

Assume all air compressors are available and are aligned in a normal system alignment.

- A. All compressors running and loaded
- B. only Rotary compressors 0-5 & 0-6 running and loaded
- C. only Reciprocating compressors 0-1 through 0-4 running and loaded
- D. All compressors running but only Rotary compressors 0-5 & 0-6 loaded

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

68957	Explain the basic principles of operation for the Compressed Air System.
37565	Analyze Compressed Air System control logic
68959	Describe Automatic operations of the Compressed Air System.

Reference Id: S-1254
 Must appear: No
 Status: Active
 User Text:
 User Number 1:
 User Number 2:
 Difficulty: 3.00

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

- LCO 3.9.4 The containment penetrations shall be in the following status:
- a. The equipment hatch capable of being closed and held in place by four bolts;
 - b. One door in each air lock capable of being closed; and
 - c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
 2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation valve.

-----NOTE-----
 Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
 During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge and exhaust ventilation isolation valves actuates to the isolation position on an actual or simulated actuation signal.	24 months

QuestionId	22497		
ExamType	ILO		
ExamDate	10/1/2002		
AbbrevLocNam	Diablo Canyon Unit 1	Cognitive Level	2
NSSSType	PWR	ExamLevel	R
KaNumber	..103.K3.01		
QuestionStem	Which one of the following conditions concerning the Personnel Air Lock would exceed a Limiting Condition for Operation and require entering a Tech Spec Action Statement?		
Answer	The outer and inner doors are opened simultaneously for a normal transient entry into containment while in MODE 4.		
Distract1	Both air lock doors fail acceptance test criteria while the plant is in MODE 6.		
Distract2	Welding cables are laid through both airlock doors while the plant is in MODE 5.		
Distract3	The outer door is opened for a normal transit entry into containment while in MODE 3.		

#

1.00

A nuclear reactor is exactly critical just below the point of adding heat when a single control rod drops into the core. Assuming no operator or automatic actions occur, when the plant stabilizes, reactor power will be _____ and average reactor coolant temperature will be _____.

- A. lower; the same
- B. lower; lower
- C. the same; the same
- D. the same; lower

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

65661	Explain reactor response to a control rod insertion.
-------	------------------------------------------------------

Reference Id: F-40627
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 2.00

OP AP-12C, Dropped Control Rod

AP-12C, Scope and Symptoms

Scope
Obj 1

OP AP-12C, Dropped Control Rod, provides instructions for plant operation when a control rod becomes disengaged from its drive mechanism and drops into the core.

Symptoms or Entry Conditions
Obj 2, 8

Symptoms of dropped rod are:

- control bank rods stepping out (if in auto).
- rapid drop in T_{AVG} & power, with corresponding decrease in pressurizer level and pressure.
- rod bottom light.
- power range high rate status light (1/4 on negative rate) and PPC printout.
- alarms relevant to dropped control rod as listed in OP AP-12C (review list of alarms, as necessary).

Description of Malfunction
Obj 3

Control rods may drop for the following reasons.

- Blown stationary gripper fuse or other power failure in power cabinet.
- Failed stationary gripper coil in CRDM or open circuit to CRDM.
- High CRDM temperature causing degraded magnetic flux coupling.
- Faulty current order from slave cyclor in logic cabinet or failed firing circuit in power cabinet.

Effects and Consequences of Dropped Rods
Obj 8

Some of the effects and consequences of dropped rods are:

- After power recovers when T_{AVG} decreases, a dropped rod takes on the characteristics of an excessively mispositioned rod.
- It can produce significant distortions in local core power distributions and eventual xenon changes of nearby fuel assemblies, with the potential for localized fuel damage during recovery if the rod is withdrawn too quickly or power is not reduced (depending on how long rod is dropped).
- Depending on location of dropped rod, QPTR may be severely affected and AFD affected somewhat.
- Available SDM will remain essentially constant as total power defect varies by the same amount as the worth of the dropped rod.
- However, if the rod cannot be recovered, available SDM by calculation is reduced and the RIL is raised by an amount equal to the reactivity worth of the dropped rod from the full out position.

P-72907#

1.00

Given the following conditions:

Unit 2 tripped from 29% power.

RCP 2-1 breaker tripped open when the busses swapped.

Which ONE of the following describes the response of T_{hot} and T_{cold} in Loop 2-1?

- A. T_{hot} lowers to approximately equal T_{cold} .
- B. T_{cold} rises to approximately equal T_{hot} .
- C. T_{hot} rises, T_{cold} remains approximately stable.
- D. T_{cold} lowers, T_{hot} remains approximately stable.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

Reference Id:	P-72907
Must appear:	No
Status:	Active
Difficulty:	3.00
Time to complete:	2
Topic:	Idle Loop Indications
Cross Reference:	LA6
Comment:	New 3/31/04

Reviewed by CNH4, 4/17/06

Changed link from LAR1 obj 6104 to STG A-6 obj 35743 since this is more of a systems / transient analysis question than a procedures question.

Ref: STG A-6 Page 3-6 (explains reverse flow in idle loop). See also TH18T Page 29 Table 18-3 for T_{hot} and T_{cold} behavior if the reactor does not trip.

TH18T page 29

Stopping a Reactor Coolant Pump (RCP) with No Resultant Reactor Trip

Stopping a Reactor Coolant Pump (RCP) will normally result in a reactor trip. For this analysis, however, a reactor trip is assumed to not occur. Reactor coolant normally flows from the discharge of each RCP to the reactor vessel. The coolant exits the reactor vessel as hot coolant, enters the steam generator, then exits the steam generator as cold coolant to continue the cycle.

If an RCP is stopped, the high pressure discharge of the other three pumps forces the coolant to flow backwards through the newly idled loop. Therefore, the maximum temperature attainable, anywhere in the idle loop, is T_C of the operating loops. Thus, T_{AVG} of the idle loop can be no greater than T_C of the operating loops.

Due to the reduced backpressure of the idle loop, the flow rate in each of the operating loops increases from 100% to about 108% of rated flow. The idle loop has an equivalent flow of about 32%, in the reverse direction. Total core flow will decrease since only three RCPs are operating, and some of their flow is bypassing the core through the idle loop.

T_C in the idle loop becomes the same as the T_C of the operating loops, due to backflow. T_H in the idle loop is strongly affected by the rate at which feedwater is fed into the idle loop's steam generator, and will be less than T_C . Core ΔT will increase significantly due to core power remaining constant with reduced core flow.

The following table depicts the typical values of RCS parameters resulting from tripping one RCP from an initial power level of 60%, with no reactor trip or safety injection occurring.

**Table 18-3
DCPP PLANT RESPONSE TO TRIPPING ONE RCP FROM 60% POWER**

Parameter	Initial Value	Operating Loop Values	Idled Loop Values
Loop T_H	580°F	584°F	528°F
Loop T_C	544°F	541°F	541°F
Loop ΔT	36°F	43°F	-13°F
Loop T_{AVG}	562°F	562.5°F	534.5°F
Loop Flow	100%	108%	-32%
Core Flow	100%	73%	N/A
T_{SAT}	532°F	524°F	524°F

Pstm	885 psig	825 psig	825 psig
FW Flow, #/hr	2.16 M	2.76 M	0.36 M

With reactor power level maintained at 60%, and core flow reduced from 100% to 73%, the core ΔT will increase by 37% (from 36°F to 49°F). This causes the core exit temperature to increase from 580°F to 590°F. Due to this coolant mixing with the cold coolant (528°F) from the idle loop, the vessel outlet temperature will be 584°F.

The operating loops share most of the steam load previously carried by the idle loop. The steam generators in the operating loops to steam harder, causing steam pressure to decrease. The output of each operating steam generator will increase by 28%, with 8% additional RCS flow. This equates to a 19% increase in loop ΔT s (from 36°F to 43°F). The loop ΔT s do not change as much as the core ΔT due to the increased loop flow, and mixing with the idle loop return flow.

T_H in the operating loops increases (to 584°F), due to core ΔT being greater. T_C in the operating loops decreases (to 541°F), due to the increased heat removal. T_{AVG} of the operating loops, however, remains relatively constant (562.5°F compared to an initial value of 562°F). As mentioned previously, T_H in the idle loop is strongly affected by the rate at which steam generator level is allowed to recover. The T_H value of 528°F results from very closely monitored feedwater flow rates to the idle steam generator.

Also mentioned previously is the fact that the reactor normally trips when stopping a RCP from any significant power level. This is caused by a rapid reduction in steam generator level in the idle loop. If a reactor trip does not occur immediately after the RCP is stopped, another trip signal is generally initiated several minutes after the RCP is stopped. This is caused by the low flow rate of reactor coolant through the idle loop. Feedwater is fed into the idle steam generator at a reasonable rate, but is heated at a very slow rate. The steam generator level control system allows feeding the steam generator to some nominal level, then reduces the feedwater flow rate to some minimal value.

The slow heating of the water in the idle steam generator causes the water to expand, increasing the level to the hi-hi level setpoint (67%). This will cause the reactor, turbine, and main feedwater pumps to trip off the line. This may be avoided by taking manual control of the feedwater control system and feeding the steam generator very slowly.

Discounting any reactor trips associated with tripping one RCP, the most significant effects are the decreased core flow and increased core ΔT . Some control systems utilize a T_{AVG} input, but this is generally an auctioneered high T_{AVG} . Since T_{AVG} of the operating loops remains virtually unchanged, and the idle loop T_{AVG} decreases, stopping one RCP has little or no affect on these control systems.

Pressurizer Heaters

Purpose
Obj 11 The purpose of the Pressurizer Heaters (78 electric heaters) is to maintain Pressurizer water at the saturation temperature corresponding to the desired RCS pressure.

Location The Pressurizer Heaters are located in stainless steel sheaths that are seal welded to a heater well assembly in the lower head of the Pressurizer.

Power supplies
Obj 14 The Pressurizer Heater group power supply information is shown below. Breakers are located in buses.

Group Number	480 VAC Bus	Breaker Number	No. of Heaters	kW Capacity
Proportional group 1	13D	52-13 (23)D5	18	415
Backup group 2	13D/1G	52-13 (23)D6 52-1G(2G)-72*	21	485
Backup group 3	13E/1H	52-13E(23E)-2 52-1H(2H)-74*	21	485
Backup group 4	13E	52-13E(23E)-5	18	415

*Note: Backup vital supply.

Physical description The Heater design characteristics are listed below.

Characteristic	Details
Total number of heaters	78 heaters in 4 groups:
Total rated capacity	1800 kW
Individual heater capacity	23.08 kW
Operating voltage, Hz	480 VAC, 60 Hz
Design pressure	2485 psig
Design temperature	680°F
Design heatup rate	55°F per hour

Continued on next page

#

1.00

In accordance with OP AP-20, "CONDENSER TUBE LEAK," the parameter that is monitored as an indication of demineralizer (polisher) system breakthrough is:

- A. Increase in feedwater cation conductivity - PT 32 on Tracor Westronics Secondary Chemistry Recorder.
- B. Increase in condensate pump cation conductivity >200 $\mu\text{S}/\text{cm}$ - PT 19 on Tracor Westronics Secondary Chemistry Recorder.
- C. Increase in blowdown cation conductivity - PK12-10 S/G BLOWDOWN CATION CONDT'Y HI alarm in.
- D. Increase in condenser tray/tube sheet leak detection conductivity - PK12-14 COND TRAY/TUBE SKT LEAK DET CONDT'Y HI alarm in.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3478	State the entry conditions for abnormal operating procedures
4976	State the actions for increasing condensate cation conductivity

Reference Id: P-5506
 Must appear: No
 Status: Active
 User Text: 4976.050553
 User Number 1: 0000003.00
 User Number 2: 0000002.90
 Difficulty: 2.00
 Time to complete: 3
 Topic: LPA20 - Demineralizer breakthrough.
 Cross Reference: OP AP-20, LPA20
 Comment: REF: OP AP-20 step 4.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP AP-20
REVISION 11
PAGE 3 OF 6
UNITS 1 AND 2

TITLE: Condenser Tube Leak

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

CAUTION: FCV-230 must remain closed to maintain the Condensate Demineralizers in service. A Reactor Trip is required if FCV-230 opens for any reason.

3. **PREVENT Condensate Demin**

Bypassing:

- a. Check FCV-230 CLOSED
- b. OPEN the supply breaker for FCV-230

- a. TRIP the Reactor and GO TO EOP E-0

UNIT ONE / UNIT TWO
52-15J-17 52-25I-17

4. **MONITOR For Condensate Demin Breakthrough:**

TRIP the Reactor AND GO TO EOP E-0.

Feedwater Cation Conductivity, PT 32 on Secondary Chemistry Recorder, is LESS THAN 0.25 μ S/cm

5. **REDUCE Power as Required by Attachment 4.1, While Continuing On In This Procedure:**

6. **SECURE any Liquid Radwaste Discharge in Progress in Preps for Securing a CWP**

7. **PREVENT CST Contamination, If Necessary:**

Condensate Cation Conductivity LESS THAN 15 μ S/cm

- 1) Place HC-3 in MANUAL and CLOSE LCV-12.
- 2) Notify Chemistry to sample the condensate prior to rejecting the hotwell.
- 3) Reject the hotwell overboard per OP C-7A:III.
 - a. Do not decrease hotwell level to LESS THAN 55 inches.
 - b. Monitor Feedwater Suction pressure and start an additional condensate booster pump set if pressure approaches 300 psig.

The design feature of the Spent Fuel Pool will prevent inadvertent draining of the pool below elevation:

- A. 125 ft.
- B. 130 ft.
- C. 132 ft.
- D. 133 ft.

Answer: D

ASSOCIATED INFORMATION:

Associated objective(s):

9697	Identify Technical Specification LCOs
35482	Identify Tech Spec LCOs.
35694	Describe the operation of the Spent Fuel Pool Cooling system. ? Make-up ? Normal alignments ? Refueling alignments ? Malfunctions

Reference Id: R-55487
Must appear: No
Status: Active
User Text: 9555.120111
User Number 1:
User Number 2:
Difficulty: 2.00
Time to complete: 2
Topic: Design features of the spent fuel pool

TS 4.3.2

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133 ft.

4.3.3 Capacity

The permanent spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies. For cycles 14-16, the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 154 fuel assemblies. For cycles 14-16, the total combined spent fuel pool capacity in the permanent and cask pit storage racks is limited to no more than 1478 fuel assemblies.

Spent Fuel Pool Bridge Crane, Continued

Interlocks Obj 4, 28

The purpose of interlocks on the spent fuel pool bridge crane are to prevent damage to spent fuel assemblies thereby preventing a radioactive release.

The table below describes the two classes of interlocks.

Class of Interlock	Prevents ...
Hoist operation	lifting a load greater than 2000 pounds (overload).
	lowering a load less than 1400 pounds (underload) • bypassed with UNDERLOAD PERMISSIVE button.
	fast speed by hoist when close to upper height limit
	lifting a load higher than the hoist is designed for.
	two blocking the hoist block and tackle (block hitting the winch).
	overextending the hoist (letting out too much cable).
Multiple actions	more than one of these operations at a time: • hoist operation or • trolley movement, or • bridge movement.

The table below summarizes the interlocks on the Spent Fuel Pool Bridge Crane.

Interlock	Is active when..	Action
Power transfer switch	not in the OFF position.	One hoist may be energized at a time.
Upper slow zone limit switch	hoist increases to 58 on readout	Shifts hoist speed to slow prior to reaching upper limit switch.
Overload on hoist	load monitor senses greater than 2000 pounds load.	• Stops hoist operation in raise direction. • Energizes overload lt.
Full up limit switch *	hoist increases to 59.7 on readout.	Stops hoist operation in raise direction.

Continued on next page

Spent Fuel Pool Bridge Crane, Continued

Interlocks (continued)

Interlock	Is active when..	Action
Block at motor limit switch	block activates limit switch at hoist motor.	
Underload on hoist (bypassed with UNDERLOAD PERMISSIVE button)	load monitor senses less than 1400 pounds load	<ul style="list-style-type: none"> Stops hoist operation in lower direction. Energizes underload lt.
Full down limit sw. *	hoist decreases to lower limit.	Stops hoist operation in lower direction.
Hoist	hoist is in operation.	Stops trolley or bridge motion.
Bridge	bridge is in operation.	Prevents trolley motion.
Trolley	trolley is in operation.	Prevents bridge motion.

* All GEMCO limit switch height interlocks may be bypassed by use of a LIMIT SWITCH BYPASS push-button located in the GEMCO control box on the bridge end support.

Operation Obj 4, 19

The spent fuel pool bridge crane operations are described in the table below.

Step	Action
1	Close circuit breaker 52-12M-21 to provide power
2	Energize controls by closing disconnect transfer switch for the hoist to be used
3	Move the bridge and trolley to the location desired
4	Lower the hoist to pick up the desired item
5	Pick up the desired item, monitor load
6	Move the bridge and trolley to the desired location
7	Lower the hook until the load is minimized, un-hook the load.
8	Raise the hook clear

***** UNCONTROLLED PROCEDURE - DO NOT USE TO PERFORM WORK or ISSUE FOR USE *****

PACIFIC GAS AND ELECTRIC COMPANY
 NUCLEAR POWER GENERATION
 DIABLO CANYON POWER PLANT
 ANNUNCIATOR RESPONSE

NUMBER AR PK09-02
 REVISION 10
 PAGE 1 OF 2
 UNIT

TITLE: SG 1-2 PRESS, LVL, FLOW

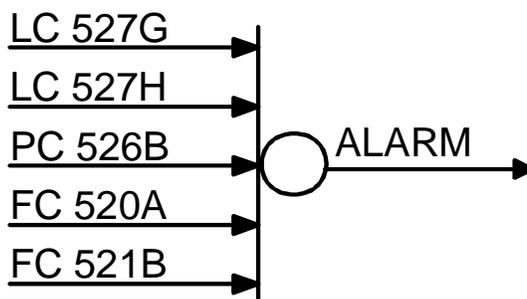
1

07/26/00

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
FC 520A	354	Stm Gen 1-2 Stm Flo less than FW Flow	0.70 x 10E6 lbs/hr
PC 526B	419	Stm Gen 1-2 Press Lo	LT 650 psig
LC 527G	342	Stm Gen 1-2 Lvl Hi From Ref	+ 5%
LC 527H	343	Stm Gen 1-2 Lvl Lo From Ref	- 5%
FC 521B	735	Stm Gen 1-2 FW Flow less than Stm Flow	0.70 10E6 lbs/hr

3. PROBABLE CAUSE

- 3.1 Stm. gen. level deviation.
- 3.2 Stm. flow LT feedwater flow.
- 3.3 Stm. gen. low pressure.
- 3.4 Feedwater flow LT Stm. flow.

4. AUTOMATIC ACTIONS

None

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

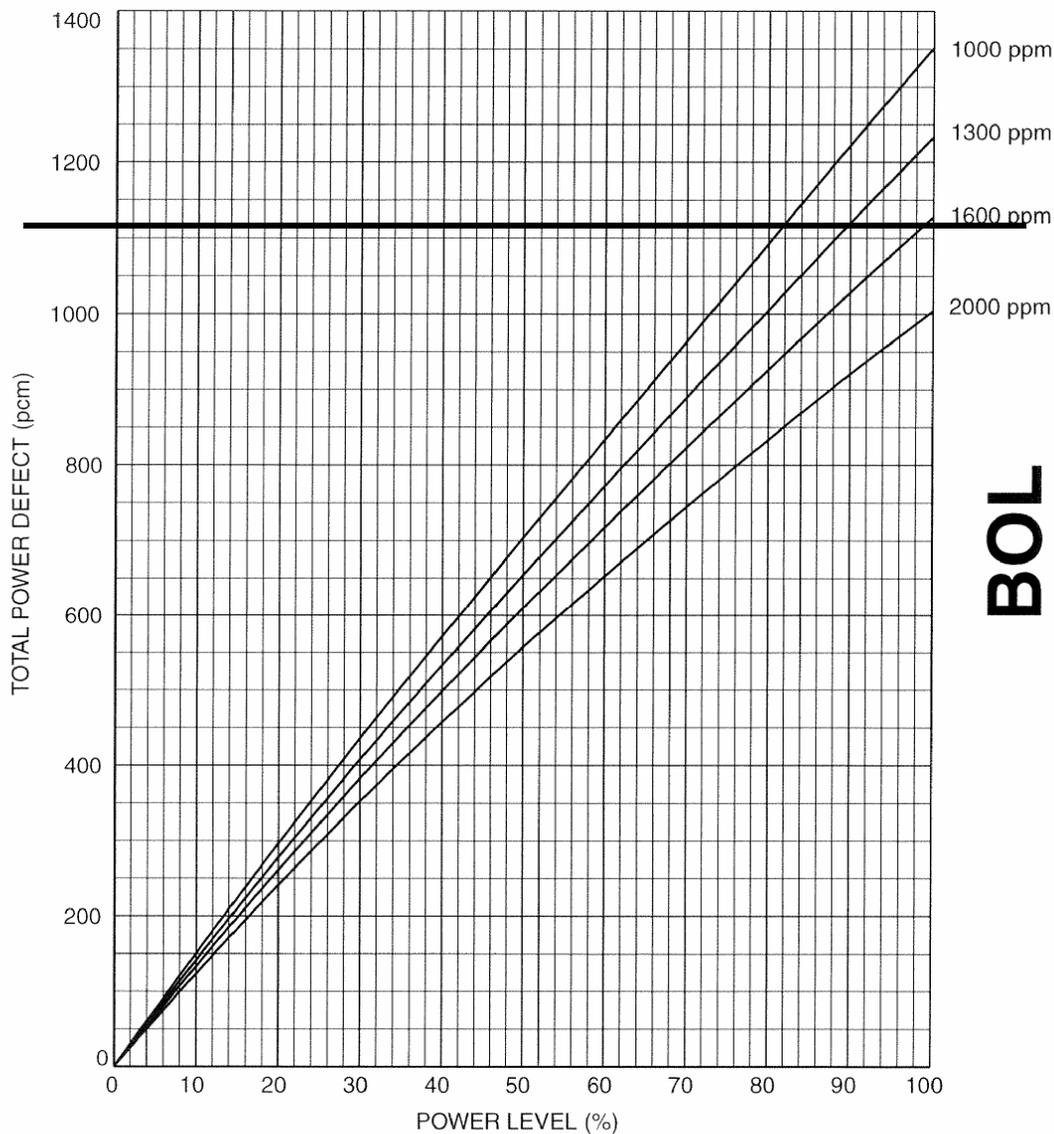
NUMBER AR PK09-02
REVISION 10
PAGE 2 OF 2
UNIT 1

TITLE: SG 1-2 PRESS, LVL, FLOW

5. OPERATOR ACTIONS

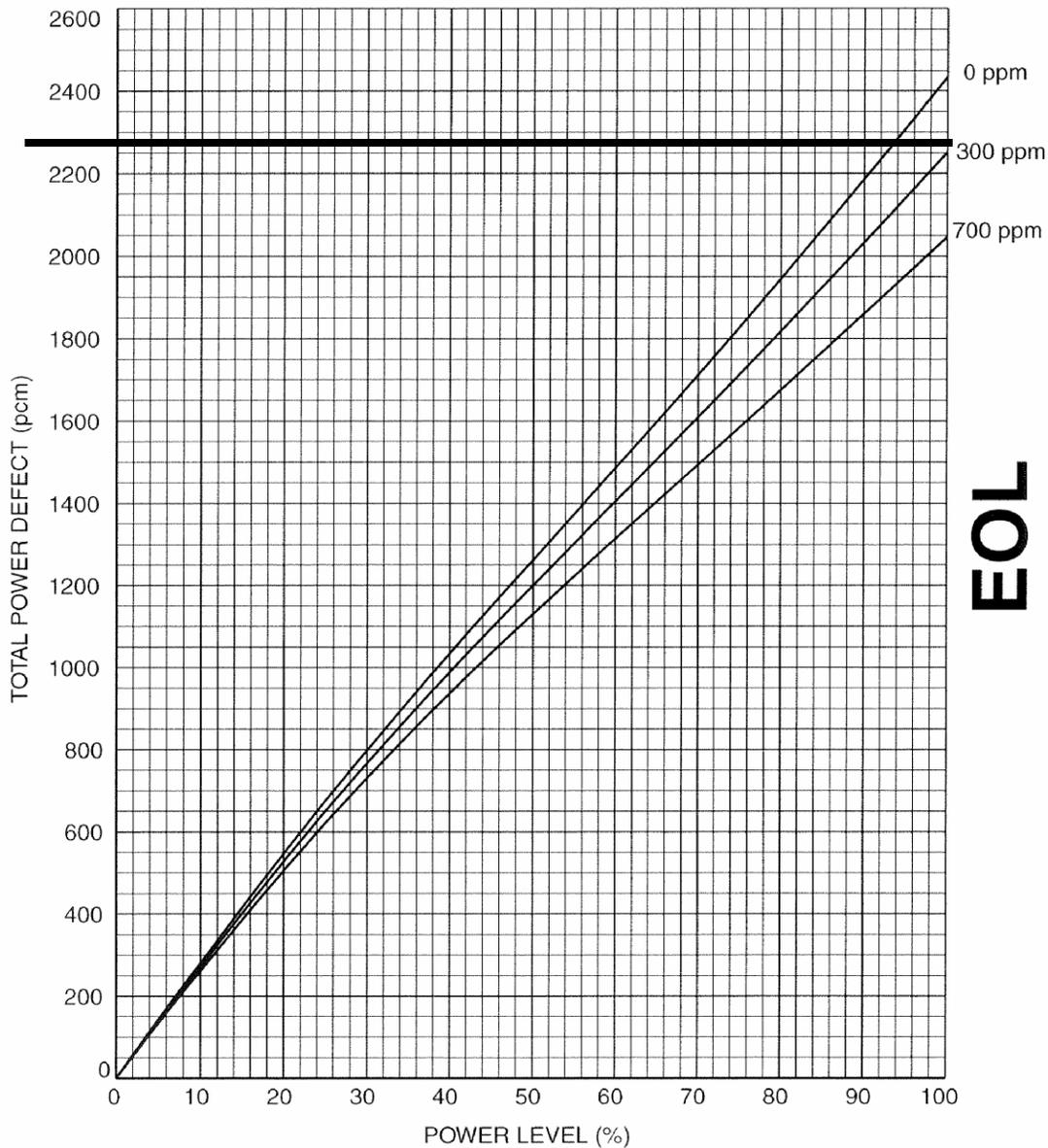
- 5.1 If Reactor Trips, GO TO EOP E-0.
- 5.2 Perform channel check of Stm. Gen. level, press., and flow indications.
- 5.3 Verify steam generator water level control is performing properly in AUTO.
- 5.4 Take Manual control of Stm. Gen. water level and restore Stm. Gen. level and/or steam flow/feed flow as necessary.
- 5.5 If a safety valve has lifted and fails to reseat, trip the reactor and GO TO EOP E-0.
- 5.6 If a malfunction of a Stm. Gen. pressure, level, or flow channel is indicated, GO TO OP AP-5.

DIABLO CANYON POWER PLANT OPERATION DATA
FIGURE R17-1F-3
Total Power Defect as a Function of Power Level at BOL
Cycle 14 for Burnup 0 - 8000 MWD/MTU



SOURCE: WCAP-16466-P, Rev 0, Figure D.4-1

DIABLO CANYON POWER PLANT OPERATION DATA
FIGURE R17-1F-3
Total Power Defect as a Function of Power Level at EOL
Cycle 14 for Burnup 16000 MWD/MTU - EOL



SOURCE: WCAP-16466-P, Rev 0, Figure D.4-3

#

1.00

The liquid radwaste system is discharging to the ocean when a reactor trip occurs. The discharge flow:

- A. must be terminated.
- B. may continue unchanged.
- C. must be reduced.
- D. must be diverted to the other Unit's discharge path.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

69249	Explain the basic principles of operation for the Liquid Radwaste System and the major components and equipment.
4800	Explain liquid rad waste system operation

Reference Id: NS-54684
 Must appear: No
 Status: Active
 User Text: 4800
 User Number 1:
 User Number 2:
 Difficulty: 3.00

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP O-19
REVISION 23
PAGE 5 OF 7
UNITS 1 AND 2

TITLE: Operating Order O-19 - Nuclear Operator Actions
Following a Reactor Trip

NOTE: Chemistry chart used in the following step can be found in EDMS under "NPG Library, Chemistry, Chemistry S/U & S/D".

- Adjust chemical injection for current total AFW flowrate, Chemistry instructions and the Chemistry chart. (See Attachment 9.3)
- Adjust make-up water flow to the CST & TT per the Control Room.
- Channel check FR-53 and RM-23.
- For a Unit 1 reactor trip, notify SCARP that Zinc Injection has been secured and Argon Gas Injection will need to be isolated.

6.4 AUXILIARY BUILDING SENIOR WATCH

6.4.1 Immediate Actions for Auxiliary Building Senior Watch

- **Secure any liquid or gaseous radwaste discharges in progress.**
- Select Boric Acid Evaporator to RECYCLE if in service. Manually isolate Auxiliary Steam to the Evaporator Preheater and Shell by closing AXS-39.
- Check the system's status as a bus transfer may have caused the fluidic logic to change its alignment.
- If "Containment Isolation" alarm is in, secure RCDT Pps, Containment Structure Sump Pps and Rx Cavity Sump Pps. Log pump status in Aux Senior Daily Log and Abnormal Status board. Notify SFM of pump status as soon as plant conditions stabilize.

6.4.2 SUBSEQUENT ACTIONS FOR AUXILIARY BUILDING SENIOR WATCH

None

6.5 INTAKE/OUTSIDE WATCH

6.5.1 Immediate Actions for Intake/Outside Watch

- Performing a normal shutdown of the CW Bay Chlorination Skid. Reference OP E-3:II as necessary.
- If less than two of the four CWP's remain in service, shutdown all ASW continuous chlorination.
- Verify Intake Cooling Pump in service to running CWP.
- Check status of CWP that tripped.
- Walkdown the intake to check for other abnormalities, shutdown any unnecessary equipment that auto started.
- **Secure overboard discharge of WHAT Tank if in progress.**

Which one of the following actions will automatically occur if the gaseous radwaste discharge header radiation monitor (RE-22) detects high radiation?

- A. RCV-17 (GDT'S to Plant Vent) will close.
- B. FCV-410 (GDT'S Outlet Header to Plant Vent) will close.
- C. Gas decay tank vent valves will close.
- D. Gas decay tank fill valves will close.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

37706	Describe the operation of the gaseous radwaste vent valves, including: ? Control switches ? Interlocks and logic ? Automatic actions
69285	Describe Gaseous Radwaste System automatic features.

Reference Id: R-52308
Must appear: No
Status: Active
User Text: 7425.110711
User Number 1: 0000002.10
User Number 2: 0000003.40
Difficulty: 1.00

GDT Outlet HDR to Plant Vent FCV-410, Continued

Interlocks
Obj 9, 12 FCV-410 has an administrative control with a mechanical key interlock for its switch in order to open and close the valve.

Indications
Obj 9 The following indications are available for FCV-410 in the Aux. Building on the Aux. Control Board, 85' elevation (Unit 2 is the same):

FCV-410 Key Control Switch		
Indicating Lights	Meaning	Normal Status
Red	Valve is OPEN	OFF/ON
Green	Valve is CLOSED	ON/OFF

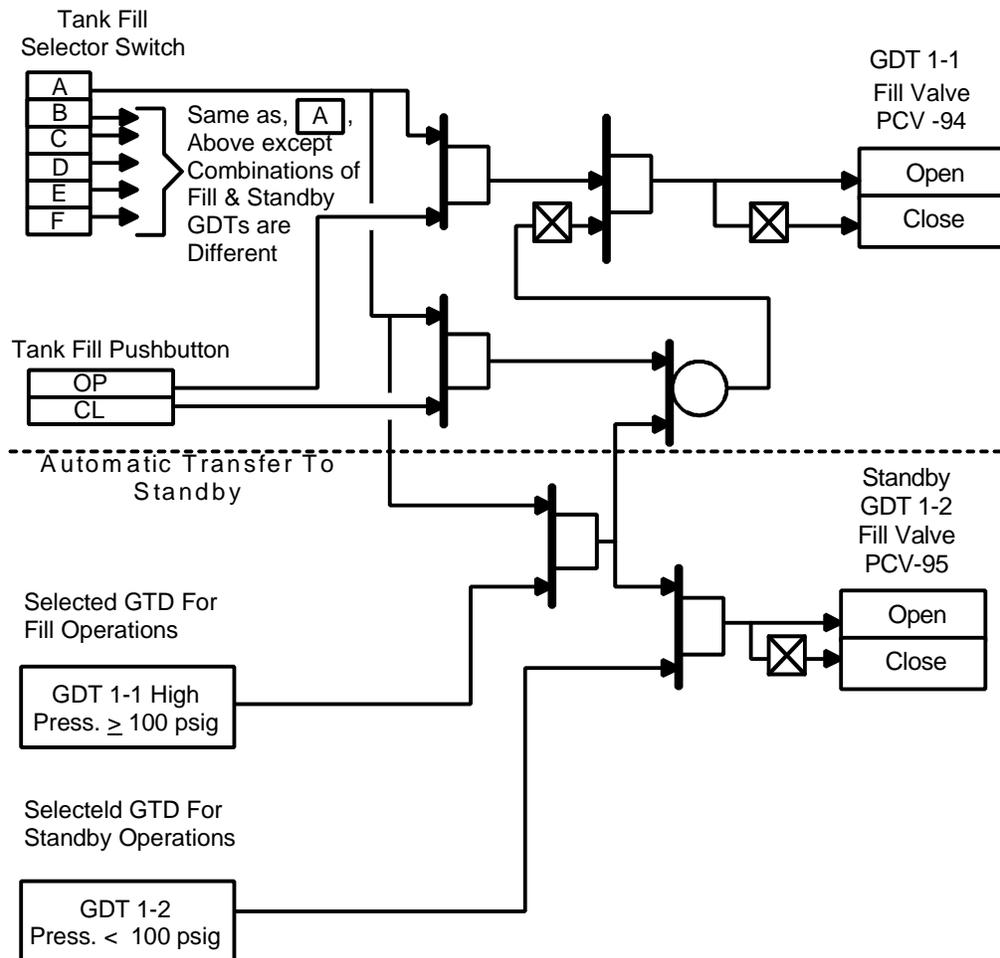
Alarm The following alarm exists for the vent header down stream of FCV-410 and FCV-101 as described below.

Parameter	Source	Location
HIGH vent pressure	PS-183	Aux. Board

* This condition is common when FCV-410 is first opened until PCV-101 responds. If the alarm persists then the operator should close FCV-410 and remove the key.

Gas Decay Tank Fill Valves PCV-94, 95, 96, Continued

Logic diagram The following diagram represents a typical logic for the selected GDT for fill, standby, and automatic transfer operations.



WGS-17

Interlocks

- The positions on the Tank Fill Selector switch allow only one GDT to be filled at a time.
- The interlocks prevent the GDT selected for fill from being vented or purged.
- The standby tank can be purged while being held in standby.

Continued on next page

GDT to Plant Vent Valve RCV-17/RE-22, Continued

Controls *Obj 9, 12*

Each unit's valve RCV-17 is controlled by its respective RE-22. RCV-17 will open, and remains open whenever:

- radiation monitor RE-22 is reset, and
- there is no high radiation alarm actuated.

Automatic Actions *Obj 11, 12*

The automatic actions and function of RCV-17 are described below.

Automatic actions IF...	Function THEN...
High radiation setpoint on RE-22 is exceeded: <ul style="list-style-type: none"> • RCV-17 automatically CLOSES or • RCV-17 is prevented from OPENING. 	The dose, at any time, at and beyond the site boundary, from gaseous effluents will be within the prescribed dose limits of 10CFR20, Appendix B.
RCV-17 fails closed on loss of instrument air.	It ensures that an uncontrolled or unmonitored release is prevented.

Logic

The control logic for RCV-17 is described in the table below. RE-22 is located in area K at 54' elevation of the Auxiliary Building.

Control	Function
The setpoint on RE-22 is variable.	It is set prior to each discharge to ensure that the administrative limit for off site dosage is not exceeded. <ul style="list-style-type: none"> • Whenever the setpoint is exceeded, the alarm must be reset from the Radiation Monitoring System control panel to reopen RCV-17

Interlocks *Obj 12*

RE-22 must be reset for RCV-17 to open.

Continued on next page

GDT Outlet HDR to Plant Vent FCV-410, Continued

Interlocks
Obj 9, 12 FCV-410 has an administrative control with a mechanical key interlock for its switch in order to open and close the valve.

Indications
Obj 9 The following indications are available for FCV-410 in the Aux. Building on the Aux. Control Board, 85' elevation (Unit 2 is the same):

FCV-410 Key Control Switch		
Indicating Lights	Meaning	Normal Status
Red	Valve is OPEN	OFF/ON
Green	Valve is CLOSED	ON/OFF

Alarm The following alarm exists for the vent header down stream of FCV-410 and FCV-101 as described below.

Parameter	Source	Location
HIGH vent pressure	PS-183	Aux. Board

* This condition is common when FCV-410 is first opened until PCV-101 responds. If the alarm persists then the operator should close FCV-410 and remove the key.

PACIFIC GAS AND ELECTRIC COMPANY
OPERATIONS SECTION POLICY

Number: A-21
Revision: 6
Date: 02/09/05
Page: 1 of 2

DIABLO CANYON POWER PLANT UNITS 1 AND 2

TITLE: **CONTROL ROOM PROTOCOL**

APPROVED: _____
Operations Manager

SCOPE

This policy establishes the basic expectations for the conduct of personnel in the control room. The intent is to minimize distractions to the control room watchstanders and to allow them to focus on plant parameters. The expectation is that all activities performed in the control room are conducted in a professional and businesslike manner. The on watch Operations shift team is responsible for the appearance and formality of the control room.

AUTHORITY

The Shift Manager shall implement this policy and is responsible for the conduct of all personnel in the control room. Additionally, the Shift Manager and the Shift Foremen are responsible for implementing any additional requirements consistent with the spirit of this policy.

REQUIREMENTS

The following list is intended to establish a formal control room environment. It is not intended to be all-inclusive. Personnel in the control room are expected to comply with the spirit of this policy for all areas not specifically covered.

- 1. A formal request to enter the “at the controls” (see OP1.DC12 for details) area should be made by everyone except the ON WATCH operators and Control Room Assistant. The request should be made to the SFM. If the SFM is not available, the BOPCO should be asked. The CO may be asked if no other licensed operator is available on that unit.***

Other on shift Operations personnel (OST and Asset Team operators, WWMs, etc.) may enter the elevated area to use or file OVIDs and procedures. This is not intended to be a short cut to the SFM or to be used as a passageway to the kitchen area.

While operating at 100% power, a valid reactor trip signal is received. The REACTOR TRIP INITIATED alarm is in, but the REACTOR TRIP ACTUATED alarm is NOT in, the reactor did not automatically trip.

The BOPCO successfully initiates a reactor trip by de-energizing 480v buses 13D and 13E when the crew enters E-0, "Reactor Trip or Safety Injection," and the reactor indicates TRIPPED.

The SCO transitions to FR-S-1, "Response to Nuclear Power Generation/ ATWS," and begins performing step 1.

The procedural response to this situation was:

- A. Incorrect, since E-0 should be continued once the reactor has been tripped in Step 1, RNO.
- B. Incorrect, since E-0 , Step 1, should have been performed prior to going to FR-S.1.
- C. Correct, direct entry in FR-S.1 is allowed based on the indications.
- D. Correct, since FR-S.1, Step 1, RNO states "Manually trip reactor."

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

38107	Apply the Rules of Usage in EOPs for the CSFSTs and FRGs, including: the six status trees the priority of use of the status trees the priority of use of the color of each CSF when to monitor and/or implement the CSFSTs and FRGs
-------	-------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------

Reference Id: P-68832
 Must appear: No
 Status: Active
 Difficulty: 3.00
 Time to complete: 2
 Topic: Ability to execute procedural steps\

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER EOP E-0
REVISION 30A
PAGE 2 OF 33**

TITLE: Reactor Trip or Safety Injection

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

1.

VERIFY Reactor Trip:

- a. Reactor trip and bypass breakers - OPEN
- b. Reactor Power - DECREASING
- c. DRPI indicates
 - 1) Rods - FULLY INSERTED
 - 2) Rod Bottom lights – LIT

Manually Trip Reactor

IF The reactor trip breakers are closed,

THEN Manually de-energize 480V Buses 13D and 13E (52-HD-13 and 52-HE-4).

IF Reactor will NOT trip,

THEN GO TO FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION / ATWS

WHEN The reactor trips,

THEN Perform the following:

- 1) Re-energize 480V Buses 13D and 13E (52-HD-13 and 52-HE-4).
 - 2) Locally open the reactor trip breakers.
-

2.

VERIFY Turbine Trip:

- a. All Turb Stm Stop Vlvs - CLOSED

a. Manually Trip Turbine,

IF At Least one Stop vlv OR Governor vlv in each lead is NOT Closed

THEN Close MSIVs and MSIV Bypass vlvs.

#

1.00

A surveillance is required only because the unit transitioned to another mode. This is therefore a _____ surveillance.

- A. conditional
- B. periodic
- C. mode-specific
- D. recurring

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3588	Identify the definitions associated with the surveillance testing and inspection program
------	------------------------------------------------------------------------------------------

Reference Id: P-69640
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 1.00

TITLE: Control of the Surveillance Testing Program

- 2.4 The RT W/O module of PIMS allows the plant to keep track of testing activities that are to be performed on a recurring basis. Next due dates are generated based on the last test performance date and the test frequency. The library W/O contains the permanent record of administrative requirements for a given test, for example, the test title and procedure number, which organization performs the test, who is responsible for the procedure, which plant modes the test is required to be current, the test frequency and reference to applicable license requirements. The active W/O is cloned from the library W/O and is used to plan, schedule, assign and update the satisfactory performance of the test. Each test performance will have its history documented on an individual active W/O.
- 2.5 Technical Specification required surveillance requirements can be categorized as conditional or periodic.
- 2.5.1 This procedure only applies to conditional surveillances required to be performed by Tech Spec surveillance requirements. They are to be specifically referenced in a controlling procedure such as an operating, abnormal, emergency, equipment or function control procedure.
- 2.5.2 All periodic surveillances conducted at a frequency less than seven days are controlled by shift turnover lists, or other controlling procedures. All periodic surveillances required on a frequency of seven days or greater are controlled by the master surveillance schedule and are scheduled by designated test coordinators for each section.
- 2.6 Sections other than the responsible section are often required to assist and support in the conduct or analysis of a surveillance procedure. Surveillance tests and inspections often refer to other procedures which are required to accomplish the test. Assistance and support requirements shall be outlined within the controlling surveillance procedure.

3. DEFINITIONS**3.1 Conditional Surveillance**

Any surveillance that is performed on other than the normal periodic time-based surveillance interval. Conditional surveillances may be required for events such as inoperable alarms and monitors or for changing plant conditions such as entering a new mode.

3.2 Periodic Surveillance

A surveillance conducted on a routine surveillance interval.

3.3 Recurring Task Scheduler (RTS)

A computerized scheduling program is used to track accomplishments of the STPs in the master surveillance schedule. This program is the PIMS recurring task scheduler (RTS) and is used for tracking STPs with a test frequency of 7 days or greater.

3.4 Master Surveillance Schedule

The data in RTS that defines the surveillance test program. The data includes frequencies, license requirements, required in "MODE", work "MODE", last performed date, next due date and Tech Spec date. When a procedure step refers to updating the master surveillance schedule, it refers to the act of updating the activity completion data or "taking a work order activity to ACTCMP."

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$, MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1. Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM to be within limits.	24 hours

Following a plant shutdown, the SFM determines the SDM using STP R-19, "Shutdown Margin Determination."

If the SDM is **NOT** satisfactory, how quickly must a boration be initiated to restore SDM?

- A. immediately
- B. 15 minutes
- C. 30 minutes
- D. 60 minutes

Answer: B

ASSOCIATED INFORMATION:

Associated objective(s):

9697A	Identify 3.1 Technical Specification LCOs
-------	-------------------------------------------

Reference Id: B-0125
Must appear: No
Status: Active
User Text: 9697.13ALLN
User Number 1: 3.3
User Number 2: 4.1
Difficulty: 2.00
Time to complete: 2
Topic: R051C5 - T.S. 3.1.1 - Time to initiate boration in MODE 3 if SDM not met.

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
SURVEILLANCE TEST PROCEDURE

NUMBER STP R-2B1
REVISION 22
PAGE 1 OF 17
UNITS

TITLE: PPC Operator Heat Balance

1 AND **2**

11/29/06

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure provides a method of performing a secondary side heat balance using the PPC. The results of the heat balance (given in terms of reactor power level) are then used to verify the calibration of the NIS power range channels.

2. DISCUSSION

2.1 If the results of the heat balance indicate that a nuclear instrument channel calibration does not meet acceptance criteria, perform NIS power range channel adjustments. Occasions will occur where the acceptance criteria are met, but adjustment is desirable. In these situations, NIS adjustments may be made at the discretion of the shift foreman (SFM). NIS adjustments shall be made via performance of Attachment 9.1 of this procedure (or perform STP R-2B2).

2.2 With an evaluated amount of feedwater nozzle fouling, applying the nozzle fouling factor (NFF) will decrease the calculated power to correct for nozzle fouling. To ensure that the calculated power is not overcorrected, when nozzle defouling has occurred or is expected to occur, the NFF will be verified acceptable or will be reset to ≥ 1.0 .

3. RESPONSIBILITIES

3.1 SFM (Test Director), for obtaining data, data reduction, and reporting of results.

4. FREQUENCY

4.1 Either this procedure or STP R-2B2 shall be performed daily when the reactor power level is greater than or equal to 15% RATED THERMAL POWER (RTP). Either this procedure or STP R-2B2 is required within 24 hours of reaching 15% RTP or prior to exceeding 30% RTP. This applies during either a reactor startup or a power increase following a down power to below 15% RTP.

5. TECHNICAL SPECIFICATIONS

5.1 This test partially satisfies Technical Specification SR 3.3.1.2, Table 3.3.1-1, Item 2.a Power Range Neutron Flux, High Trip Setpoint Channel Calibration surveillance requirement. NIS power range channel adjustments are required if calorimetric heat balance calculation results exceed power range channel output by more than +2% RTP.

TITLE: Personnel Dose Limits and Monitoring Requirements

-
- 5.12 Additional Exposure Authorization
- 5.12.1 Annual Administrative Limit for whole body radiation exposure is 4.5 rem per year. In unusual cases, plant management may wish to permit exposures up to, but not exceeding, the Federal Limit (5 rem per year).
- To exceed an Administrative Limit, Attachment 8.3, "Additional Exposure Authorization for Administrative Limits," should be initiated by the requesting organization and approved per procedure.
 - The person initiating this form should be the Supervisor of the worker for whom the extension is being requested.
- 5.12.2 Annual Administrative Guideline for whole body exposure to radiation is 2 rem per year. This guideline may be extended up to the Administrative Limit of 4.5 rem per year.
- To exceed an Administrative Guideline at the plant, Attachment 8.2, "Additional Exposure Authorization for Administrative Guidelines," should be initiated by the requesting organization and approved per procedure.
 - The person initiating this form should be the Supervisor of the worker for whom the extension is being requested.
- 5.12.3 Lifetime Guideline: For a worker who has exceeded the Lifetime Guideline (numerical value, in rem, of an individual's age, in years), an extension is required for the worker to receive greater than 1 rem/yr at the plant for contractors OR 1 rem/yr total for company employees.
- 5.13 INSTRUCTIONS / GUIDANCE for completing PAGE 1, Attachment 8.2 and 8.3:
- "Additional Exposure Authorization for Administrative Guidelines" - Attachment 8.2.
- "Additional Exposure Authorization for Administrative Limits" - Attachment 8.3.
- NOTE:** The requesting organization should complete Page 1 before asking the worker to read, sign and date the Employee Acknowledgment:
- 5.13.1 Worker information: Requester to provide worker information.
- 5.13.2 New Exposure Guideline Requested: Requester to provide requested exposure limit.

NUCLEAR POWER GENERATION
RP1.ID6
ATTACHMENT 8.1

TITLE: Permissible Levels of Exposure

1. Annual Limits and Guidelines

Part of Body Irradiated	Federal Limit	Administrative Limit	Administrative Guideline
<u>Whole body</u> - head, trunk, gonad, arms above the elbow and legs above the knee - (rem TEDE)	5.0	4.5	2.0
<u>Skin & any Extremity</u> - (rem SDE)	50	45	N/A
<u>Lens of the eye</u> - (rem LDE)	15	13.5	N/A
Any <u>individual organ</u> or tissue other than lens of the eye - (rem TODD)	50	45	N/A

2. Five Year Guideline

The whole body exposure (TEDE) for a five year period should not exceed 10 rem. This is implemented as the 2 rem per year Administrative Guideline.

3. Lifetime Guideline

Total accumulated lifetime occupational exposure (TEDE) should not exceed the numerical value (in rem) of an individual's age (in years).

NOTES 1: Individuals who exceed the Lifetime Guideline are restricted to an additional exposure of 1 rem / year.

NOTES 2: In-processing personnel may be restricted to lower limits until their records are complete.

NOTES 3: Declared Pregnant Workers and Minors are restricted to 10% of the Federal Limits listed above.

DEFINITIONS:	
TEDE	Total Effective Dose Equivalent which is the summed total of the Deep Dose Equivalent (external whole body exposure) and Committed Effective Dose Equivalent.
SDE	Shallow Dose Equivalent to the skin and extremities at a depth of 0.007 cm
LDE	Eye Dose Equivalent to the lens of the eye at a depth of 0.3 cm
TODE	Total Organ Dose Equivalent which is the summed total of the Deep Dose Equivalent (external whole body exposure) and the Committed Dose Equivalent (internal dose to the organs or tissues from internal exposure from non-stochastic ALI), when CDE is not zero.

NUCLEAR POWER GENERATION
RP1.ID6
ATTACHMENT 8.2

TITLE: Additional Exposure Authorization for Administrative Guidelines

This page should be completed by the Requesting Organization and forwarded to Dosimetry:

Individual's Name: _____ SSN: _____ / _____ / _____

Department/Company: _____ Time/Date: _____ / _____

New Exposure Guideline Requested: (Enter new guidelines requested)

Annual Administrative Guideline - Annual radiation worker exposure guideline is 2 rem (TEDE) per calendar year. The guideline is extendible to 4.5 rem.

Request: _____

_____ : Request lowered by plant ALARA personnel

Allowed: _____

Lifetime Administrative Guideline - Lifetime exposure is greater than age in years and individual will receive greater than 1 rem/yr at DCCP. The guideline is extendible beyond this exposure, when required.

Request: _____

_____ : Request lowered by plant ALARA personnel

Allowed: _____

Justification: Discuss below (attach additional pages, if needed), with adequate detail, why no one else can reasonably perform the task(s) for which the additional exposure is being requested. Additionally, provide information to justify the amount of the exposure being requested and what actions will be taken on the job to limit the individual's exposure to ALARA (i.e., restrictions to performing lower exposure tasks).

ALARA REVIEW:

[] Denied [] Acceptable for further processing: _____ / _____
(RP Engineering/Supervision) (Date)

Employee Acknowledgment

I acknowledge that I have been informed of a request for additional exposure authorization above the normal guidelines established for the plant and have reviewed the above justification and actions to be taken to restrict my exposure to as low as reasonably achievable. Further, I have reviewed my radiation exposure history and certify that I have received no other occupational exposure at another facility since the beginning of this work assignment at the plant (history information in Dosimetry should be complete).

Signature: _____ Time/Date: _____ / _____

30401

2/2/2006

PWR

Kewaunee Unit 1

1

10CFR20 limits the radiation exposure (dose) to a qualified radiation worker to _____ per year.

Kewaunee Power Station limits the radiation dose to a qualified radiation worker to _____ per year WITHOUT special authorization.

10CFR20 limit

KPS limit

5000 mrem

2000 mrem

3000 mrem

1200 mrem

3000 mrem corresponds to the second line administrative limit (the limit employed after first level of approval, at 60% of the NRC normal limits). Previous 10CFR20 limits also used to specify 3 rem/qtr limit. 1500 mrem correspond to the typical first line limit employed at 40% of the NRC allowed limit.

3000 mrem

1800 mrem

3000 mrem corresponds to the 2nd stage limit (the limit employed after first level of approval). Previous 10CFR20 limits also used to specify 3 rem/qtr limit. 1800 mrem correspond to the typical second line limit employed at 60% of the NRC allowed limit.

5000 mrem

3000 mrem

5000 mrem corresponds to the correct NRC limit. 3000 mrem correspond to the typical second line limit employed at 60% of the NRC allowed limit.

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK15-06
Rev. 17
Page 1 of 6

CONTROL
ROOM
VENT

06/15/05

Effective Date

QUALITY RELATED

1. **ALARM INPUT DESCRIPTION**

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
1055	Cont Rm Vent Supply Fan S-35 and S-36 Flo Lo	FS-5021 <u>AND</u> FS-5023	no flow <u>AND</u> no flow	2.1
1102	CRVS Hi Rad and/or Mode 4 Actuation Trn A	RE25X (U1)	≥ 1.6 mR/hr	2.1
		RE25X (U2)	≥ 1.6 mR/hr	
		RE51X (U1)	≥ 1.6 mR/hr	
		RE53X (U2)	≥ 1.6 mR/hr	
1103	CRVS Hi Rad and/or Mode 4 Actuation Trn B	RE26X (U1)	≥ 1.6 mR/hr	2.1
		RE26X (U2)	≥ 1.6 mR/hr	
		RE52X (U1)	≥ 1.6 mR/hr	
		RE54X (U2)	≥ 1.6 mR/hr	
1135	Control Room Press Htr TEH27A	TS-5005	≤ 75°F	2.1
1168	Control Room Press Htr TEH27B	TS-5007	≤ 75°F	2.1
1170	Control Room Dampers 2, 3, 7 or 8 in Man or UV	42X-2	manual control or UV detected	2.1
		42X-3	manual control or UV detected	
		42X-7	manual control or UV detected	
		42X-8	manual control or UV detected	

(Continued)

CONTROL ROOM VENT

UNIT 1

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
1171	Control Room Press Fan S-99 Trouble	42X-1	loss of power	2.1
		42X-1A	loss of power	
		42X-99	loss of power	
1214	Radiation Instrument Failure	74HRCM1	UV / failure	2.1
1215	Control Room Dampers 2A, 3A, 7A, 8A/Htr HE-N Troub	42X-2A	manual control or UV detected	2.1
		42X-3A	manual control or UV detected	
		42X-7A	manual control or UV detected	
		42X-8A	manual control or UV detected	
1216	Control Room Press Fan S-98 Trouble	42X-1B	loss of power	2.1
		42X-1C	loss of power	
		42X-98	loss of power	
1230	Control Rm Vent Fan OC Trip	49-A-1	thermal overload	2.1
		49-A-2	thermal overload	
		49-A-3	thermal overload	
		49-B-1	thermal overload	
		49-B-2	thermal overload	
		49-B-3	thermal overload	
		49X-A-4	thermal overload	
		49X-B-4	thermal overload	

2. **OPERATOR ACTIONS**

2.1 **General Actions (All Inputs)**

2.1.1 IF high radiation exists as indicated on the RMS Panel on Unit 2 side of the Control Room,
THEN:

CAUTION: Do not manually select both units for Mode 4 operation, as this will start two pressurization fans.

- a. Check Mode 4 status lights illuminated on both units. []

NOTE: Pressurization fans are automatically swapped due to high radiation at intake of running fan. This will occur only once.

- b. IF both units have a high radiation condition,
THEN:
 - 1. On the unit having the lowest pressurization intake radiation, place BOTH selector switches in "MODE 4". []
 - 2. On the other unit, place BOTH selector switches in "MODE 3". []
 - 3. Periodically repeat step 2.1.1b to ensure that the operating fan is selected to supply from the unit with the lowest radiation. []
- c. IF only one unit has a pressurization intake high radiation condition,
THEN:
 - 1. On the unit without the alarming monitor, place BOTH selector switches in "MODE 4". []
 - 2. On the other unit, place BOTH selector switches in "MODE 3". []
- d. Check that damper failure status lights are OFF to confirm proper system lineup and operation. []
- e. IF any damper failure status lights are ON,
THEN select the unselected subtrain. []
- f. IF CRVS streamers are hanging down,
THEN on the affected unit, select the unselected subtrain. []
- g. IF Mode 4 is initiated,
THEN verify Unit 2 pressurization fan S96 OR S97 is running. []

- h. REFER TO reporting requirements in XI1.ID2, "Regulatory Reporting Requirements and Reporting Process," and make any necessary reports. []
- i. WHEN the high radiation condition is cleared, THEN reset ventilation mode PER OP H-5:IV, "Control Room Ventilation System Mode Changes." []

NOTE 1: Input 1055 may alarm during CRVS testing in Mode 4 due to flow switch design.^{Ref 4.2}

NOTE 2: For CRVS fan OC alarm, breakers are in Panels PPCA and PPCB in the CRVS Equipment Room

- 2.1.2 IF the alarm was NOT a result of radiation, THEN perform the following:
- a. Determine the affected equipment by observing printout, VB-4 position, and failure panel indicating lights. []
- b. Dispatch an Operator to inspect the affected equipment. []
- 2.1.3 IF neither train of Control Room cooling are operating, THEN:
- a. Perform a walkdown of equipment cabinets in the Control Room to identify local hot areas. []
- b. Open cabinet doors as necessary to provide additional cooling. []
- c. REFER TO CP M-10, "Fire Protection of Safe Shutdown Equipment," for guidance on providing temporary cooling to the Control Room. []
- 2.1.4 REFER TO TS 3.7.10. []
- 2.1.5 REFER TO the following as necessary to diagnose the problem: []
- OP H-5:IV, "Control Room Ventilation System Mode Changes"
 - STP M-6A, "Routine Surveillance Testing of Control Room Ventilation System"

2.1.6 Probable Causes

- High radiation or radiation monitor failure
- Supply fans S-35 and S-36 failed
- Pressurization fans S-98 or S-99 failed
- Damper failure
- Dampers 2, 2A, 3, 3A, 7, 7A, 8, 8A placed in manual or breaker open
- Pressurization system HEPA filter inlet heater failure
- Overcurrent trip of S-35, CR-35, CP-35, S-36, CR-36, CP-36, S-39 or S-40
- Overcurrent trip alarm caused by deenergizing CP-35 or CP-36 control circuit^{Ref 4.3}

3. AUTOMATIC ACTIONS

3.1 If Mode 3 initiated, both units will shift to Mode 3.

3.2 High radiation at the normal intake or Phase A isolation will initiate Mode 4:

3.2.1 One unit will be in ventilation Mode 4, the other unit in Mode 3.

3.2.2 Failure of a pressurization fan will start an alternate fan.

3.3 If the opposite intake is not already in alarm for high radiation, a high radiation condition at either North or South pressurization intakes will cause a swap to the opposite intake.

3.4 If only one Control Room ventilation (normal intake) radiation monitor is in alarm, only the associated train of equipment will shift:

<u>If RM in alarm is:</u>	<u>these dampers close:</u>	<u>and these dampers possibly fail:</u>
25	2, 3, 7, 8	2A, 3A, 7A, 8A
26	2A, 3A, 7A, 8A	2, 3, 7, 8

4. REFERENCES

4.1 Technical Specification 3.7.10, "Control Room Ventilation System (CRVS)"

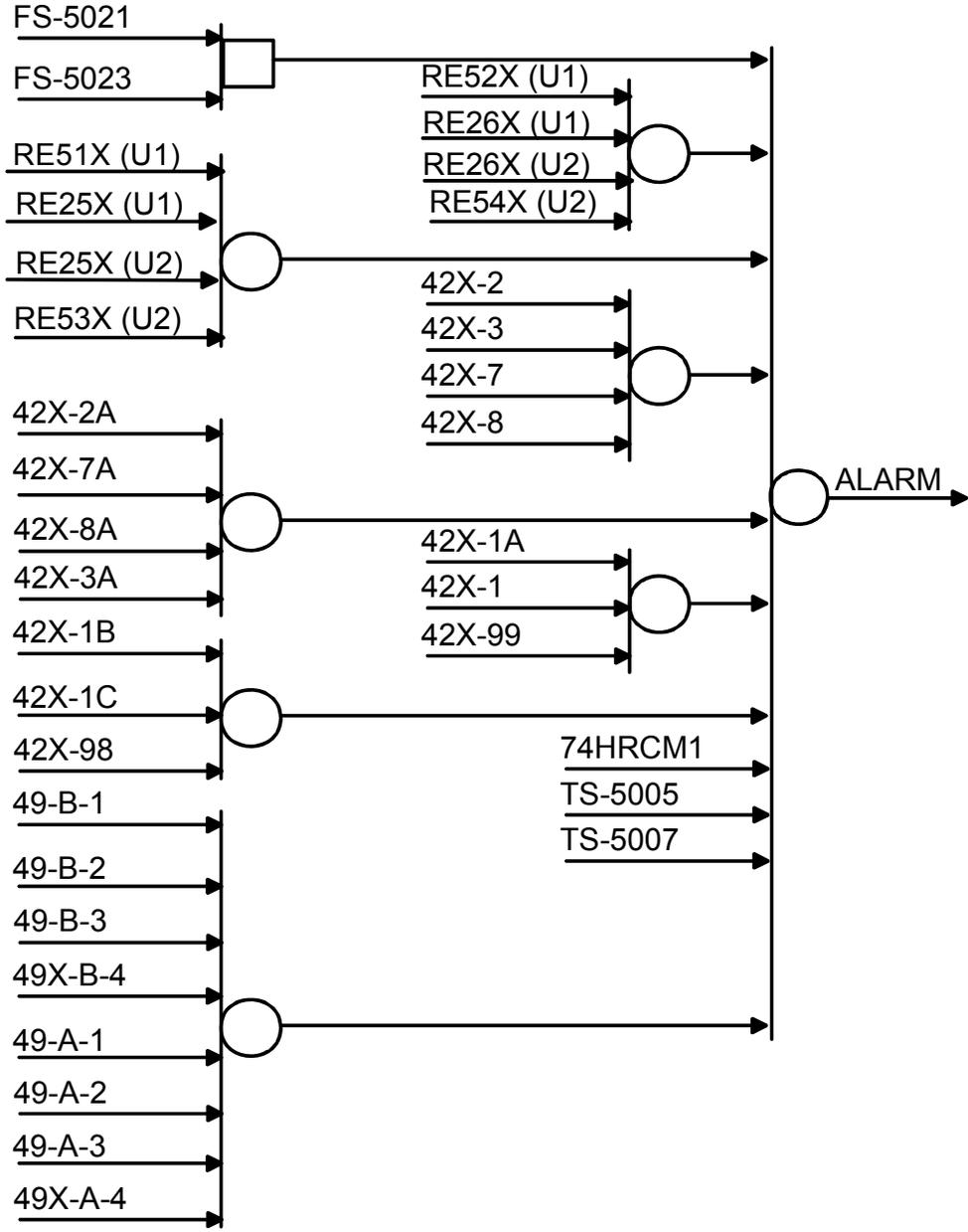
4.2 A0426513, "S-35/36 Flow Low"

4.3 A0467182, "Control Room Vent. Fan OC Trip is in Alarm (CBAR)"

4.4 Main Annunciator Schematics (Electrical Drawing Section 8)

- 501134
- 501141 (Input 1055)
- 501143 (Inputs 1102, 1103)
- 502115 (Inputs 1135, 1168)
- 502116 (Inputs 1170, 1171, 1214, 1215, 1216, 1230)

5. LOGIC DIAGRAM



Mode 4 Operation, Continued

**Fan Select
Logic**
Obj 12

The control circuit attempts to minimize radiation exposure to the Control Room Operators by selecting the best pressurization fan to run.

1. When a Phase A Isolation occurs OR if high radiation is detected at the normal Control Room Ventilation Intake (RE-25 or RE-26), relays will close to start the SELECTED fan on the OPPOSITE Unit. If the opposite Unit's pressurization fan is not running and the radiation at the Pressurization Fan (RE-51, 52, 53, or 54) is not in alarm, then the selected fan will start.
2. If the SELECTED fan fails to start within 15 seconds, the NON-SELECTED fan will start, provided that the opposite Unit's pressurization fan is not running and the radiation at the Pressurization Fan is not in alarm.
3. If high radiation prevents either fan from running, relays will start the selected fan on the affected unit. *This will happen regardless of the radiation level at this fan.*
4. If a pressurization fan is running with a high radiation, the operator may start a pressurization fan on the other end by:
 - a. Selecting Mode 3 on the side to be shutdown.
 - b. Selecting Mode 4 on the side to be run.

The fan to be run will then start.

The previous fan will then shutdown. ("Opposite Fans Shutdown" signal is lost.)

The fan will continue to run as long as a high radiation exists at the shutdown fan.

If radiation levels are normal, selecting Mode 4 will start the selected pressurization fan.

Continued on next page

KNOWLEDGE:

- o The PORV on the ruptured steam generator should remain available to limit steam generator pressure unless it fails open. This will minimize any challenges to the code safety valve.
- o Means of closing and isolating plant specific valves between the main steamline isolation valve and turbine stop valve, such as steam dump valves, Moisture Separator Reheater (MSR) valves, sample line valves, etc. These valves provide a backup means of isolating the ruptured steam generator if the associated main steamline isolation valve should fail. These valves may be closed in parallel with subsequent recovery steps after the ruptured SG(s) are isolated from the intact SG(s) by closing the appropriate main steam isolation and bypass valves.
- o If it is necessary to trip the turbine-driven AFW pump, in order to quickly terminate the radiological release from and depressurization of the ruptured SG, it might still be possible through local actions to achieve selective isolation of the ruptured SG from the turbine-driven AFW pump. In this context selective isolation means that the ruptured SG and only the ruptured SG is isolated from the turbine-driven AFW pump. Selective isolation is feasible only if at least one SG capable of supplying the turbine-driven AFW pump is intact. In terms of the reference plant, in which the turbine-driven AFW pump can be supplied from two SGs, each supply-steam line contains an air-actuated isolation valve normally operated from the control room and a locally operated manual isolation valve. Downstream of the isolation valves the two steam lines join to supply the turbine-driven AFW pump. Local actions can be directed toward closing the manual isolation valve from the ruptured SG and/or toward local-manual operation of the air-actuated isolation valve from the ruptured SG, which the control room operator tried but failed to close while attempting to isolate flow from the ruptured SG. In either case, once the ruptured SG is selectively isolated from the turbine-driven AFW pump, the pump can be restarted if necessary.

PLANT-SPECIFIC INFORMATION:

- o Identify valves between main steamline isolation valve and turbine stop valves.
- o Actions for isolating the ruptured steam generator should consider any difference in the time delay between closing individual main steamline isolation valves (slow close) and closing all main steamline valves (fast close). The benefit of steam dump to condenser must be weighed against the possibility of steam generator overfill due to a delay in stopping primary-to-secondary leakage.
- o Action should be taken to identify and isolate steam traps upstream of the MSIVs. Unisolated traps could result in a steam release causing the ruptured generator to slowly depressurize. An unisolated steam trap could also result in a small release of activity to the secondary side of the plant.
- o (O.03) Setpoint for SG PORV controller (typically 25 psi below lowest safety valve set pressure).
 - This setpoint should be greater than no-load pressure in order to minimize atmospheric releases from the ruptured steam generator and less than the minimum safety valve setpoint to prevent lifting of the code safety valves. The 25 psi margin is a typical value to allow for opening of the PORV prior to lifting of the safety valve.
- o If the turbine-driven AFW pump will not trip or cannot be tripped from the control room, then actions to isolate it from the ruptured SG must be pursued locally. These actions depend on the plant-specific design, and range from local-manual trip of the turbine-driven AFW pump to local operation of manual isolation valves. Priority should first be given to terminating the radiological release from and depressurization of the ruptured SG through the turbine-driven AFW pump. Then time and effort can be directed toward selective isolation of the ruptured SG from the turbine-driven AFW pump. If selective isolation of the ruptured SG can be achieved, then the turbine-driven AFW pump can be restarted if necessary.

#

1.00

A reactor Trip and SI have occurred on Unit 1 due to a Steam Generator Tube Rupture. E-3, Steam Generator Tube Rupture, is currently in progress. The ruptured S/G has been identified, and its 10% Steam Dump controller is in AUTO and set for 1040 PSIG.

Why is the ruptured S/G 10% Steam Dump controller setpoint increased to 1040 PSIG?

- A. This will minimize the radiological releases while preventing lifting of the code safety valves.
- B. If the S/G goes water solid, the 10% Steam Dump valve can discharge water.
- C. To increase ruptured S/G pressure in order to decrease the primary to secondary leakage.
- D. To ensure that S/G pressure can NOT increase above 1055 PSIG.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7920	Explain basis of emergency procedure step
------	-------------------------------------------

Reference Id: P-27379

Must appear: No

Status: Active

Difficulty: 3.00

*** UNCONTROLLED PROCEDURE - DO NOT USE TO PERFORM WORK or ISSUE FOR USE ***

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP FR-H.1
REVISION 21
PAGE 2 OF 28

TITLE: Response to Loss of Secondary Heat Sink

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

1. CHECK Total Feedflow LESS THAN 435 GPM DUE TO OPERATOR ACTION:

a. RETURN TO procedure and step in effect

a. GO TO Step 2.

CAUTION: Feedflow should not be reestablished to any Faulted S/G if a Nonfaulted S/G is available.

2. CHECK If Secondary Heat Sink Is Required:

a. RCS Pressure - GREATER THAN ANY Intact S/G

a. Return to Procedure and Step in Effect.

b. RCS Hot Leg Temperature - GREATER THAN 350°F

b. IMPLEMENT OP B-2:V, RHR - PLACE IN SERVICE DURING PLANT COOLDOWN while continuing in this procedure.

IF Adequate cooling with RHR System established,

THEN RETURN TO procedure and step in effect.

3. CHECK CCP Status - AT LEAST ONE AVAILABLE

Perform the following:

a. Stop all RCPs.

b. GO TO Step 12 (Page 13). OBSERVE CAUTION prior to step.

STEP DESCRIPTION TABLE FOR FR-H.1 Step 1 – CAUTION 1

CAUTION: If total feed flow is less than (S.02) gpm due to operator action, this guideline should not be performed.

PURPOSE: To alert the operator that the performance of FR-H.1 is required only if minimum feed flow capability is lost

BASIS:

During the performance of certain guidelines (e.g., ECA-2.1, FR-P.1, FR-S.1 and FR-Z.1), it is possible that the SG level is below the narrow range and the total feed flow is throttled to less than the minimum AFW flow requirement. If the feed flow is reduced due to operator action to minimize feed flow as instructed in these guidelines and the capability of providing the minimum feed flow is available (i.e., pumps and valves in the feedwater system are capable of being used if necessary), then a transfer from these guidelines is not required and guideline FR-H.1 should not be performed.

ACTIONS:

Determine if total feed flow is less than (S.02) gpm due to operator action

INSTRUMENTATION:

- o Total feed flow indication
- o Feed flow control valve position indication
- o Feed pump status indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

The operator must be aware that the availability to provide feed flow to the steam generators determines if FR-H.1 is to be performed.

#

1.00

Under which of the following set of conditions would the BOPCO be expected to take the specified corrective action without obtaining prior permission and then report his actions to the SFM "after the fact"?

- A. Starting the standby condensate/booster set upon its failure to automatically start, during an LTB.
- B. Placing the standby ASW/CCW HX in service in response to alarm AR PK01-01, "ASW SYS HX dp/HDR PRESS."
- C. Starting CCP 1-1 following an overcurrent trip of operating CCP 1-2.
- D. Starting a standby CFCU due to an observed increase in containment temperature.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3705	Explain the responsibilities of the control room staff positions during emergencies
------	-------------------------------------------------------------------------------------

Reference Id: R-55094
 Must appear: No
 Status: Active
 User Text:
 User Number 1:
 User Number 2:
 Difficulty: 2.00

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP1.DC11
REVISION 26
PAGE 4 OF 13

TITLE: Conduct of Operations-Abnormal Plant Conditions

- 5.2.3 The shift foreman has control room command and provides overall direction and supervision of the unit while maintaining an overall view of the plant situation. The SFM should remain in the "controls" areas when the plant is in an unstable condition.
- 5.2.4 Two licensed operators should be assigned to perform control board manipulations and perform necessary verifications or respond to information requests. To best perform these duties these operators should:
- a. Communicate in accordance with the operations communication standard (OP1.DC12).
 - b. Organize their activities so that one operator is responsible for the control console and vertical boards one and two, and the other operator is responsible for vertical boards three, four, and five.
 - c. Promptly report all unexpected or unexplained equipment operation or plant response to the shift foreman.
 - d. Utilize self-verification techniques for all equipment operation.
 - e. Take time critical actions to ensure nuclear safety, prevent injury or damage to property, or to maintain service of critical equipment.
 - f. Actuate those safety systems/components which have failed to respond to actuation signals and promptly inform the SFM.
- 5.2.5 The WCSFM shall report to the Control Room during abnormal plant operations.
- a. Following a reactor trip, the WCSFM shall assist the crew by assessing plant response and informing the SFM appropriately. The WCSFM should use plant instrumentation, SPDS displays and the main annunciator system to monitor:
 - Reactor coolant system (RCS) and pressurizer (PZR) heatup/cooldown rates
 - RCS pressure for unexpected changes
 - Containment parameter changes
 - PORV lifts or leakage
 - Safety valve lifts or leakage
 - Equipment malfunctions the crew is unaware of
 - Condenser availability
 - Unexpected annunciator alarms

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
ABNORMAL OPERATING PROCEDURE

NUMBER OP AP-1
REVISION 16
PAGE 1 OF 10
UNITS

TITLE: Excessive Reactor Coolant System Leakage

1 AND 2

07/19/06

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure covers RCS leakage conditions where the charging system is capable of maintaining normal PZR level while the PZR heaters maintain normal system pressure. The goal is to limit the release of radioactive material by isolating defective component or reduce the magnitude of the leakage to within Tech Spec limits. ITS 3.4.13, "RCS Operational Leakage," applies in Modes 1-4.

2. SYMPTOMS

- 2.1 Irregular RCP seal flow (FR-157 or 159, VB2)
- 2.2 Charging/letdown flow mismatch (FI-134A, VB2; FI-128A, CC2)
- 2.3 Decreasing VCT level (LI-112, VB2)
- 2.4 Abnormal seal injection flow (FI-144, 143, 115, 116; VB2)
- 2.5 Abnormal letdown flow (FI-134A, VB2)
- 2.6 PRT high level LI-470, high press PI-472, or high temp TI-471 (VB2)
- 2.7 High RCDT level (LI-188, Aux Board), or increased pumpdown frequency
- 2.8 PZR safety valve or PORV discharge line high temp (TI-465, 467, 469, or 463; VB2)
- 2.9 Rx vessel flange leakoff temp high (TI-401, VB-2)
- 2.10 High Containment Sump or Rx Cavity Sump levels (LR-60, 61, 62; PAM1)
- 2.11 High Containment air temp or pressure (YR-26; VB1)
- 2.12 Increased Containment radiation (RM-11 or 12, RMS CAB II)
- 2.13 Increasing rad levels (RE-17A/B) with possible autoclosure of RCV-16 (VB1)
- 2.14 Increasing CCW surge tank level (VB1)
- 2.15 Possible autoclosure of FCV-357 (VB1)
- 2.16 Increasing CCW header "C" flow (FI-46, VB1)
- 2.17 Increasing letdown temp (TI-130, VB2)
- 2.18 TCV-130, CCW to Letdown HX, goes full open (VB2)
- 2.19 Possible flashing in letdown line (PI-135, VB2)

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER OP AP-1
REVISION 16
PAGE 2 OF 10
UNITS 1 AND 2**

TITLE: Excessive Reactor Coolant System Leakage

- 2.20 Possible Main Annunciator Alarms:
 - 2.20.1 (PK01-07) CCW SYS SURGE TK LVL/MK-UP
 - 2.20.2 (PK01-08) CCW HEADER C
 - 2.20.3 (PK01-16) CONTMT ENVIRONMENT P250
 - 2.20.4 (PK01-17) CFCU DRAIN LEVEL HI
 - 2.20.5 (PK04-24) VCT PRESS/LVL/TEMP
 - 2.20.6 (PK05-01 to 04) RCP NO.____
 - 2.20.7 (PK05-14) RCDT LEVEL HI-LO
 - 2.20.8 (PK05-17) PZR PRESSURE LOW
 - 2.20.9 (PK05-20) PZR RELIEF/SAFETY VLVS OPEN
 - 2.20.10 (PK05-21) PZR LEVEL HI/LO
 - 2.20.11 (PK05-22) PZR LEVEL HI/LO Control
 - 2.20.12 (PK05-23) PZR SAFETY OR RELIEF LINE TEMP
 - 2.20.13 (PK05-25) PRT PRESS/LVL/TEMP
 - 2.20.14 (PK11-02) RX FLANGE LEAK-OFF TEMP HI
 - 2.20.15 (PK11-06) SJAE HI-RAD
 - 2.20.16 (PK11-17) SG BLOWDOWN HI RAD
 - 2.20.17 (PK11-18) MAIN STM LINE HI RAD
 - 2.20.18 (PK11-21) HIGH RADIATION

***** UNCONTROLLED PROCEDURE - DO NOT USE TO PERFORM WORK or ISSUE FOR USE *****

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
OPERATING PROCEDURE

NUMBER OP O-21
REVISION 1A
PAGE 1 OF 4
UNITS

TITLE: Operating Order 0-21: Isolating Leaking Thimble Tubes

1 AND 2

01/06/00

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This Operating Order provides instructions for isolating leaking incore thimble tubes from their respective 10 Path Rotary Transfer Device. These instructions should be implemented whenever the Moveable Incore Detector System leak detection system alarm activates and the alarm is considered to be valid.

2. DISCUSSION

2.1 This procedure ensures that thimble tube leaks are properly verified and isolated in a safe and timely manner.

3. RESPONSIBILITIES

- 3.1 Shift Foreman (SFM) for implementing this procedure.
- 3.2 Radiation Protection Section for High Rad Area support and deconning support, if required.
- 3.3 Maintenance Services (MS) for troubleshooting assistance (removal of 10 Path Rotary Transfer Device side covers).

4. PREREQUISITES

- 4.1 All Moveable Incore Detectors have been placed in their STORAGE locations.
- 4.2 An SWP is written and approved for Seal Table Room entry.
- 4.3 Handheld pyrometer is available for tube identification.

5. PRECAUTIONS AND LIMITATIONS

5.1 A Special Work Permit is required for work performed in the Seal Table Room under these instructions.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP O-21
REVISION 1A
PAGE 2 OF 4

TITLE: Operating Order 0-21: Isolating Leaking Thimble Tubes

UNITS 1 AND 2

6. INSTRUCTIONS

CAUTION: If at any time during the implementation of this procedure it is observed that Reactor Coolant System charging flow increases to maintain PZR level, implement OP AP-1 and discontinue this procedure. The Seal Table Room conditions would prohibit entry for isolation of the leak.

- 6.1 Verify that all Moveable Incore Detectors are placed in their STORAGE locations.
- 6.2 Notify Radiation Protection Section that a containment entry will be required to isolate a thimble tube leak in the Seal Table Room.
- 6.3 Attempt to quantify the leak rate while performing subsequent steps in this procedure.
 - 6.3.1 Perform STP R-10C.
 - 6.3.2 Refer to Technical Specifications 3.4.6.2 (ITS 3.4.13), Reactor Coolant System - Operational Leakage.
- 6.4 Monitor RM-7 for changes in Seal Table Room radiation levels as an indication of whether the leak is large since the leak detection system drains onto the seal table.
- 6.5 Make an entry with Radiation Protection and MS personnel to locate and isolate the leaking thimble tube. Take a 10-foot ladder, a handheld pyrometer, and a straight-slot screwdriver suitable for use on #10-24 pan-head screws.
- 6.6 Upon entering the Seal Table Room, visually inspect the six 10 Path Rotary Transfer Devices for obvious signs of leakage. If no leakage is observed, proceed to Step 6.7. If a leak is observed at one of the 10 Path Rotary Transfer Devices proceed as follows:
 - 6.6.1 Immediately close all of the isolation valves (9 or 10) above the seal table for those thimble tubes associated with the 10 Path Rotary Transfer Device which is leaking.
 - 6.6.2 Exit the Seal Table Room and notify Control Room personnel to reset the leak alarm at the MIDS Power Distribution Panel.

NOTE: It may be necessary to reset the leak detection system alarm numerous times if there is some residual water still draining out of the 10 Path Rotary Transfer Device.
 - 6.6.3 After a few minutes verify that the alarm stays out and that no water is leaking out of the open-ended drain tubing.
 - 6.6.4 Verify that the 10 Path Rotary Transfer Device is no longer leaking.

SECTION B: RCP NO. 1 SEAL FAILURE (Continued)

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

1. CHECK No. 1 Seal Leakoff Flow on Affected RCP(s)

- FR-159, RCP 1 **OR** RCP 2 SEAL No. 1 RTN FLO
- FR-157, RCP 3 **OR** RCP 4 SEAL No. 1 RTN FLO
- Decision Table

	RCP Radial Bearing Outlet Temperature AND No. 1 Seal Outlet Temperatures are: STABLE	RCP Radial Bearing Outlet Temperature OR No. 1 Seal Outlet Temperatures are: INCREASING
--	-------------------------------------------------------------------------------------------------------------	----------------------------------------------------------------------------------------------------------------

No. 1 Seal Leak Off Less Than 0.8 GPM	1. Shutdown Unit per OP AP-25 within 8 hours AND continue with this section. 2. Monitor for RCP vibration trip criteria 3. When Stable in Mode 3 Then <ol style="list-style-type: none"> a. Stop affected RCP b. Close valve affected spray if RCP 1 or RCP 2 	Perform ALL of the Following: <ol style="list-style-type: none"> 1. Trip the reactor AND GO TO EOP E-0, "Reactor Trip or Safety Injection" while completing this step. 2. Stop affected RCP(s) 3. Close spray valve (if RCP 1 or RCP 2).
No. 1 Seal Leak Off is: 0.8 GPM to 6 GPM	1. Continue pump operation. 2. Check seal injection flow GREATER THAN No. 1 seal leakoff flow. 3. Monitor Attachment 4.1 parameters for affected RCP(s). 4. Contact Operations Management and EFIN for further evaluation. 5. Return to procedure and step in effect.	<ol style="list-style-type: none"> 4. Close Seal No. 1 outlet isolation valve 3 to 5 minutes after RCP is stopped. 5. Monitor Attachment 4.1 parameters for affected RCP(s).
No. 1 Seal Leak Off Greater Than 6 GPM AND Less Than 7 GPM	1. Shutdown Unit per OP AP-25, "Rapid Local Reduction" within 8 hours AND continue with this section. 2. Monitor for RCP vibration trip criteria. 3. Stop affected RCP(s) 4. Monitor Attachment 4.1 parameters for affected RCP(s). 5. Close Seal No. 1 outlet isolation valve 3 to 5 minutes after RCP is stopped. 6. Close spray valve (if RCP 1 or RCP 2).	

SECTION B: RCP NO. 1 SEAL FAILURE (Continued)

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

NOTE 1: Total Seal Leakoff Flow is the sum of No. 1 and No. 2 Seal Leakoff indications for affected RCP(s).

NOTE 2: It is acceptable to assume a No. 2 seal leakoff of less than 1 gpm if PK05 "RCP SEAL No. 2 LKOFF FLO HI" is NOT IN.

2. CHECK RCP No. 2 Seal Leakoff Flow LESS THAN 1 GPM

- RCP Seal No. 2 Leakoff FLO HI Alarm NOT IN

Perform ALL of the following:

- Trip Reactor **AND** GO TO EOP E-0, "Reactor Trip or Safety Injection" while completing this step.^{T35185}
- TRIP affected RCP(s)
- Between 3 to 5 minutes after RCP stopped, CLOSE Seal No. 1 outlet Isolation Valve for affected RCP(s).
- Monitor CCW temperature and flow to affected RCP(s).
- Close spray valve (if RCP 1 or RCP 2).

3. CHECK Total Seal Leakoff Flow - LESS THAN 8.0 GPM

- RCP No. 1 SEAL LKOFF **FLO** Less Than 7 GPM:
 - FR-159, RCP 1 or RCP 2
 - FR-157, RCP 3 or RCP 4

Perform ALL of the following:

- Trip Reactor **AND** GO TO EOP E-0, "Reactor Trip or Safety Injection" while completing this step.^{T35185}
- TRIP affected RCP(s)
- Between 3 to 5 minutes after RCP stopped, CLOSE Seal No. 1 outlet Isolation Valve for affected RCP(s).
- Monitor CCW temperature and flow to affected RCP(s).
- Close spray valve (if RCP 1 or RCP 2).

SECTION B: RCP NO. 1 SEAL FAILURE (Continued)

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

4. CHECK RCP Parameters within OPERATING LIMITS per Attachment 4.1

Perform ALL of the following:

- a. Trip Reactor **AND** GO TO EOP E-0, "Reactor Trip or Safety Injection" while continuing this section.
- b. TRIP affected RCP(s).
- c. **IF** RCP RAD BRG OUT TEMP **OR** SEAL WTR OUTLET TEMP INCREASING, **THEN** Between 3 to 5 minutes after RCP stopped, CLOSE Seal No. 1 Outlet Isolation Valve for affected RCP(s).
- d. Monitor CCW temperature and flow to affected RCP(s).
- e. Close spray valve (if RCP 1 or RCP 2).

5. PERFORM STP R-10C, "Reactor Coolant System Inventory" or Estimate to Determine if RCS Leakage Excessive

Implement AP-1, "Excessive Reactor Coolant System Leakage" while completing this step.

6. VERIFY RCPs Thermal Barrier Cooling In Service:

Perform the following:

- a. CCW-FCV-750 RCP Thermal Barrier Return IC Valve - OPEN
- b. CCW-FCV-357 RCP Thermal Barrier Return OC Valve - OPEN
- c. RCP No. 1 Seal Outlet temperatures-NORMAL
- d. RCP Radial Brg Outlet temperatures-NORMAL
- e. CCW Header C alarm (PK01-08) - OFF

- a. Verify Seal Injection flow to all RCPs greater than or equal to 6 gpm per pump.
- b. **IF** Seal Injection flow to all RCPs less than 6 gpm per pump **THEN** GO TO Section F, pg. 29, of this procedures

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK05-01
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Page 1 of 11

RCP
NO. 11

12/27/05

Effective Date

QUALITY RELATED
CONTAINS REACTOR TRIP CRITERIA

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
358	RCP 1-1 Seal Leakoff Flo Lo	FC-159A	<0.9 gpm	2.1
359	RCP 1-1 Seal Water Delta P Lo	PC-188	<250 psi	2.1
361	RCP 1-1 Pp Radial Brg Temp Hi	TC-155	>160°F	2.1
366	RCP 1-1 Seal Water Inj Flo Lo	FC-144	<6.5 gpm	2.1
370	RCP 1-1 No. 1 Seal Outlet Temp Hi	TC-148	>170°F	2.1
500	RCP 1-1 Temp PPC	Y0401C		2.1
	<u>INPUTS TO Y0401C</u>	<u>PPC ID</u>		
	RCP 1-1 Stator Temp	T0412A	≥248°F	
	RCP Mtr 1-1 Top Radial Brg Temp	T0413A	≥180°F	
	RCP Mtr 1-1 Top Thrust Brg Temp	T0414A	≥180°F	
	RCP Mtr 1-1 Lo Radial Brg Temp	T0415A	≥180°F	
	RCP Mtr 1-1 Lo Thrust Brg Temp	T0416A	≥180°F	
501	RCP 1-1 OC Trip	51XVE3	max ≥400 A pri inst ≥3680 A pri th ≥2160 A pri	2.1
		51XVE3R	max ≥440 A pri inst ≥4000 A pri	
502	RCP 1-1 Fdr Grd	50NVE3	grounded	2.1
503	RCP 1-1 Standpipe Lvl Hi	LC-486	>123 ft 7 in el.	2.1
504	RCP 1-1 Standpipe Lvl Lo	LC-487	<119 ft 9 in el.	2.1

(Continued)

2.2.5 Probable Causes

- Number 1 or Number 2 Seal failure
- Loss of injection water
- Crud in Number 1 Seal
- Low differential pressure across Number 1 Seal

2.3 RCP 1-1 Seal Water Low Differential Pressure (Input 359)

2.3.1 IF RCP seal water differential pressure is below 255 psid,
THEN perform the following:

- | | |
|-------------------------------------------------------------------------------------------|-----|
| a. TRIP the Reactor. | [] |
| b. <u>While completing this step</u> , GO TO EOP E-0, "Reactor Trip or Safety Injection." | [] |
| c. TRIP RCP 1-1. | [] |
| d. Place in "MANUAL" and CLOSE PCV-455A, Pressurizer Spray Valve. | [] |
| e. IMPLEMENT OP AP-28, Section B, "RCP Number 1 Seal Failure". | [] |

2.3.2 Raise RCS pressure OR lower VCT pressure. []

2.3.3 Probable Causes

- Low RCS pressure
- High VCT pressure
- Number 1 Seal Failure

2.4 RCP Radial Bearing Temperature High (Input 361)

2.4.1 IF RCS pressure is above 1000 psig,
THEN GO TO OP AP-28, Section E, "Loss of CCW to an RCP or RCP High Temperature." []

2.4.2 IF ALL of the following conditions exist:

- Seal injection flow to each RCP—ABOVE 8 gpm
 - Number 1 Seal leakoff flow—BELOW 1 gpm
 - RCS pressure—BETWEEN 100 and 1000 psig
 - CVCS-1-8141A, B, C, D, Number 1 Seal Leakoff Valves—ALL OPEN
- THEN OPEN CVCS-1-8142, Number 1 Seal Bypass Valve, as needed to lower temperature. []

2.4.3 IF ALL of the above conditions are normal,
THEN direct maintenance to check for an RTD failure. []

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP AP-11
REVISION 22
PAGE 15 OF 38
UNITS 1 AND 2

TITLE: Malfunction of Component Cooling Water System

SECTION E: LOSS OF CCW FLOW TO RCPs

SYMPTOMS

- 1. Thermal barrier and lube oil cooler cooling water return high temperature indication.
- 2. Possible Main Annunciator Alarms
 - a. CCW HEADER C (PK01-08)
 - 1) RCP L.O. Clr CCW Flo Lo
 - 2) RCP Thermal Barrier CCW Flo Lo
 - 3) CCW Hdr C Flo Lo
 - b. RCP No. ____ (PK05-01, 02, 03, 04)
 - 1) RCP ____ Temp PPC

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

CAUTION: IF RCP No. 1 Seal Outlet Temperature exceeds 235°F **OR** RCP Radial Bearing Temperature exceeds 225°F, DO NOT restore RCP seal cooling.

- 1. VERIFY CCW Flow To All RCP Lube Oil Coolers:
 - a. Verify CCW Vlvs - OPEN
 - FCV-355
 - FCV-356
 - FCV-749
 - FCV-363
 - b. RCP L.O. Clr CCW Flow LO Alarm (PK01-08) – NOT IN
 - c. RCP Temp PPC Alarm (PK05-01, 02, 03, 04) - NOT IN

IF CCW Flow to RCP(s) CANNOT be restored to Lube Oil Coolers within 5 minutes,
THEN 1) TRIP reactor.
 2) TRIP affected RCP.
 3) GO TO EOP E-O, REACTOR TRIP OR SAFETY INJECTION.

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER OP AP-11
REVISION 22
PAGE 16 OF 38**

TITLE: Malfunction of Component Cooling Water System

UNITS 1 AND 2

SECTION E: LOSS OF CCW FLOW TO RCPs (Continued)

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

- 2. **VERIFY RCP Seal Injection - IN SERVICE:**
 - RCP Seal Injection Flow between 8 and 13 GPM
 - RCP Seal No. 1 Outlet Temp - NORMAL
 - RCP Radial Brg Outlet Temp - NORMAL

- IF** Both thermal barrier and seal injection flow are lost **AND CANNOT** be immediately restored,
- THEN**
 - 1) Manually TRIP reactor.
 - 2) TRIP affected RCP(s).
 - 3) GO TO EOP E-O, REACTOR TRIP OR SAFETY INJECTION while implementing the next three steps, as applicable.
 - 4) Isolate seal injection to the RCP(s) before restarting a charging pump by locally CLOSING the following valves:
 - CVCS-8382A, Seal Inj Filter 1-2 (2-1) Outlet
 - CVCS-8382B, Seal Inj Filter 1-1 (2-2) Outlet
 - CVCS-8387A, Filter Bypass
 - 5) If all RCPs are affected, close FCV-357, RCP Thermal Barrier CCW Return Isolation.
 - 6) Initiate cooldown to MODE 5 per OP L-5 "Plant Cooldown From Minimum Load to Cold Shutdown" after exiting the EOPs

- 3. **VERIFY CCW Flow To All RCP Thermal Barriers - NORMAL:**

CAUTION: If FCV-357 closed on high flow, do not attempt to open FCV-357 until condition causing high flow is cleared.

- | | |
|------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|----------------------------------------------------------------------------------------|
| <ul style="list-style-type: none"> a. Verify FCV-357 did not close on high flow b. Verify Thermal Barrier Return Vlvs FCV-750 and FCV-357 - OPEN c. Verify RCP Thermal Barrier CCW Flow Lo Alarm (PK01-08) - NOT IN | <ul style="list-style-type: none"> a. GO TO Section B Step 5.b, Page 6. |
|------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|----------------------------------------------------------------------------------------|

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER OP AP-11
REVISION 22
PAGE 6 OF 38
UNITS 1 AND 2**

TITLE: Malfunction of Component Cooling Water System

SECTION B: CCW SYSTEM INLEAKAGE (Continued)

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

**5. DETERMINE Leak Location: _
(Continued)**

b. RCP Thermal barriers

- b. Isolate as follows:
- 1) Verify FCV-750 CLOSED
 - 2) Locally ISOLATE Thermal Barrier CCW return (inside containment) by closing as applicable:
 - RCP 1: CCW-234
 - RCP 2: CCW-242
 - RCP 3: CCW-251
 - RCP 4: CCW-262
 - 3) Monitor containment sump for expected level increase
 - 4) IMPLEMENT OP AP-1, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE

c. Excess Letdown Heat Exchanger

- c. Isolate as follows:
- 1) ISOLATE RCS flow to Heat Exchanger (VB2)
 - Close CVCS-8166
- OR -
Close CVCS-8167
 - Close HCV-123
 - 2) ISOLATE CCW flow to Heat Exchanger:
 - Locally Close CCW-426
 - Locally Close CCW-431
- OR -
Close FCV-361
 - 3) Adjust charging flow to minimum or restore normal letdown.

THIS STEP CONTINUED ON NEXT PAGE

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP AP-15
REVISION 17
PAGE 7 OF 15
UNITS 1 AND 2

TITLE: Loss of Feedwater Flow

SECTION A: ONE MFP TRIPS WITH BOTH MFPs OPERATING (Continued)

- | <u>ACTION/EXPECTED RESPONSE</u> | <u>RESPONSE NOT OBTAINED</u> |
|--------------------------------------------------------------------------------------------------------|------------------------------|
| 16. <u>Reset LOAD TRANSIENT BYPASS RELAY</u> ^{T35511} | |
| 17. <u>COMPLETE Secondary Realignment:</u> ^{T35511} | |
| a. Notify the Condensate Polisher Watch and CLOSE FCV-230. | |
| b. CLOSE FCV-55 <u>AND</u> place in AUTO. | |
| c. Place the following mode selector switches from CONT ONLY to AUTO: | |
| • TCV-23 | |
| • FCV-31 | |
| • FCV-30 | |
| • LCV-12 | |
| d. Shut down one condensate/booster pump set <u>AND</u> return to AUTO. | |
| 18. <u>NOTIFY Chemistry If Power Reduction Exceeded 15% Within One Hour</u> | |
| 19. <u>GO TO OP L-4, Normal Operation At Power:</u> | |
| a. Verify applicable steps have been performed and comply with applicable Precautions and Limitations. | |

- END -

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) \geq 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity > 1.0 μ Ci/gm.	-----NOTE----- LCO 3.0.4c is applicable. -----	Once per 4 hours
	A.1 Verify DOSE EQUIVALENT I-131 specific activity within the acceptable region of Figure 3.4.16-1.	
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 specific activity to within limit.	48 hours
B. Gross specific activity of the reactor coolant 100 \bar{E} μ Ci/gm.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 specific activity in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} to < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ $\mu\text{Ci/gm}$.	7 days
SR 3.4.16.2	-----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq 1.0 \mu\text{Ci/gm}$.	14 days <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period.
SR 3.4.16.3	-----NOTE----- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ----- Determine \bar{E} from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

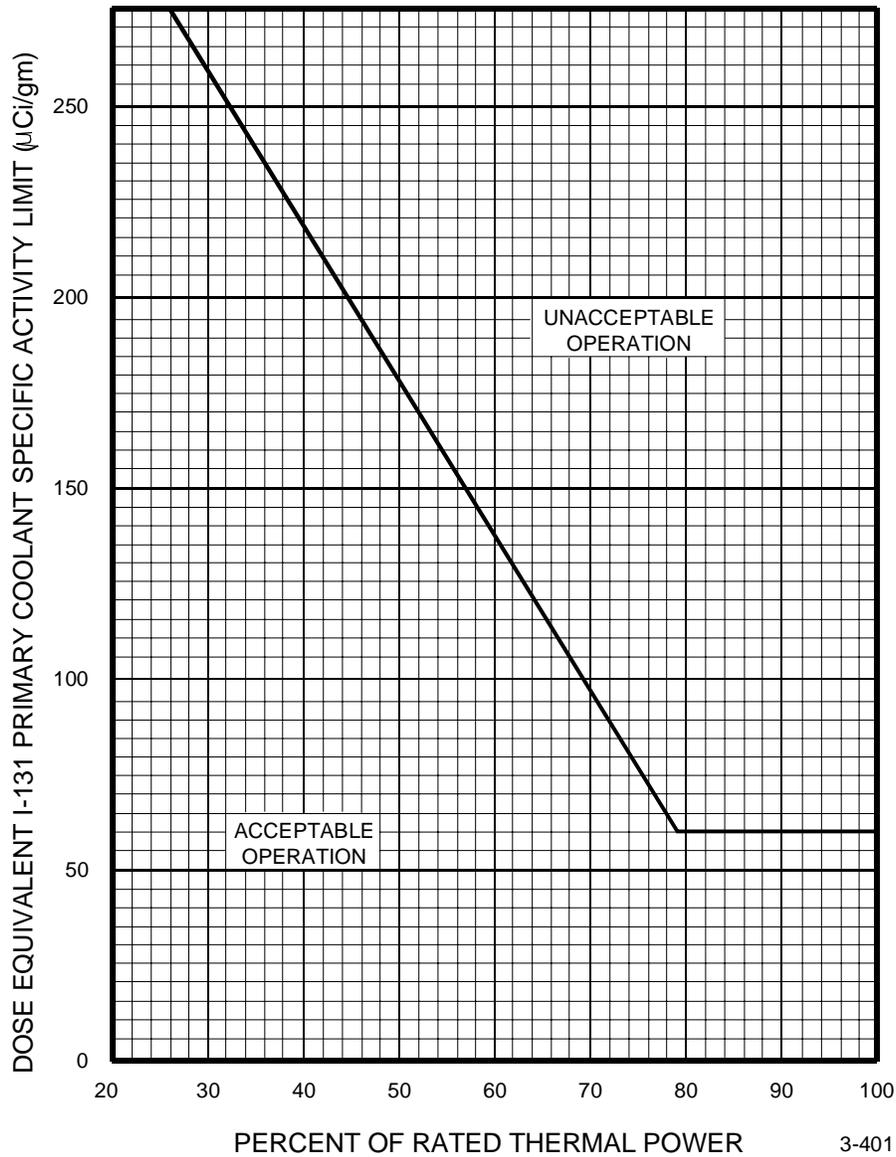


Figure 3.4.16-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT
VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT
SPECIFIC ACTIVITY > 1 µCi/GRAM DOSE EQUIVALENT I-131.

With the unit at 100% power, a loss of a vital DC bus will:

- A. Result in loss of a 4kV vital bus when the unit trips.
- B. Prevent a manual Unit Trip.
- C. Result in a loss of DC power to the main generator air side seal oil back-up pump.
- D. Prevent a train of phase "A" isolation valves from closing.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7116	Explain the consequences of loss of DC vital bus
------	--------------------------------------------------

Reference Id: A-0735
 Must appear: No
 Status: Active
 User Text: 7116.07633
 User Number 1: 0000003.70
 User Number 2: 0000004.30
 Difficulty: 3.00
 Time to complete: 3
 Topic: LPA-23 Loads supplied by vital DC
 Cross Reference: DUTY AREA 63, AP-23 APPX A/B/C

**PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
ABNORMAL OPERATING PROCEDURE**

**NUMBER OP AP-23
REVISION 11
PAGE 1 OF 17
UNITS**

TITLE: Loss of Vital DC Bus

1 AND 2

05/02/03

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure covers the steps to be taken in the event of a loss of one vital DC electrical bus. Steps to stabilize the plant are presented first. Loss of each DC bus and subsequent recovery will then be covered separately in appendices. The appendices are written for loss of DC during any mode. It should be noted that vital AC power associated with the lost DC bus may also be lost and corrective actions are taken in this procedure.

2. SYMPTOMS

1.1 Possible Annunciator Alarms

1.1.1 PK20-18 "125V DC Panels 11 (21), 12 (22), or 13 (23)"

(0122) "125V DC PNL PD-11 (21) FDR BKR OPN OR UV"

(0202) "125V DC PNL PD-12 (22) FDR BKR OPN OR UV"

(0278) "125V DC BUS 13 (23) SYS UV"

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER OP AP-23
REVISION 11
PAGE 2 OF 17
UNITS 1 AND 2**

TITLE: Loss of Vital DC Bus

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

1. VERIFY Reactor Tripped:

- a. Reactor trip and bypass breakers open
- b. Rod bottom lights lit
- c. Neutron flux decreasing
- d. Implement EP E-0, Rx Trip or SI, and continue in this procedure at Step 2

2. After 30 Seconds, Verify Main Generator Not Motoring:

- a. PCBs OPEN
- b. Either 86G1 (86G2) or 86G11 (86G21) tripped, (VB4)

Initiate manual unit trip.

CAUTION: Personnel shall wear protective flash gear if actuating the Exciter Field Breaker as described in the following step.

3. After Main Generator Trips, Verify No Gen. Field Voltage on CC3:

Locally trip Exciter Field Breaker.

CAUTION: Since the MFW pump recirc valves fail closed and MFP trip solenoids lose power, immediate action is required to runback and locally trip running MFPs to prevent over-pressurization.

4. Verify Both MFPs Tripped:

Remove MFP from service:

- 1. Verify MFP turbine speed is at minimum.
- 2. Locally trip MFP.

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER OP AP-23
REVISION 11
PAGE 9 OF 17
UNITS 1 AND 2**

TITLE: Loss of Vital DC Bus

APPENDIX B

LOSS OF VITAL DC BUS 12 (22)
RECOVERY AND RESTORATION

SYMPTOMS

- a) Train B FW Reg solenoids de-energize, fail closed, plant trips on Lo-Lo SG level
- b) 35% and 40% Steam Dumps fail closed, 10% Steam Dumps functional
- c) MFP #2 trips, MFP #1 running with recircs failed closed, manual trip of MFPs required
- d) PCBs fail to open, DEHC sense PCBs open and drive Gov Valvs closed which may cause Pzr Hi Pressure Trip if this occurs from high power level, Unit trip required
- e) 4kV Bus G does not transfer to startup
 - PY-12 lost with Bus G
 - Voltage transient of PY-14 during transfer to startup could result in 2/4 SI and MSLI
- f) 4kV and 12kV Bus Es fail to transfer to startup
- g) Letdown isolated, Excess LD not available, Pressurizer level controlled by start/stop of charging pump
- h) PORV-455C failed closed
- i) Pzr Htrs Groups 3 and 4 unavailable due to loss of power, vital backup power available for Group 3 only
- j) No control of D/G 1-2

INSTRUCTIONS

CAUTION: Reenergizing a DC bus without first stripping the bus of its loads may result in unexpected adverse equipment operation.

1. Evaluate plant status and take appropriate action for loss of the following components/functions:
 - a. CVCS-8471 and Charging Injection are the only boration flowpaths available when 480V Bus G is deenergized.
 - b. If the Turbine Driven AFW pump 1-1 (2-1) is needed:
 - 1) FCV-95, Steam Isolation valve, will have to be manually opened.
 - 2) Turbine Driven AFW pump motor operated valves LCV-106, 107, 108, and 109 will be deenergized if 480V bus G is lost and personnel will have to be dispatched locally to control AFW flow.
 - c. Place PY17 on backup supply from 480V bus F to keep Rotary Air Compressor master unloader energized if PY-17 is de-energized.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP E-0
REVISION 30
PAGE 3 OF 33

TITLE: Reactor Trip or Safety Injection

UNIT 1

ACTION/EXPECTED RESPONSE

3. VERIFY Vital 4KV Bus Status:

- a. Verify ALL vital 4KV buses -
ENERGIZED

RESPONSE NOT OBTAINED

- a. Perform the following:

- 1) IF All vital buses are
deenergized,
THEN GO TO EOP ECA-0.0,
LOSS OF ALL VITAL
AC POWER.
- 2) IF Only one vital bus is
energized,
THEN IMPLEMENT EOP ECA-0.3,
RESTORE 4KV BUSES.
- 3) IF Two vital buses are
energized,
THEN Perform the following:
 - o REFER TO EOP ECA-0.3,
RESTORE 4KV BUSES,
while continuing in this
procedure.
 - o REFER TO Appendix A for
guidance on available ESF
equipment.

*** UNCONTROLLED PROCEDURE – DO NOT USE TO PERFORM WORK or ISSUE FOR USE ***

PACIFIC GAS AND ELECTRIC COMPANY
 DIABLO CANYON POWER PLANT

NUMBER EOP ECA-1.1
 REVISION 18
 PAGE 3 OF 31

TITLE: Loss of Emergency Coolant Recirculation

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

5. <u>CHECK RWST Level GREATER THAN 4%</u>	GO TO Step 29 (Page 15).
---------------------------------------------------	--------------------------

6. **DETERMINE Containment Spray Requirements:**

- a. Determine number of Spray Pps required using table:

RWST LEVEL	CONTAINMENT PRESSURE	TOTAL NUMBER OF FAN COOLERS RUNNING	NUMBER OF SPRAY PUMPS REQUIRED
GREATER THAN 33%	GREATER THAN 47 PSIG	N/A	2
	BETWEEN 22 PSIG AND 47 PSIG	2 or LESS	2
		3 or 4	1
	ALL	0	
	LESS THAN 22 PSIG	N/A	0
LESS THAN 33%	GREATER THAN 22 PSIG	3 or LESS	1
		4 or 5	0
	LESS THAN 22 PSIG	N/A	0
LESS THAN 4%	N/A	N/A	0

- b. Spray Pps running - EQUAL TO NUMBER REQUIRED

- b. Reset Spray Pps and manually operate Spray Pps as necessary to obtain number required.

IF Containment Pressure increases,
THEN Start additional Containment Spray Pps.

IF Spray Pps CANNOT be operated,
THEN Continue efforts to establish Containment Fan Cooler Operation.

*** UNCONTROLLED PROCEDURE – DO NOT USE TO PERFORM WORK or ISSUE FOR USE ***

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP ECA-1.1
REVISION 18
PAGE 22 OF 31

TITLE: Loss of Emergency Coolant Recirculation

UNIT 1

APPENDIX A

BLACKOUT EMERGENCY LOADING OF VITAL BUSES

1. If the vital buses lose voltage prior to resetting the Safety Injection Signal, the vital buses will automatically load the vital equipment given below. Verify the equipment has been loaded by observing breaker lights on the control board.
2. If the vital buses lose voltage after the Safety Injection Signal has been reset, load or verify loaded the equipment given below onto the vital buses manually. Allow approximately 4 seconds between loading of each piece of equipment onto a given vital bus. Load or verify that the CFCU are running in Low Speed.

<u>VITAL BUS F</u> (D/G No. 3)	<u>VITAL BUS G</u> (D/G No. 2)	<u>VITAL BUS H</u> (D/G No. 1)
<u>CCP No. 1</u> <u>SI Pp No. 1</u> <u>CCW Pp No.1</u> <u>ASW Pp No.1</u> <u>AFW Pp No.3</u> <u>CFCU No. 1</u> <u>CFCU No. 2</u> <u>MCC F</u>	<u>RHR Pp No. 1</u> <u>CCW Pp No. 2</u> <u>ASW Pp No. 2</u> <u>CCP No. 2</u> <u>CFCU No. 3</u> <u>CFCU No. 5</u> <u>MCC G</u>	<u>SI Pp No. 2</u> <u>RHR Pp No. 2</u> <u>CCW Pp No. 3</u> <u>AFW Pp No. 2</u> <u>CFCU No. 4</u> <u>MCC H</u>

3. Load the Containment Spray Pps only if they were running prior to the blackout.

<u>VITAL BUS G</u>	<u>VITAL BUS H</u>
<u>Contmt Spray Pp No. 1</u>	<u>Contmt Spray Pp No. 2</u>

Mark
QuestionPrint
RecorNew
Search

Exit

The following sequence of events occurs:

"Unit 2 was operating at 100% power

"Small break LOCA occurred 25 minutes ago.

"The Team is currently implementing ECA 1.1, Loss of Emergency Coolant Recirculation due to loss of recirculation capability.

"RCS Pressure is 450 psig

"Containment pressure is 4 psig.

Given the attached reference from ECA 1.1, Loss Of Containment Sump Recirculation, which of the following indicates the REQUIRED correct combination of Containment Fan Cooler Units and Containment Spray Pumps that are required to be operating under these conditions?

D.0 FCU, 2 Spray Pumps

A.4 FCUs, 0 Spray Pumps

B.1 FCU, 1 Spray Pump

C.0 FCU, 1 Spray Pump

Unit 1 is recovering from a dropped control rod in accordance with OP AP-12C, DROPPED CONTROL ROD. While attempting to recover the dropped rod, a second control rod drops into the core. Your actions are to:

- A. Manually trip the reactor and GO TO EP E-O, REACTOR TRIP OR SAFETY INJECTION.
- B. Initiate a technical specification shutdown in accordance with Tech. Specs. 3.1.3.1.d.
- C. Terminate rod movement, do NOT reset any URGENT alarms, and contact I&C immediately to investigate the problem.
- D. Terminate rod motion and call the Plant Reactor Engineering Group for guidance.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

9934	Explain the operator actions if more than 1 control rod drops
------	---------------------------------------------------------------

Reference Id: P-5466
 Must appear: No
 Status: Active
 User Text: 9934.010013
 User Number 1: 0000003.40
 User Number 2: 0000004.60
 Difficulty: 2.00
 Time to complete: 3
 Topic: AP-12C, Actions to be done if two rods drop.
 Cross Reference: LPA-12, AP-12C
 Comment: Reviewed 11/8/96 jpsj

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP AP-12C
REVISION 10
PAGE 2 OF 13
UNITS 1 AND 2

TITLE: Dropped Control Rod

ACTION/EXPECTED RESPONSE

1. **ONLY One Control Rod Dropped**
2. **PLACE Rod Control in MANUAL**
3. **STOP Any Load Change In Progress AND Allow Conditions To Stabilize:**
 - a. Reactor is critical.

RESPONSE NOT OBTAINED

Trip the Reactor and GO TO EOP E-0, REACTOR TRIP OR SAFETY INJECTION

- 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

NOTE: The initial determination of SDM (STP R-19) is simplified if dilution is not used to raise temperature.

4. **STOP Any RCS Boration or Dilution In Progress**
5. **ADJUST Turbine Load To Match TAVG AND TREF**
6. **CHECK Axial Flux Difference Within Tech Spec Limits**
7. **Calculate OPTR per STP R-25:**
 - a. Verify LESS than 1.02.
8. **VERIFY SDM within COLR limits:**
 - a. Perform STP R-19 Data Sheet 4, within 1 hour
9. **VERIFY Rod Control System Had No Urgent Failure:**
 - a. Verify Rod Cont Urgent Failure (PK03-17) - OFF

Refer to Tech Spec 3.2.3.

- a. Refer to Tech Spec 3.2.4.A.

Refer to Tech Spec 3.1.1 and 3.1.4

- a. If PK03-17 is ON, THEN
 1. Do not attempt to move rods or reset the Urgent Failure.
 2. Contact maintenance for trouble shooting.
 3. Refer to AR PK03-17, ROD CONT URGENT FAILURE.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP AP-12C
REVISION 10A
PAGE 3 OF 11
UNITS 1 AND 2

TITLE: Dropped Control Rod

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

10. 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 maintenance to initiate troubleshooting and repairs

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.

11. CONTACT Reactor Engineering Regarding the Dropped Rod to Obtain:^{T30977}

- a. Guidance on rate of control rod movement during recovery. Inform Reactor Engineering if fuel has not been conditioned at the current power level and estimated time rod has been misaligned.
- b. Power level at which recovery should be performed
- c. Use of other control rods during recovery.

12. ESTABLISH Initial Recovery Conditions:

- | | |
|--------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|--------------------------------------------------------------------------------------------------------------------------------------------------|
| <ol style="list-style-type: none"> a. Check the time since the rod dropped - LESS THAN ONE hour 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. Adjust T_{AVG} 1.0 to 1.5°F below T_{REF} by inserting control bank rods as necessary | <ol style="list-style-type: none"> a. Reduce reactor power to LESS THAN 50%. Continue with Step 12.c when power is LESS THAN 50%. |
|--------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|--------------------------------------------------------------------------------------------------------------------------------------------------|

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER OP AP-12C
REVISION 10A
PAGE 4 OF 11
UNITS 1 AND 2**

TITLE: Dropped Control Rod

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

13. 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 except the dropped rod. (Lift Coil Disconnect Cabinet 115' Elev Aux Bldg)
- e. Check dropped rod is in a control bank
- e. GO TO Step 13.g (page 5).

NOTE: A ROD LO LO INSERTION LIMIT ALARM (PK03-14) may occur during the performance of this step.

- f. Locally reset the pulse to analog converter to ZERO for the affected control bank as follows: (P/A Converter Cabinet, 115' Elev, Aux Bldg)
 - 1) Select the affected bank on the digital display switch
 - 2) Record the position displayed
Bank _____
Position _____
 - 3) Hold the AUTO-MANUAL switch in the MANUAL position
 - 4) Depress the DOWN push button the required number of times to rezero the digital display
 - 5) Release the AUTO-MANUAL switch and verify it spring returns to AUTO position
- g. UPDATE the Plant Process Computer Rod Bank Step Count to ZERO for the affected Bank. (Refer to Appendix B for address)

DIABLO CANYON POWER PLANT
ABNORMAL OPERATING PROCEDUREUNITS **1 & 2**OP AP-21
REV. 10
PAGE 1 OF 4**Irradiated Fuel Damage**05/06/03
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

This procedure gives guidance in the event an irradiated fuel assembly is damaged during fuel handling or is found to have existing damage sustained from the previous cycle.

2. SYMPTOMS

- 2.1 Refueling crew observation of fuel damage.
- 2.2 While in the process of withdrawing a fuel assembly from the core, the refueling crew might observe release of gas bubbles from the fuel assembly with the following possible additional symptoms:
 - 2.2.1 Drag withdrawal limit exceeded (Dillon Cell at Manipulator Crane Control Console).
 - 2.2.2 Red overload light is ON (Manipulator Crane Control Console).
 - 2.2.3 Null meter indicates excessive positive drag accompanied by an audible alarm (Manipulator Crane Control Console).
 - 2.2.4 Auto Stop of upward hoist movement.
- 2.3 Local alarm/revolving red beacon light from area radiation monitors or portable continuous air monitors (e.g., SPING-3A/AMS-3/IM-11), as applicable.
- 2.4 Fuel Handling Building (FHB) evacuation horn is automatically sounded, if fuel handling incident is in the FHB.
- 2.5 Possible Main Annunciator Alarms:
 - 2.5.1 FHB HIGH RADIATION RE-58 and 59 (PK11-10)
 - 2.5.2 HIGH RADIATION (PK11-21)
 - 2.5.3 CONTMT VENT ISOLATION (PK02-06)
 - 2.5.4 If in Unit 1
CONTMT RADIATION (PK11-19)
 - 2.5.5 If in Unit 2
HI-LEVEL RAD MONITOR SYSTEM (PK11-19)
- 2.6 Possible Containment Ventilation Isolation (CVI) if fuel handling incident is in Containment.
- 2.7 FHB ventilation transfers to Iodine removal mode if incident is in FHB. (As this is an expected action along with the automatic sounding of the FHB evacuation horn whenever RE-58 or 59 alarm is activated, Control Room call to the crew is to confirm the incident.)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. 1 gpm unidentified LEAKAGE;
- c. 10 gpm identified LEAKAGE;
- d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3*, and 4*.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

* For MODES 3 and 4, if steam generator water samples indicate less than the minimum detectable activity of 5.0 E-7 microcuries/ml for principal gamma emitters, the leakage requirement of specification 3.4.13.d. may be considered met.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

LEAKAGE
(continued)

-
3. *Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System.*

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

#

1.00

A LBLOCA has occurred, and the ECCS system is in the Cold Leg Recirc phase. Containment radiation is noted to be 7 R/hr on RM-30/31. The EEC has identified the status of the CSFSTs and the TSC has recommended starting a Containment Spray pump. Which of the following should be done:

- A. Refer to EOP E-1.3, Transfer to Cold Leg Recirc, for instructions.
- B. Start the containment spray pumps.
- C. Stop the iodine removal units, then start the containment spray pumps.
- D. Return to the procedure and step in effect.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

68453	Explain the basic principles of operation for the Containment Fan Cooling Units.
69406	Explain the basic principles of operation for the Containment Systems.
6177	State the means of reducing containment radiation levels

Reference Id: P-49251
 Must appear: No
 Status: Active
 User Text:
 User Number 1:
 User Number 2:
 Difficulty: 3.00
 Time to complete: 2

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
EMERGENCY OPERATING PROCEDURE

NUMBER EOP FR-Z.3
REVISION 9
PAGE 1 OF 3
UNIT

TITLE: Response to High Containment Radiation Level

1

06/29/06

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

- 1.1 This procedure provides actions to respond to High Containment Radiation Level.
- 1.2 The major actions in EOP FR-Z.3 are:
 - Verify containment ventilation isolation,
 - Verify Containment Fan Cooler Units are in service
 - Notify the plant engineering staff of containment radiation level.

2. VERIFY ENTRY CONDITION FOR EOP FR-Z.3

- 2.1 EOP F-0.5, YELLOW Condition

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP FR-Z.3
REVISION 9
PAGE 2 OF 3
UNIT 1

TITLE: Response to High Containment Radiation Level

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

- 1. CHECK Containment Vent Isol:
 - a. Containment Vent Isol portion of Monitor Light Box B:
 - Red Activated Light - ON
 - White Status Lights - OFF

- Do one of the following:
- Manually Actuate CONTMT ISOL PHASE A
- OR
- Manually Close the CVI Vlvs with White Status Lights - ON
-

CAUTION: The Iodine Removal Units must not be placed in service during a Loss of Primary OR Secondary Coolant.

2. CHECK Containment Fan Cooler Units, RUNNING IN SLOW SPEED:

- a. Monitor Light Box C:
 - Red Activated Light - ON
 - White Status Light - OFF

NOTE: If the TSC recommends starting or restarting Containment Spray to reduce the Radiation Level and the ECCS System is in the Recirc Mode, REFER TO EOP E-1.3, TRANSFER TO COLD LEG RECIRCULATION.

3. NOTIFY TSC Of Containment Radiation Level To Obtain Recommended Action:

4. RETURN TO Procedure And Step In Effect

- END -

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP FR-Z.3
REVISION 9
PAGE 3 OF 3
UNIT 1

TITLE: Response to High Containment Radiation Level

3. APPENDICES

None

4. ATTACHMENTS

None

Previous exam question

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #		003 A2.05
	Importance		2.8

The crew is making preparations to start a RCP using Attachment B in E-0.2, "Natural Circulation Cooldown". The CO reports Seal Leakoff flow is low. All other conditions for starting the RCP are met.

Which of the following actions should the SFM direct the operator to perform?

- A. Start the RCP.
- B. Increase VCT level.
- C. Increase charging flow.
- D. Decrease VCT pressure.

Proposed Answer:

- D. Decrease VCT pressure.

Explanation:

A incorrect, this is not a procedure in which RCPs are started if normal conditions are not met.

B incorrect, increasing VCT level will increase VCT pressure, which will further decrease seal leakoff flow.

C incorrect, increasing charging will not affect appreciably affect seal leakoff.

D correct, decreasing VCT pressure will decrease the DP and increase seal leakoff flow.

Technical Reference(s):

E-0.2, Natural Circulation Cooldown, attachment B, Restart of Reactor Coolant Pump

Proposed references to be provided to applicants during examination: none

Learning Objective: 4892 - State the cause/effect relationship between VCT and RCPs

Question Source:

New X

Question History: Last NRC Exam N/A

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

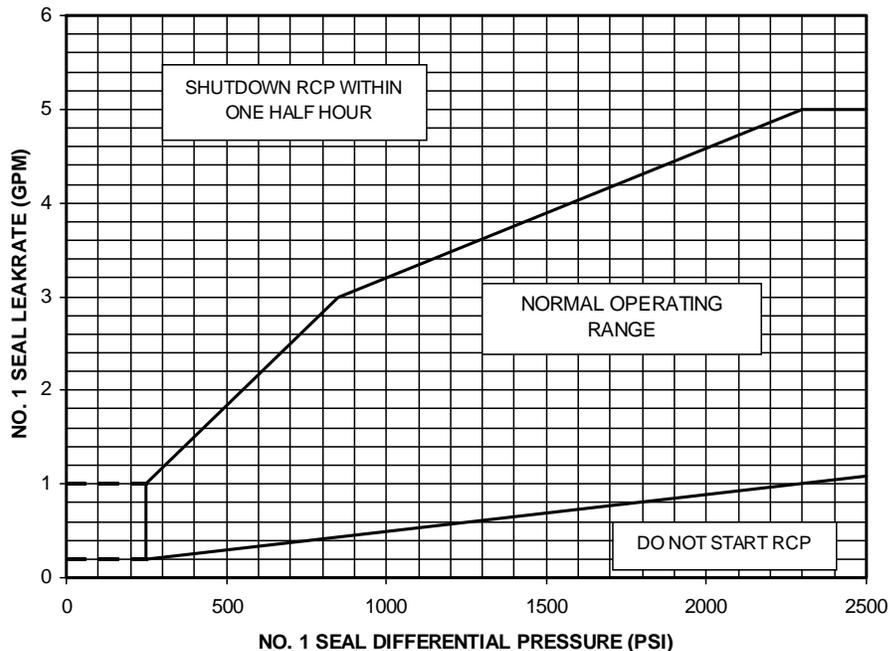
K/A: 003 A2.05 – Reactor Coolant Pump - Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of VCT pressure on RCP seal leakoff flows

Appendix B: Restart of Reactor Coolant Pump

NOTE: RCP No. 2 (preferred) or RCP No.1 should be given priority for purpose of normal PZR Spray. If RCP 1 is desired for PZR Spray then RCP 3 and 4 should also be started to provide sufficient PZR Spray DELTA-P.

CAUTION: RCP Seals may be damaged if RCP Seal Cooling was lost AND an RCP is started WITHOUT an RCP Seal Status Evaluation.

1. Start oil lift pump and run for 2 minutes.
2. Reset SI, Phase A or Phase B as necessary to provide RCP support systems.
3. Verify the following CCW valves open to RCP thermal barrier and oil coolers:
 - FCV-355 • FCV-356 • FCV-749 • FCV-363 • FCV-750 • FCV-357
4. Verify seal DP GREATER THAN 255 PSID, Depressurize VCT as necessary.
5. Verify Seal Injection flow between 8 GPM TO 13 GPM.
6. Verify Seal Leak Off flow WITHIN limits shown on graph.
7. Verify closed PCV-455A and B, Normal PZR Spray Vlvs, AUTO optional.
8. Start RCP. Observe RCP Pump Motor AMPS and FLOW to verify NORMAL operation.



3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2, MODE 3 with RCS pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u>	6 hours
	C.2 Reduce RCS pressure to \leq 1000 psig.	12 hours
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is \geq 814 ft ³ and \leq 886 ft ³ .	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is \geq 579 psig and \leq 664 psig.	12 hours

(continued)

STP I-

**1A APPL
MODE**

**TECH SPEC
REFERENCE**

A. CHECK / VERIFICATION FROM CONTROL ROOM

PERF

1 SR 3.5.1.2
SR 3.5.1.4

1. RCS Accumulator Volume

- a. Record current RCS accumulator volume and latest STP C-2 volume, then calculate volume change.

NOTE: PPC Quality Code color must not be light blue (cyan) for use in this surveillance.

Check the indicator used.

RCS Accumulator	RCS ACCUMULATOR VB2 IND OR PPC Point	CURRENT LEVEL %	C-2 LEVEL %	% Level Change
1-1	LI-950 [] LI950R []	_____ -	_____ =	_____
1-2	LI-951 [] LI951R []	_____ -	_____ =	_____
1-3	LI-952 [] LI952R []	_____ -	_____ =	_____
1-4	LI-953 [] LI953R []	_____ -	_____ =	_____
	LI-954 [] LI954R []	_____ -	_____ =	_____
	LI-955 [] LI955R []	_____ -	_____ =	_____
	LI-956 [] LI956R []	_____ -	_____ =	_____
	LI-957 [] LI957R []	_____ -	_____ =	_____

- b. Verify all current RCS accumulator levels are $\geq 59\%$ and $\leq 75\%$. [52% to 82%]

- c. Verify all RCS accumulator level changes are $\leq +3\%$ [5.6%].

- d. If any operational draining or filling of an accumulator occurs or if level increase is unacceptable, request chemistry to perform STP C-2 within 6 hours for affected accumulators.

N/A [] _____

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK02-05
Rev. 18A
Page 1 of 5

**ACCUM
PRESSURE
HI-LO**

07/06/05

Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
351	Accum 1-1 Press Lo PC-961B	PC961B	< 595.5 psig	2.1
404	Accum 1-3 Press Lo PC-965B	PC965B	< 595.5 psig	2.1
405	Accum 1-4 Press Lo PC-967B	PC967B	< 595.5 psig	2.1
409	Accum 1-2 Press Lo PC-963B	PC963B	< 595.5 psig	2.1
1381	Accum 1-1 Press Hi PC-961A	PC961A	> 647.5 psig	2.1
1386	Accum 1-2 Press Hi PC-963A	PC963A	> 647.5 psig	2.1
1388	Accum 1-1 Press Hi PC-960A	PC960A	> 647.5 psig	2.1
1389	Accum 1-1 Press Lo PC-960B	PC960B	< 595.5 psig	2.1
1395	Accum 1-2 Press Hi PC-962A	PC962A	> 647.5 psig	2.1
1396	Accum 1-2 Press Lo PC-962B	PC962B	< 595.5 psig	2.1
1406	Accum 1-3 Press Hi PC-965A	PC965A	> 647.5 psig	2.1
1407	Accum 1-4 Press Hi PC-967A	PC967A	> 647.5 psig	2.1
1415	Accum 1-3 Press Hi PC-964A	PC964A	> 647.5 psig	2.1
1416	Accum 1-3 Press Lo PC-964B	PC964B	< 595.5 psig	2.1
1417	Accum 1-4 Press Hi PC-966A	PC966A	> 647.5 psig	2.1
1418	Accum 1-4 Press Lo PC-966B	PC966B	< 595.5 psig	2.1

2. OPERATOR ACTIONS**2.1 General Actions (All Inputs)**

NOTE: The high and low pressure alarm setpoints are more accurate than the Control Board instrument readings.

- 2.1.1 Check both pressure indicators to confirm the alarm and the high or low pressure condition. []
- 2.1.2 REFER TO TS 3.5.1 to determine accumulator OPERABILITY. []
- 2.1.3 IF Annunciator PK02-10, "ACCUM LEVEL HI-LO," is ON, THEN GO TO AR PK02-10, "Accum Level Hi-Lo," for further action. []
- 2.1.4 GO TO the applicable section:
- | <u>IF</u> alarm is due to: | <u>THEN</u> GO TO: | |
|----------------------------|--------------------|-----|
| High Pressure | Section 2.2 | [] |
| Low pressure | Section 2.3 | [] |

2.2 High Pressure (Inputs 1381, 1386, 1388, 1395, 1406, 1407, 1415, 1417)

- 2.2.1 IF high pressure is due to a level increase AND attempting to lower pressure by draining is desired, THEN perform the following:
- a. OPEN accumulator drain to RCDT valve on affected accumulator until accumulator level reaches approximately 67%: []
 - Accumulator 1: SI-1-8876A
 - Accumulator 2: SI-1-8876B
 - Accumulator 3: SI-1-8876C
 - Accumulator 4: SI-1-8876D
 - b. Direct chemistry to sample accumulator. []
 - c. Determine if either of the following procedures should be performed:
 - STP V-5A1, "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8819 A-D and 8956 A-D" []
 - STP V-5A2, "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8948 A-D and 8818 A-D" []

- 2.2.2 Depressurize the affected accumulator as necessary PER OP B-3B:I, "Accumulators - Fill and Pressurize". []

NOTE: A malfunction in SI-1-PCV-199, N₂ Pressure Regulator To Accumulators, could cause high pressure nitrogen to enter the accumulator through a leaking fill and vent valve.

- 2.2.3 IF after depressurizing, accumulator pressure continues to rise, THEN dispatch an Operator to check SI-1-PCV-199, N₂ Pressure Regulator To Accumulators, for possible malfunction. []

2.2.4 Probable Causes

- Accumulator and/or associated gas piping/valves leakage
- Accumulator level increase due to temperature change

2.3 Low Pressure (Inputs 351, 404, 405, 409, 1389, 1396, 1416, 1418)

- 2.3.1 IF low pressure is due to a level drop AND attempting to raise pressure by filling is desired, THEN fill the affected accumulator PER OP B-3B:I, "Accumulators - Fill and Pressurize". []

- 2.3.2 Attempt to pressurize the affected accumulator as necessary PER OP B-3B:I, "Accumulators - Fill and Pressurize". []

- 2.3.3 IF the nitrogen regulator did not raise accumulator pressure THEN perform the following:
- a. Check HP nitrogen supply pressure at Aux Control Board or at 115' nitrogen bottle racks. []
 - b. Verify CLOSED all accumulator nitrogen fill and vent valves: []
 - Accumulator 1: SI-1-8875A
 - Accumulator 2: SI-1-8875B
 - Accumulator 3: SI-1-8875C
 - Accumulator 4: SI-1-8875D
 - c. CLOSE SI-1-8880, Accumulator Nitrogen Fill Header Isolation Valve. []

CAUTION: Opening HCV-943 fully reduces pressure very quickly and can result in pressure being less than the Technical Specification minimum.

- d. Cycle SI-1-HCV-943, Accumulator Nitrogen Vent Valve, as necessary to lower pressure downstream of the nitrogen regulator in small slow increments. []

CAUTION: Do not open more than one accumulator nitrogen fill and vent valve (8875A-D) valve at the same time. This cross-connects the accumulator nitrogen supplies and may result in a loss of design basis injection capability.^{T35830}

e. Perform the next two steps in rapid succession:

1. OPEN SI-1-8880, Accumulator Nitrogen Fill Header Isolation Valve. []
2. OPEN the nitrogen fill and vent valve on the affected accumulator: []
 - Accumulator 1: SI-1-8875A
 - Accumulator 2: SI-1-8875B
 - Accumulator 3: SI-1-8875C
 - Accumulator 4: SI-1-8875D

f. WHEN pressure returns to normal,
THEN CLOSE appropriate accumulator nitrogen fill and vent valve. []

2.3.4 IF accumulator pressure continues to drop,
THEN dispatch an Operator to check the following for leakage: []

- Safety valve
- Manual vent
- Blank flange

2.3.5 Probable Causes

- Accumulator and/or associated gas piping/valves leakage
- Accumulator level decrease due to temperature changes
- Inadvertent opening of accumulator vent valve
- Accumulator discharge
- Sampling of accumulator volume

3. AUTOMATIC ACTIONS

NONE

4. REFERENCES

- 4.1 AR PK02-10, "Accumulator Level Hi-Lo"
- 4.2 OP B-3B:I, "Accumulators - Fill and Pressurize"
- 4.3 501122, "Schematic Diagram – Main Annunciator" (Electrical Drawing Section 8)

*** ISSUED FOR USE BY: _____ DATE: _____ EXPIRES: _____ ***

DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

UNIT **1**

AR PK02-10
Rev. 17A
Page 1 of 4

**ACCUM
LEVEL
HI-LO**

07/20/05

Effective Date

QUALITY RELATED

1. **ALARM INPUT DESCRIPTION**

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
352	Accum 1-1 Lvl Hi LC-950A	LC950A	> 72.6%	2.1
353	Accum 1-2 Lvl Hi LC-952A	LC952A	> 72.6%	2.1
406	Accum 1-3 Lvl Hi LC-954A	LC954A	> 72.6%	2.1
407	Accum 1-4 Lvl Hi LC-956A	LC956A	> 72.6%	2.1
1382	Accum 1-1 Lvl Hi LC-951A	LC951A	> 72.6%	2.1
1383	Accum 1-1 Lvl Lo LC-951B	LC951B	< 60.8%	2.1
1384	Accum 1-2 Lvl Hi LC-953A	LC953A	> 72.6%	2.1
1385	Accum 1-2 Lvl Lo LC-953B	LC953B	< 60.8%	2.1
1390	Accum 1-1 Lvl Lo LC-950B	LC950B	< 60.8%	2.1
1391	Accum 1-2 Lvl Lo LC-952B	LC952B	< 60.8%	2.1
1408	Accum 1-3 Lvl Hi LC-955A	LC955A	> 72.6%	2.1
1409	Accum 1-3 Lvl Lo LC-955B	LC955B	< 60.8%	2.1
1410	Accum 1-4 Lvl Hi LC-957A	LC957A	> 72.6%	2.1
1411	Accum 1-4 Lvl Lo LC-957B	LC957B	< 60.8%	2.1
1419	Accum 1-3 Lvl Lo LC-954B	LC954B	< 60.8%	2.1
1420	Accum 1-4 Lvl Lo LC-956B	LC956B	< 60.8%	2.1

2. OPERATOR ACTIONS**2.1 General Actions (All Inputs)**

NOTE: The high and low level alarm setpoints are more accurate than The Control Board instrument readings.

- 2.1.1 Check both level indicators on the affected accumulator to confirm the alarm and the high or low level condition. []
- 2.1.2 REFER TO TS 3.5.1 to determine accumulator OPERABILITY. []
- 2.1.3 Verify CLOSED the following:
- a. NSS-1-9357A, Accumulator Sample Line Inside Containment []
 - b. NSS-1-9357B, Accumulator Sample Line Outside Containment []
 - c. Accumulator Drains To RCDT:
 - SI-1-8876A []
 - SI-1-8876B []
 - SI-1-8876C []
 - SI-1-8876D []
 - d. Accumulator Fill Lines From SI Pump:
 - SI-1-8878A []
 - SI-1-8878B []
 - SI-1-8878C []
 - SI-1-8878D []
- 2.1.4 Dispatch an Operator to verify CLOSED NSS-1-9352A, B, C, and D, Accumulator Sample Line Isolations (Aux Bldg, 100 ft Primary Sample Panel). []

- 2.1.5 IF accumulator level is high,
THEN return level to normal as follows:
- a. OPEN accumulator drain to RCDT valve on affected accumulator until accumulator level reaches approximately 67%: []
 - Accumulator 1: SI-1-8876A
 - Accumulator 2: SI-1-8876B
 - Accumulator 3: SI-1-8876C
 - Accumulator 4: SI-1-8876D
 - b. Direct chemistry to sample accumulator. []
 - c. Determine if either of the following procedures should be performed:
 - STP V-5A1, "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8819 A-D and 8956 A-D" []
 - STP V-5A2. "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8948 A-D and 8818 A-D." []
- 2.1.6 IF accumulator level is low,
THEN raise affected accumulator level PER OP B-3B:I, "Accumulators - Fill and Pressurize". []
- 2.1.7 IF PK02-05, "ACCUM PRESSURE HI-LO", alarm is ON,
THEN GO TO AR PK02-05, "Accum Pressure Hi-Lo". []
- 2.1.8 Probable Causes
- a. Low level
 - One or more open/leaking accumulator drains
 - One or more open/leaking sample valves
 - One or more open/leaking fill line valves
 - Accumulator system piping leakage
 - Accumulator discharge to Primary System
 - b. High level
 - Discharge check valve leakage
 - Fill line valve leakage

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take normal suction through valve MU-671 on the single suction line from the condensate storage tank (CST) (LCO 3.7.6) (this valve must remain open for the applicable accident analysis assumptions to be valid) and are capable of being aligned to the firewater storage tank (FWST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the condenser steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity, and the turbine driven pump provides 200% of the required capacity to the steam generators, with 100% capacity defined as the flow required to two steam generators during the AFW design basis accident analysis (loss of normal feedwater flow (Ref. 1)). The pumps are equipped with recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators, although each pump has the capability to be manually realigned to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

(continued)

BASES

BACKGROUND
(continued)

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with vital AC powered control valves. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The AFW System (both the one turbine-driven and two motor-driven AFW pumps) actuates automatically upon actuation of the anticipated transient without scram mitigating system actuation circuitry (AMSAC). The motor-driven pumps are additionally actuated by: (1) safety injection; (2) an associated bus transfer to the diesel generator signal; (3) a trip of both MFW pumps; or (4) steam generator water level—low-low in one of four SGs. The turbine-driven pump is additionally actuated by 12 kV bus undervoltage or steam generator low-low level in two of four SGs via ESFAS (LCO 3.3.2).

The AFW System is discussed in the FSAR, Section 6.5 (Ref. 1).

APPLICABLE
SAFETY
ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to at least two steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3% tolerance plus 3% accumulation within 1 minute after event initiation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and AFW spillage through feedwater line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

- a. Feedwater (FWLB) or Main Steam Line Break (MSLB); and
- b. Loss of MFW (the coincident loss of offsite power is a less limiting transient since RCP heat input is lost).

(continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

In addition, the minimum available AFW flow and system characteristics must be considered in the analysis of normal cooldown and small break loss of coolant accident (LOCA) due to their potential impact.

The AFW System is also designed for decay heat removal following a Steam Generator Tube Rupture (SGTR). As such the steam turbine driven AFW pump has redundant steam supplies to assure continued availability following a SGTR or MSLB event.

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the broken MFW header at the pump maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump when required to ensure an adequate feedwater supply to the steam generators during loss of power. Vital AC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of decay and residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses and having the third AFW pump powered by a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs. To assure steam turbine driven AFW pump operability via redundant steam supplies, steam traps 104, 105 and 106 on the supply lines must be operable or bypassed to ensure adequate condensate removal and check valves MS-5166 and MS-5167 must be operable.

The AFW System supply is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps,

(continued)

#

1.00

What is the MINIMUM acceptable auxiliary feedwater alignment to ensure that adequate feedwater flow is available to remove decay heat?

- A. One motor driven AFW pump delivering flow to two S/Gs
- B. One motor driven AFW pump delivering flow to one S/G
- C. Two motor driven AFW pumps delivering flow to two S/Gs
- D. One motor driven and the steam driven AFW pump delivering flow to 2 S/Gs

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

9694	Discuss Technical Specification bases
9694G	Discuss 3.7 Technical Specification bases

Reference Id: B-0401
Must appear: No
Status: Active
User Text: 9694.13ALLN
User Number 1: 000002.70

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER STP M-9A
REVISION 72
PAGE 9 OF 35
UNITS 1 AND 2**

TITLE: Diesel Engine Generator Routine Surveillance Test

INITIALS

- 10.7 There should be fuel oil in the priming tank. If there is not, the priming tank should be filled using the magnetic pump. Document problem in an AR. _____
- 10.8 The fuel oil pressure should increase to above 40 PSIG within 60 seconds of engine start. Gauge response is about 15 seconds. _____
- 10.9 Do not violate the following limits during normal steady-state operation:
 - 10.9.1 Maximum continuous generator current is 451 amperes. _____
 - 10.9.2 Maximum stator temperature is 240°F. _____
 - 10.9.3 Minimum lube oil pressure is 60 PSIG. _____
 - 10.9.4 Maximum lube oil temperature is 195°F. _____
 - 10.9.5 Power factor: 1.0 to 0.8 lag (see Appendix 8.1). _____
 - 10.9.6 Load: 2.60 MW at 0.8 PF (see Appendix 8.2 for maximum limits). _____
- NOTE:** The D/G may operate at > 2.5 and ≤ 2.75 MW at 0.8 PF for up to 2000 hours per year.
- 10.10 Whenever a D/G is paralleled to the grid and operating at full load, the MVAR output should be reduced prior to starting large loads. Diesel generators are highly responsive to starting large motor loads and could potentially exceed their KVA rating. The following good operating practices apply: (AR A0602120)
 - 10.10.1 If a large motor load will be started while the D/G is paralleled and at full load, reduce to 0.5 MVARs or less. _____
 - 10.10.2 These loads include 12KV pumps, condensate / booster pump set, or the #2 heater drain pump. _____
 - 10.10.3 After the load has been started, MVARs may be raised. _____
- 10.11 Normal shutdown of a D/G requires DC control power. If it becomes necessary to shutdown the D/G without control power, manually operate the trip lever on the northwest corner of the engine, forward of the fuel oil filters. _____
- 10.12 Do not operate more than one D/G at a time paralleled to any transformer (startup or unit auxiliary) in MODE 1, 2, 3, or 4 (SR 3.8.1.3 Note 3). _____
- 10.13 The applicable D/G MODE SEL switch on VB4 shall be in MANUAL prior to paralleling the D/G to the bus. _____

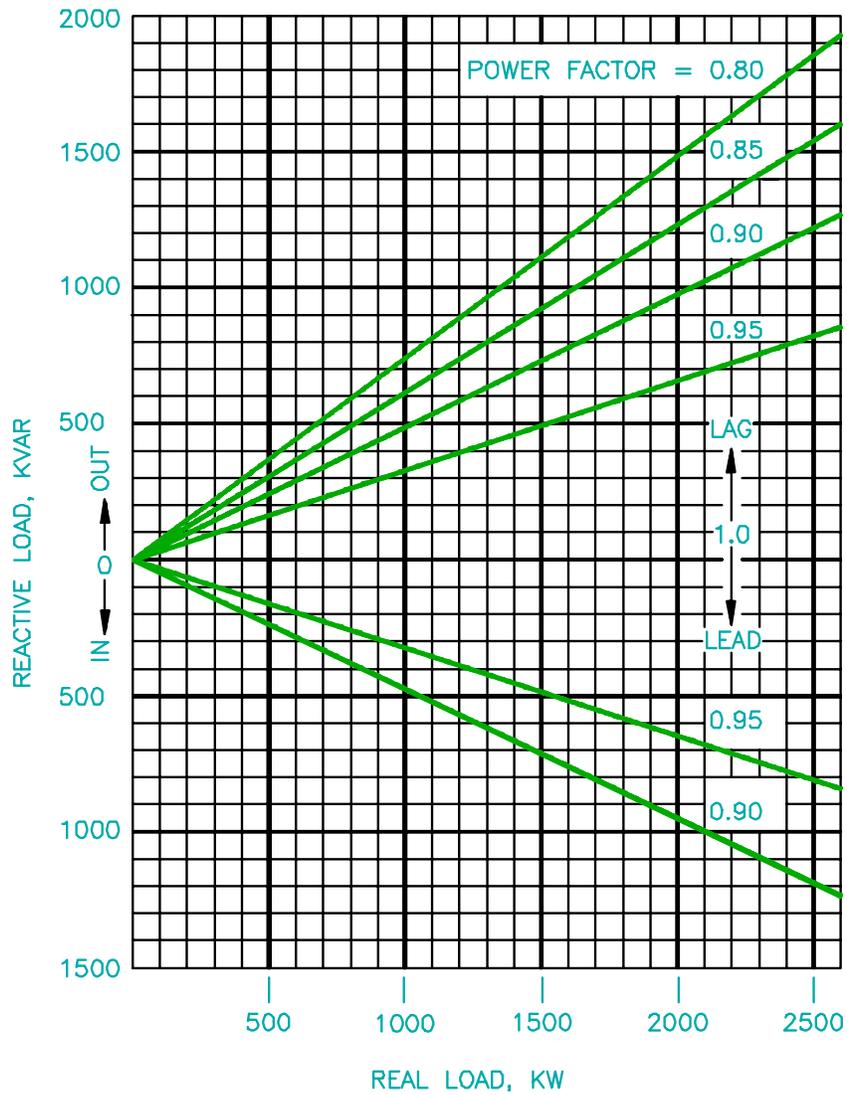
PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER STP M-9A
REVISION 73
PAGE 31 OF 30
UNITS 1 AND 2

TITLE: Diesel Engine Generator Routine Surveillance Test

APPENDIX 8.1

KW-KVAR, POWER FACTOR RELATIONS FOR DIESEL GENERATORS



3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be $\leq 120^{\circ}\text{F}$.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -1.0 psig and $\leq +1.2$ psig.

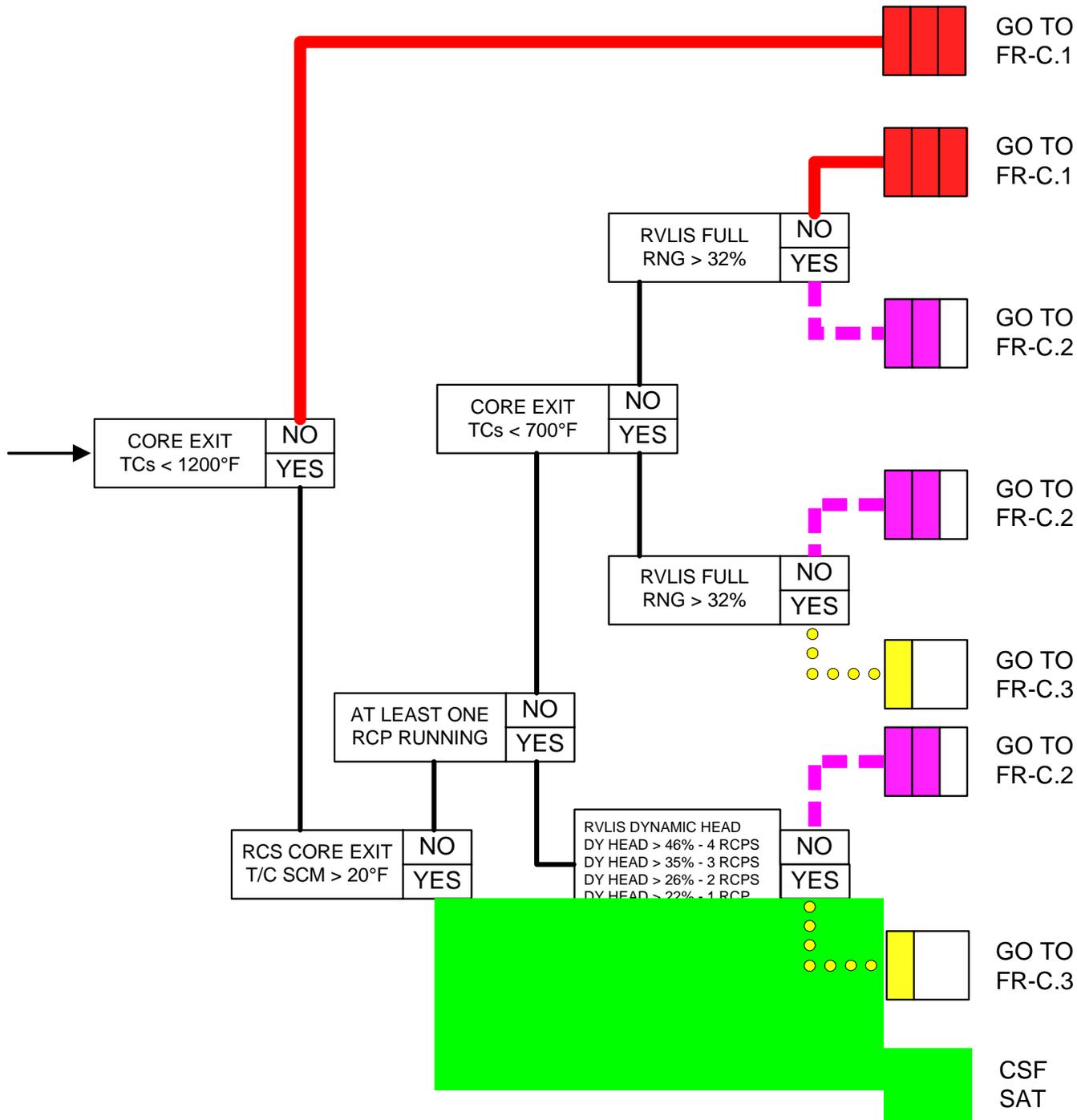
APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

F-0.2, Core Cooling

F-0.2 CORE COOLING



B 3.7.13 Fuel Handling Building Ventilation System (FHBVS) BASES

BACKGROUND

The FHBVS filters airborne radioactive particulates and radioactive iodine from the area of the fuel pool following a fuel handling accident. The FHBVS provides environmental control of temperature and humidity in the fuel pool area and for the AFW pump motors. The ventilation for the AFW pump motors is to provide cooling flow for EQ considerations, i.e., motor longevity. The ventilation is not required to function during an accident or for the few hours required to reach RHR conditions during a natural circulation cooldown.

The FHBVS consists of two independent and redundant trains. Each train consists of, an exhaust prefilter, high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and an exhaust fan. A third non-vital exhaust fan is used for normal operation and has only a prefilter and a HEPA filter. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal or loss of the normal exhaust fan E-4.

The FHBVS is a standby system, parts of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharge from the fuel handling building is isolated and the normal exhaust fan shuts down and the vital exhaust fans start and the stream of ventilation air discharges through the system filter trains. The prefilter removes any large particles in the air, to prevent excessive loading of the HEPA filter and charcoal adsorber.

The FHBVS is discussed in the FSAR, Sections 9.4.4 and 15.5 (Refs. 1, and 2, respectively) because it may be used for normal, as well as post (fuel handling) accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The FHBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident involving the handling of recently irradiated fuel. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FHBVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident. Due to radioactive decay, the

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the filled portions of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity, that have direct access to the reactor vessel, during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. The refueling boron concentration is sufficient to maintain shutdown margin (SDM) with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in MODE 6 is sufficient to maintain $k_{\text{eff}} \leq 0.95$ with the most reactive rod control assembly completely removed from its fuel assembly.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the principle system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with refueling grade borated water from the liquid hold up tanks or the refueling water storage tank.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation cooling in the RCS and assist in maintaining the boron concentration uniformity in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

(continued)

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND	<p><u>The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.</u></p>
APPLICABLE SAFETY ANALYSIS	<p>During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).</p> <p>The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within allowable limits (Refs. 4, and 5).</p> <p>Refueling cavity water level satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).</p>
LCO	<p>A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.</p>

(continued)

3.3.6

Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Not used				
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
	(a)	1 train	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Purge Radiation Gaseous and Particulate	1, 2, 3, 4	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	Per ODCM
	(a)	1	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	Per ODCM
4. Containment Isolation-SI	Refer to LCO 3.3.2, "ESFAS Instrumentation," Functions 1 and 3, for all initiation functions and requirements.			

(a) During movement of recently irradiated fuel assemblies within containment.

OP B-8D attachment 9.1

1. Temporary nozzle dams removed, or use approved by PSRC. N/A []
2. Applicable portions of the FHB and Containment have been classified as an FME area per AD4.ID6. All cleanliness requirements are in effect and being monitored by Maintenance.
3. Verify STP M-42 was performed within 100 hours of Rx head removal. (ECG 42.3)
4. After the Refueling Cavity has been filled, initiate a separate SFM Clearance and tag the following valves closed. CR NUMBER _____
Do **NOT** take credit for tags hung per OP L-6.
 - a. LWS-91 Refueling Cavity Drain Line
 - b. LWS-92 Refueling Canal Drain to RCDT
 - c. LWS-HCV-111 Refueling Canal Flushing Valve
 - d. The flange downstream of HCV-111 is installed, or the filter assembly flange isol valve is CAUTION tagged closed.
 - e. RCS-8032 Leak Detection Return to RCDT.
5. Portable radiation monitor(s) are available for use on the manipulator crane and spent fuel pool bridge crane, and a continuous air monitor is located in Containment (140' el.) and the Fuel Handling Building.

OP _____/_____/_____

(FHB) MECH _____/_____/_____

(CONT) MECH _____/_____/_____

ENG _____/_____/_____

OP _____/_____/_____

RP _____/_____/_____

ITEM NO.	DESCRIPTION	PRIOR TO CORE ALT. COMPLETE WITHIN	RESPONSIBLE DEPARTMENT INIT./DATE	REVIEWED (SFM) INIT. HOUR/DATE
6.	Verify all OP L-0 items for Mode 6/ Core Alterations are complete.	2 hours	OP ___/___	___ ___/___
7.	ECG 42.5, Verify <u>refueling cavity</u> water level >23 feet above irradiated fuel assemblies within the reactor. (elev. 126 ft. 6 in.)	2 hours	OP ___/___	___ ___/___
8.	ECG 42.2, Establish direct communications between the control room and the refueling stations.	1 hour	OP ___/___	___ ___/___
9.	Test containment evacuation alarm.	1 hour	OP ___/___	___ ___/___
10.	SRO for fuel handling operations only <u>and</u> licensed RO required for continuous monitoring of count rate data during CORE ALTERATIONS.		OP ___/___	___ ___/___
RCCA Unlatching Commenced _____/_____/_____			RCCA Unlatching Completed _____/_____/_____	
	TIME DATE		SFM TIME DATE	SFM

ITEM NO.	DESCRIPTION	RESPONSIBLE DEPT INITIALS/DATE	REVIEWED (SFM) INITIALS/DATE
NOTE: Prior to core unloading, verify Items 1 through 5 and 11 through 25 are current.			
11.	Fuel accountability tracking software for control of fuel assembly and insert locations during refueling operations are available for use in the appropriate areas (See TS6.ID2).	RX ENG _____/_____	_____/_____
12.	If handling recently irradiated fuel per TS 3.7.13 ensure FHB Ventilation is capable of supporting fuel movement by verifying the following are satisfied: N/A []	OP _____/_____	_____/_____
	a. Complete OP H-7:I, Attachment 9.2.		
	b. Active SFM clearance exists on the 140' partition wall locks and 115' roll-up door chains. CR NUMBER _____		
	c. The Refueling SRO or designee has possession of the keys to the roll-up doors to prevent them from being opened.		
13.	Verify an active SFM Admin Clearance exists on the source range speaker in containment.	OP _____/_____	_____/_____
14.	All critical personnel participating in the core unloading have been adequately trained for their part in the fuel handling operations including nuclear engineering/operations fuel handlers.	RX ENG _____/_____	_____/_____
		OP _____/_____	_____/_____
15.	Core unloading sequence completed and issued by reactor engineering to determine:	RX ENG _____/_____	_____/_____
	• Acceptable spent fuel storage locations (TS 3.7.17)		
	• The fuel movement sequence insures that the core is neutronically coupled to the two source range detectors to be used.		
16.	Verify that RCP D-220 Controls are in place (High Rad Areas locked for fuel transfer). Setting the controls too early may hinder maintenance activities.	RP _____/_____	_____/_____
17.	Perform a spot check of the Manipulator Crane indexing after upper internals removal. Check a minimum of two widely separated core locations.	OP _____/_____	_____/_____
18.	A test run of the fuel transfer system and manipulator crane should be made with the dummy assembly if possible.	OP _____/_____	_____/_____

ITEM NO.	DESCRIPTION	PRIOR TO UNLOADING COMPLETE WITHIN	RESPONSIBLE DEPARTMENT INIT./DATE	REVIEWED (SFM) INIT. HOUR/DATE	RETEST FREQUENCY	NEXT DUE DATE	(SFM) RETEST COMPLETED INIT. HOUR/DATE
19.	Procedure MP I-4.4-2A, Spent Fuel Pool crane load cell calibration.	6 months	ELE ___/___	___ ___/___	6 mo.	___	___ ___/___
20.	LT 13-2, Spent Fuel Pool Temperature Channel TIC-651 Calibration.	31 days	I&C ___/___	___ ___/___	92 days	___	___ ___/___
21.	STP M-27, Fuel handling interlocks.	7 days	OP ___/___	___ ___/___	N/A		
22.	Verify acceptability of the SFP cooling system status per Attachment 9.4 of OP B-8DS1.	12 hours	OP ___/___	___ ___/___	12 hours		See OP B-8DS1
23.	TS 3.9.7, Verify <u>refueling cavity</u> water level >23 feet over the reactor vessel flange. (elev. 138 ft. 2 in.)	2 hours	OP ___/___	___ ___/___			
24.	ECG 42.2, Establish direct communications between the control room and the refueling stations. a. Test containment evacuation alarm.						1 hour OP ___/___ C. O. Logs
25.	SRO for fuel handling operations only <u>and</u> licensed operator required for continuous monitoring of count rate data during CORE ALTERATIONS. MODE 6	OP ___/___	___ ___/___	ALL TIME			
Core Unloading Commenced		___/___		Core Unloading Completed	___/___	___	

Degraded Condenser**U1&2 OP AP-7****REV. 34****PAGE 9 OF 16**

SECTION C: TRAVELING SCREEN PROBLEM**4. SYMPTOMS**

- 4.1 Heavy kelp loading on traveling screens caused by storm conditions and/or increasing sea swell.
- 4.2 Possible Main Annunciator Alarms
BAR RACKS/SCREENS (PK13-01)

SECTION C: TRAVELING SCREEN PROBLEM (Continued)

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

CAUTION 1: If it appears that continuation of deliberate operator actions to stabilize the plant ON LINE will not be successful, manually trip the reactor and go to EOP E-0.

CAUTION 2: If more than one traveling screens stalls due to loading, severe damage is occurring.

CAUTION 3: The backflow from shutdown of an affected CWP could cause a large increase in loading on the screens of the CWPs still in service. The screens for the least affected pumps should be in high speed prior to the CWP shutdown.

CAUTION 4: If both CWPs are lost and the reactor is tripped, Attachment 6.1 should be implemented after EOP E-0 is performed.

NOTE: Screen D/P indication is available in the PPC. Turn on code "SCREENS."

1. CHECK Traveling Screen Status

a. CHECK traveling screen ΔP NOT NORMAL - Bar Racks/Screens (PK13-01) – ON

a. GO TO step 3 (Page 13)

- Verify traveling screens in HIGH speed.
- Dispatch an additional operator to the intake for assistance.
- Continuously monitor screen D/Ps locally or on the PPC.
- Verify Screen Wash Spray Flow NORMAL (spray pattern for each screen NORMAL, spray header pressure GREATER THAN 90 psig.)
- Verify SCW supply lined up to the most reliable source per plant conditions (Ref OP F-1:II and OP F-1:VI).

IF Screen ΔP remains continuously greater than 8 inches,

THEN Commence Unit load reduction to 50% or below.

SECTION C: TRAVELING SCREEN PROBLEM (Continued)

ACTION / EXPECTED RESPONSE**RESPONSE NOT OBTAINED****1. CHECK Traveling Screen Status**

(Continued)

- b. CHECK Screens NOT Overloaded:
- Screen ΔP LESS THAN 70 inches
 - Each CWP in service has at least two operating screens.

- b. IF Two CWPs are in operation,
THEN

- 1) Reduce unit load as necessary.
- 2) Verify least affected CWP screens in MANUAL and FAST.
- 3) Trip the CWP with heaviest screen loading.
- 4) Implement OP E-4:III for the tripped CWP.
- 5) Return step 1.b for reevaluation of screen status.

- IF Only one CWP in operation AND Reactor Power LESS THAN or EQUAL TO 25%,

THEN GO TO step 4 (Page 13)

- IF Only one CWP in operation AND Reactor Power GREATER THAN 25%,

THEN Perform the following:

- 1) Trip the Reactor,
- 2) Trip the remaining CWP,
- 3) GO TO EOP E-0

AND

Implement the following:

- a) Close the MSIVs,
- b) Adjust 10% steam dumps controllers to maintain S/G pressure LESS THAN OR EQUAL TO 1005 PSIG (8.38 turns),

c) GO TO step 5 (Page 14).

SECTION C: TRAVELING SCREEN PROBLEM (Continued)

ACTION / EXPECTED RESPONSERESPONSE NOT OBTAINED2. CHECK CWP Motor Amps –
NORMAL AND STABLE

- Approximately 560 AMPS
- LESS THAN 20 AMP swings

IF Two CWPs are in operation

THEN

- 1) Reduce unit load as necessary.
- 2) Verify least affected CWP screens in MANUAL and FAST.
- 3) Trip the CWP with heaviest screen loading.
- 4) Implement OP E-4:III for the tripped CWP.

IF One CWP is in operation AND Reactor Power LESS THAN or EQUAL TO 25%,

THEN GO TO step 4 (Page 13)

IF One CWP is in operation AND Reactor Power GREATER THAN 25%,

THEN Perform the following:

- 1) Trip the Reactor,
- 2) Trip the remaining CWP,
- 3) GO TO EOP E-0

AND

Implement the following:

- a) Close the MSIVs,
- b) Adjust 10% Steam Dumps Controllers as needed to maintain S/G pressure LESS THAN EQUAL TO 1005 PSIG (8.38 turns),
- c) GO TO step 5 (Page 14).

SECTION C: TRAVELING SCREEN PROBLEM (Continued)

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

3. CONTINUE To Monitor Screen And CWP Performance:

- a. Implement OP O-28
- b. Notify Operations Manager
- c. Screen and CWP performance per steps 1 and 2 normal
- d. Return to procedure step in effect

c. Return to step 1 (page 10) for reevaluation

4. PERFORM Reduced Load Shutdown:

- a. Announce rapid shutdown on the Public Address system
- b. Trip the turbine
 - Verify all Turb Stm Stop Vlvs – CLOSED
- c. Trip the running CWP
- d. Verify generator trip and vital bus transfer 30 sec. after trip
- e. Verify control rods inserting at maximum rate in AUTO or MANUAL to shutdown the reactor
 - Complete shutdown of the Reactor by inserting all Control and Shutdown Banks per OP A-3:II
- f. Close the MSIVs
- g. Verify 10% Steam Dump Controllers are set to maintain S/G pressure LESS THAN OR EQUAL TO 1005 PSIG (8.38 turns)

b. Trip the Reactor AND GO TO EOP E-0.

e. Trip the Reactor AND GO TO EOP E-0.

f. Locally close the MSIVs. Refer to EOP E-0, Appendix L.

THIS STEP CONTINUED ON NEXT PAGE

SECTION C: TRAVELING SCREEN PROBLEM (Continued)

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

4. PERFORM Reduced Load Shutdown:

(Continued)

h. Start both MD AFW Pps

h. IF S/G Levels CANNOT be maintained,

THEN Perform the following:

- 1) Start the TD AFW Pp,
- 2) Trip the Reactor,
- 3) GO TO EOP E-0

NOTE: S/G Blowdown Isolation will be initiated by the next step.

i. Trip the MFW Pps

5. CHECK Secondary System Shutdown

a. Perform Attachment 6.1, Hot Condenser Cooldown Instructions

CAUTION: Breaking vacuum with the turbine at greater than 200 rpm could cause serious turbine damage.

b. Perform the following as time permits:

- Verify SCW swapped to an alternate supply per OP F-1:VI
- Verify Intake Cooling Water secured per OP E-4:III
- Direct maintenance to remove RM-15/15R from service
- WHEN turbine speed is less than 200 rpm, THEN locally OPEN, CND-1-506, Condenser Vacuum Breaker.
- Secure gland steam per OP C-3A:III
- Secure the air ejectors per OP C-6:III

c. Implement OP O-19 for appropriate actions with MSIVs closed. (Perform actions as if a Reactor Trip occurred)

Degraded Condenser

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PAGE 15 OF 16

SECTION C: TRAVELING SCREEN PROBLEM (Continued)

ACTION / EXPECTED RESPONSE

RESPONSE NOT OBTAINED

6. Refer to OP L-7 to stabilize the plant or L-5 to continue to shutdown the plant.

Appendix E Actions, Continued

E-0 “Verify Auto Actions” Associated with App E (w/Basis)

Review EOP E-0, Appendix E, emphasizing the following items.

- Action/Expected Response and Response Not Obtained basis
- Procedure transitions
- Other discussion items in the list below

Obj 10, 11

Item	Basis and Discussion
Implement Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status	<ul style="list-style-type: none"> • App E is assigned in the A/ER column of E-0 just after completing the immediate action steps • App E is done concurrently with the body of E-0 so that timely diagnostics can be accomplished <ul style="list-style-type: none"> • Also facilitates more timely SI termination in the event of a very small LOCA or spurious SI (which minimizes Pzr overfill and excessive Pzr PORV lifts) • If App E is not completed by the transition out of E-0, the SFM shall be notified of progress and the SFM shall determine how to proceed <ul style="list-style-type: none"> • In most cases all of App E is expected to be done prior to transition out of E-0, but ESF verifications through step 8 (as a minimum) should be done (not set aside for other tasks) at that point • The expectation is to complete App E, even if some steps remain after transition from E-0 (SFM call if an extreme resource problem arises) • Expectation is to notify the SFM prior to taking major actions in the RNO column (unless previously discussed).

Continued on next page

Degraded Condition or Degraded

A condition of a system, structure, or component in which there has been a loss of quality or functional capability. Examples include:

- The physical or assembled condition of a system, structure, or component does not match its originally installed condition. This condition can include parts that are:
 - burned
 - cracked
 - corroded
 - damaged/broken
 - deformed (e.g., twisted, bulged)
 - missing or improperly installed
- The performance of a system, structure, or component does not meet established acceptance criteria or comply with design requirements.

Elective Maintenance(EM)

The classification used for work performed to restore a plant SSC with identified potential or actual degradation that is not significant and does not affect the SSC's design function or performance. Examples include the following:

- Minor leaks that are controlled and that do not justify immediate action to repair. For example:
 - Body to bonnet leakage that is collected by a drip bag and directed to a drain.
 - A pin hole leak in the air admission valve (SV) to a diesel generator.
- Minor SSC degradation identified by predictive, periodic, or planned preventive maintenance that must be corrected to maintain the long-term equipment reliability but that is not expected to result in failure prior to the SSC's next scheduled maintenance period. For example:
 - Vibration of a charging pump is found in the alert range during a monthly STP.
 - Grease is leaking from the valve operator of a condensate demin bypass valve.
- Other minor SSC deficiencies that do not affect plant operation, nuclear safety, plant reliability, or the ability of operators to properly respond to normal, off-normal, or accident transients or conditions. Examples include: a) damaged or broken local indication gauges that are only informational and are not required by operators to control systems for normal or emergency response; b) indications of internal valve leakage that do not affect system operation or the ability to provide maintenance isolation. For example:
 - A service cooling water pump local discharge pressure gauge has wide fluctuations.
 - One of two turbine building sump pumps will not run.
 - RCP seal injection backpressure control valve leaks by when shut.

Emergency Maintenance

Immediate maintenance to prevent or mitigate the release of radioactive material, hazards to personnel, or extensive equipment damage.

Emergent Work

Work added to the daily schedule after scope freeze. Emergent work does not include OST work. Refer to AD7.ID4 for additional requirements or information.

Maintenance

Activities performed to maintain or restore systems, structures, or components, including activities that implement design changes.

In addition to activities traditionally associated with maintaining and restoring equipment or implementing design changes, maintenance includes all supporting functions, such as planning, scheduling, procurement, testing, etc., required to perform these activities.

Examples of maintenance activities include:

- Activities that do not permanently alter the design, performance requirements, operation or control of a system, structure, or component, such as:
 - Calibration
 - Housekeeping
 - Troubleshooting
 - Refurbishment
 - Identical replacements
 - Installing design changes
 - Postmodification testing
 - Postmaintenance testing
- Temporary modifications and configuration changes that directly relate to and are necessary to support the maintenance, such as:
 - Lifting leads
 - Jumpering terminals
 - Removing barriers
 - Erecting scaffolds
 - Placing ancillary structures
 - Installing temporary power supplies
 - Placing lead shielding on pipes and equipment
 - Using temporary blocks, bypasses, and supports

Minor Maintenance

Work that can be performed safely without detailed work instructions and without overall plant coordination or scheduling.

Modification

Any change to the plant's physical configuration or operation and maintenance practices that is implemented using the modification control process.

Other Maintenance (OM)

The classification used for work that does not directly correct material condition deficiencies on plant SSCs. Examples include:

- Support activities, such as, shielding and insulation activities associated with other work.
- Painting, housekeeping, and general plant upkeep.
- Proactive actions to enhance plant equipment performance (that is, plant betterment activities) that are not directly related to correcting existing deficiencies. This classification includes implementation of design changes that are not required to restore an existing degraded component and proactive replacement of obsolete equipment or replacements based on services life expectations.
- Preoutage work required to support planned or forced outage activities.
- Work required to rebuild components for return-to-stock and reuse.
- Nonplant equipment maintenance. This category includes items such as work on auxiliary equipment used in the power block, for example mobile cranes and M&TE.

OST Work

- Work assigned to the operations support team (OST).

PACIFIC GAS AND ELECTRIC COMPANY
 NUCLEAR POWER GENERATION
 DIABLO CANYON POWER PLANT
 ADMINISTRATIVE PROCEDURE

NUMBER OP1.DC17
 REVISION 11
 PAGE 1 OF 6

TITLE: Control of Equipment Required by the Plant Technical Specifications or Other Designated Programs

10/06/05
 EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED
SPONSORING ORGANIZATION: OPERATIONS
REVIEW LEVEL: "A"

1. SCOPE

- 1.1 This procedure describes the methods for control and tracking of equipment inoperability as required by the Technical Specifications (TSs) or Equipment Control Guidelines (ECGs). In addition, risk significant equipment (designated per AD7.DC6, On-Line Maintenance Risk Management) unavailability shall also be tracked using this procedure.
- 1.2 Normally, Technical Specification Tracking sheets should only be used for the above items. However, other items (such as NPDES Permit required instrumentation) may be tracked if desired by the shift foreman (SFM).

2. DISCUSSION

- 2.1 When equipment required to comply with a TS or ECG LCO, including computer based equipment such as the PPC, EARS, SPDS and main annunciator system, is not able to meet the definition of operability, then tracking equipment status is normally performed using a TS sheet generated in PIMS.
- 2.1.1 TSs are considered either active or info. If the equipment inoperability causes entry into an action statement, then the TS sheet is considered active. Normally (Modes 1-4) both active and Info TS sheets are tracked for reference and status control purposes.
- 2.1.2 During refueling outages (Modes 5 or 6) or at other times as approved by the operations manager, tracking of Info TSs will not be done. Elimination of Info TSs reduces unnecessary work tracked by other methods such as Mode transition checklist and clarifies the important TSs for the plant status.
- 2.1.3 Operations provides a Tech Spec sheet usage guide which describes how to use the PIMS module.
- 2.2 All items which are risk significant as defined by AD7.DC6, "On-Line Maintenance Risk Management," are to be tracked per this procedure in Modes 1-4 if the equipment is risk significant for the current Mode.
- 2.3 Other items may also be tracked using a TS tracking sheet as desired by operations management.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.3.4	
	Importance		3.1

Proposed Question:

A Site Area Emergency has been declared due to a LOCA Outside Containment with limited makeup to the RWST available.

An operator volunteers to make an emergency entry into the penetration area to attempt to isolate the leak. This action would result in a significant reduction in offsite dose. The individual has all the required approvals and the following exposure history:

- Age 25 yrs.
- Total Lifetime exposure 3800 mrem TEDE
- Current Year exposure 800 mrem TEDE

What is the MAXIMUM exposure the operator may receive while performing this action?

- A. 4200 mrem TEDE
- B. 5000 mrem TEDE
- C. 24,200 mrem TEDE
- D. 25,000 mrem TEDE

Proposed Answer:

- D. 25,000 mrem TEDE

Explanation:

A incorrect, this includes a reduction of federal limit (5 rem TEDE) less current exposure

B incorrect, this is the federal limit and emergency exposure limit for radiological assessment sampling

C incorrect, this takes into account the current TEDE which does not apply

D correct, to save a life or for dose saving to population, 25 rem is the guideline.

Technical Reference(s): EP RB-2, DCPN Emergency Exposure Guidelines

Proposed references to be provided to applicants during examination: none

Learning Objective: 7954 - State the emergency dose limits

Question Source: Bank # INPO 20049

Question History: Last NRC Exam Braidwood 10/2001

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 43.4

Comments: K/A: G2.3.4 - Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

DIABLO CANYON POWER PLANT
EP RB-2
ATTACHMENT 9.6

1 AND 2

TITLE: DCPD Emergency Exposure Guidelines

The following table contains guidelines for use in authorizing emergency exposures when lower doses are not practicable:

	RADIOLOGICAL ASSESSMENT SAMPLING	PROPERTY SAVING	DOSE SAVING TO POPULATION*	LIFESAVING TO INDIVIDUAL*
Emergency Actions----> Part of Body Irradiated	Sampling Under Emergency Conditions	Mitigating Damage to Valuable Property	Corrective Actions, stop/reduce a release	Lifesaving Actions, 1st Aid, Search and rescue
Whole Body	5 rem TEDE	10 rem TEDE	25 rem TEDE	25 rem TEDE
Skin & any Extremity	50 rem SDE	100 rem SDE	250 rem SDE	250 rem SDE
Lens of the Eye	15 rem LDE	30 rem LDE	75 rem LDE	75 rem LDE
Any Organ or Tissues	50 rem (CDE+DDE)	100 rem (CDE+DDE)	250 rem (CDE+DDE)	250 rem (CDE+DDE)

- NOTES:**
- Radiological Assessment Sampling, includes collection of atmospheric, liquid, and environmental radiological activity samples as well as chemistry samples involving high activity or high radiation. Emergency exposure limits may be authorized for selected individuals, for emergency assessment functions, in addition to annual occupational dose to date.
 - Property Saving, for example, might be dispatching the Fire Brigade to extinguish a fire in a Very High Radiation Area to protect plant equipment though no immediate threat exists to compromising Plant Safety.
 - Dose Saving to Population, includes activities that justify a potential overexposure to a few workers in order to save even a small average dose in a large population. (May also include Traffic Control for Evacuees or other Security Plan Functions.)
 - Lifesaving to Individual, includes the activity of search and rescue in very high dose rates or high airborne activity.

* **Extreme situations** may occur in which a dose in excess of 25 rem TEDE would be unavoidable for either Dose Saving to (Large) Population or Lifesaving to (An) Individual.

An authorization of emergency exposure with **NO LIMITS** may be made under those conditions, but only to volunteers who are fully aware of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and the numerical estimates of the risk of delayed effects.

PACIFIC GAS AND ELECTRIC COMPANY
 NUCLEAR POWER GENERATION
 DIABLO CANYON POWER PLANT
 OPERATING PROCEDURE

NUMBER OP G-1:II
 REVISION 33
 PAGE 1 OF 10
 UNITS

TITLE: Liquid Radwaste System - Discharge of Liquid Radwaste

1 AND **2**

11/15/06

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

- 1.1 This procedure is intended to provide a method of safely discharging water from the Liquid Radwaste System. The primary goal is to prevent an inadvertent or unmonitored release. This procedure should be performed in conjunction with CAP A-5 and CAP A-11.
- 1.2 This procedure is important to safety and to environmental quality.

2. DISCUSSION

- 2.1 Since this procedure is important to safety and to environmental quality, all documentation for each discharge must be completed entirely and accurately. Under no circumstances shall any discharge be initiated without the appropriate attachments being completed and reviewed by the shift foreman (SFM).
- 2.2 All discharges must be authorized and approved by the chemistry foreman and the shift foreman. Authorization and approval is documented by signatures on the Authorization for Discharge, Attachment 9.1 and Attachment 9.2 through 9.9 as applicable.

3. RESPONSIBILITIES

- 3.1 It is the chemistry foreman's responsibility to generate an Authorization for Discharge Form (69-9325) based on analysis of the tank to be discharged.
- 3.2 It is the SFM's responsibility to issue, review and approve Attachment 9.1 and Attachment 9.2 through 9.9 as applicable prior to issuing the Authorization for Discharge. It is also the SFM's responsibility to ensure all documentation is properly filled out and that all requirements are met prior to starting a discharge.
- 3.3 It is the operators responsibility to perform Attachment 9.1 and Attachment 9.2 through 9.9 as applicable in its entirety prior to returning it to the SFM for approval. It is the operators responsibility to date strike and document the discharge number on FR-20 at the Auxiliary Control Board. The operator shall log the discharge (batch) number and tank being discharged in the aux senior log. It is also the operators responsibility to monitor the discharge and notify the SFM of any abnormalities which may occur during the discharge. The operator shall return all appropriate paper work to the SFM for review and if satisfactory it will be routed to the chemistry foreman.

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER OP AP-9
REVISION 22
PAGE 17 OF 39
UNITS 1 AND 2

TITLE: Loss of Instrument Air

APPENDIX A (Continued)

INSTRUMENT AIR HEADER TABLES
U-1 AUXILIARY BUILDING/CONTAINMENT

ISOLATION VALVE: AIR-I-1-221 (Inst. Air Receiver inlet)
Location: At Instrument Air Receiver GW areas.

PRINCIPAL AREAS SERVED:

1. U-1 Containment
2. U-1 Aux. Bldg., 100' elev., GE and in U-1 100' GW area.

PRECAUTIONS: 1. *Normally isolated by:

- MS-1-902 (sealed valve), 100' GE area, 140° and
- N₂-1-34 (100' pen GE area, 140°)

MAJOR COMPONENTS SERVED - AUX BLDG

HCV-637 and 638, RHR Hx Outlets
HCV-670, RHR Hx Bypass
8152, Letdown Hx Inlet
Containment LRW Iso's

FAILED CONDITION

Fail Open
Fail Open
Fail Closed (N₂ Backup)
Fail Closed

MAJOR COMPONENTS SERVED - CONTAINMENT

PCV-474, PORV
PCV-455C and 456, PORV's
PCV-455A and 455B, Pzr Sprays
LCV-459 and 460, Regen. Hx Letdown Iso's
8149A, B, and C, Letdown Orifice Iso's
8146 and 8147, Charging to Cold Legs
8145, Charging to Aux. Spray
8141A, B, C, and D, RCP Seal Leakoffs
8142, RCP #1 Seal Bypass
RCS Hot Leg Sample Iso's
Steam Gen Blowdown and Blowdown Samples
Manipulator Crane
FCV-584, Inst Air to Containment
8166 and 8167 Excess Letdown Hx inlet
HCV-123 Excess Letdown Hx Outlet
8143 Excess Letdown to VCT or Seal Water Hx

FAILED CONDITION

Fail Closed
Fail Closed (N₂ B/U Bottles)
Fail Closed
Fail Closed (N₂ B/U from Stm.Gen. N₂)*
Fail Closed (N₂ B/U from Stm.Gen. N₂)*
Fail Open (N₂ B/U Bottles)
Fail Closed (N₂ B/U Bottles)
Fail Open
Fail Closed
Fail Closed (Backup Air Bottles)
Fail Closed
Fails gripper engaged
Fail Closed
Fail Closed
Fail Closed
Fail to VCT position

DCPD Bank Question:

A reactor trip and SI occurred from 100% power due to a Main Steamline break upstream of the 1-1 MSIV. NO RED or MAGENTA paths exist on the CSFSTs. Select the proper procedural progression from the choices below:

- E-0 "Reactor Trip or Safety Injection"
 - E-1 "Loss of Reactor or Secondary Coolant"
 - E-1.1 "SI Termination"
 - E-1.2 "Post-LOCA Cooldown and Depressurization"
 - E-2 "Faulted Steam Generator Isolation"
- A. E-0, to E-2, to E-1, to E-1.2
- B. E-0, to E-2, to E-1, to E-1.1
- C. E-0, to E-1, to E-2, to E-1.1
- D. E-0, to E-2, to E-1.1

Answer: D

Question Type: Multiple Choice

Topic: EOP Steam Line Break procedural sequence

System ID: 32956

User ID: **B-0076**

Status: Active

Always select on test: No

Authorized for practice: No

*** UNCONTROLLED PROCEDURE - DO NOT USE TO PERFORM WORK or ISSUE FOR USE ***

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

NUMBER EOP E-0
REVISION 30A
PAGE 9 OF 33

TITLE: Reactor Trip or Safety Injection

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

11. CHECK S/Gs - NOT FAULTED

- o NO S/G Pressure decreasing in an uncontrolled manner
- o NO S/G Completely depressurized

GO TO EOP E-2, FAULTED
STEAM GENERATOR
ISOLATION

12. Check S/Gs - NOT RUPTURED

- a. Secondary Radiation Monitors - NORMAL
 - o PK11-06, SJAE HI RAD - OFF
 - o PK11-17, S/G BLOWDOWN HI RAD - OFF
 - o PK11-18, MAIN STM LINE HI RAD - OFF
- b. Secondary Radiation Recorders (VB2) - NO UPWARD TREND OR SPIKE
 - o RM-15/15R, SJAE Monitors
 - o RM-19, Blowdown Monitor
 - o RM-71, 72, 73 or 74, Main Steam Line Monitors
- c. Steam Generator Level - NO S/G LEVEL INCREASING IN AN UNEXPECTED MANNER (WR/NR)

a. IF A valid alarm exists

THEN GO TO EOP E-3,
STEAM GENERATOR
TUBE RUPTURE

b. IF A valid upward trend or spike exists,

THEN GO TO EOP E-3,
STEAM GENERATOR
TUBE RUPTURE

c. GO TO EOP E-3, STEAM GENERATOR
TUBE RUPTURE

**PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT**

**NUMBER EOP E-2
REVISION 15
PAGE 5 OF 9**

TITLE: Faulted Steam Generator Isolation

UNIT 1

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

7. Check Secondary System Radiation for S/G Tube Rupture:

- a. Steam Line Radiation - NORMAL
 - o PPC trend indicates NO upward trend or spike prior to the trip on RM-71, 72, 73, or 74 - Main Steam Line Monitors
 - o NO valid alarm on PK11-18, MAIN STEAM LINE HI RAD

- a. GO TO EOP E-3, STEAM GENERATOR TUBE RUPTURE.

- b. S/JAE Radiation - NORMAL
 - o PPC trend indicates NO upward trend or spike on RM-15 or 15R - S/JAE Monitors
 - o NO valid alarm on PK11-06, S/JAE HI RAD

- b. GO TO EOP E-3, STEAM GENERATOR TUBE RUPTURE.

- c. S/G Blowdown Radiation - NORMAL
 - o PPC trend indicates no upward trend or spike prior to the trip on RM-19 - Blowdown Monitor
 - o NO valid alarm on PK11-17, S/G BLOWDOWN HI RAD

- c. GO TO EOP E-3, STEAM GENERATOR TUBE RUPTURE.

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- d. Contact Chemistry Department to conduct periodic samples of all S/Gs - NORMAL ACTIVITY

- d. IF HIGH S/G ACTIVITY,
THEN GO TO EOP E-3,
STEAM GENERATOR TUBE RUPTURE.
