RO QUESTION 01

Question Worksheet

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

<u>Proposed Question</u>:

PLANT CONDITIONS:

- Unit 1 is at full power.
- Power range channel N43 has been removed from service and all appropriate bistables tripped in the protection racks.
- Control and instrument power fuses have not been removed.

A loss of PY-12 occurs and the reactor trips.

Which of the following describes why the reactor tripped and the bistable status indications?

A. 2/4 OTΔT, however, only one OTΔT bistable will be lit.

B. 2/4 OPΔT, however, only one OPΔT bistable will be lit.

C. 2/4 OTΔT, both OTΔT bistables will be lit.

D. 2/4 OPΔT, both OPΔT bistables will be lit.

Proposed Answer: A. 2/4 OTΔT, however, only one OTΔT bistable will be lit.

Explanation:

A correct, Per OP-5 only OTΔT bistables (TC-431C OtdeltaT and TC-431D OTdeltaT Runback) are tripped for the failed NI due to the the ΔI input. OP does not get tripped, (ΔI is zeroed out).
PY-12 provides bistable status light indication for channel 3 and power to channel 2 bistables. When it fails, channel 3 lights go out (loss of OTΔT bistable status for NI43) and trips channel 2 bistables. The coincidence for a reactor trip is satisfied but only one (channel 2) OTΔT bistable will be lit. Note, channel 2 bistable status lights are powered from PY-11A.
B incorrect, overpower bistables are not tripped.
C incorrect, the lights for channel 3 are out.
D incorrect, overpower not tripped.

Technical Reference(s): AP-5 attachment 4.1. OP AP-4, Loss of Vital or Nonvital Instrument AC, section 2, Symptoms.

NI-43 OTdeltaT (TC-431C OTdeltaT Trip)
(see Page 15) (TC-431D OTdeltaT Runback)

Proposed references to be provided to applicants during examination: None

Learning Objective: 4274 - Explain the consequences of loss of vital instrument bus.

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 41.7
55.43 _____

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

TITLE: Loss of Vital or Nonvital Instrument AC

11/09/04
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure covers the general steps to be taken in the event of a loss of power to a vital or nonvital instrument AC panel. A comprehensive list of particular symptoms precedes each section. However, particular annunciator windows and symptoms are listed below to enable quicker diagnosis of which panel was lost, and the appropriate section of the procedure to refer to. This list is not all-inclusive:

MAJOR SYMPTOMS	GO TO Section
Postage Stamp Bistable Row Illuminates (VB1) Postage Stamp Row Deenergizes (VB1) Multiple Hagan Controllers Fail to MANUAL or AUTO-HOLD Multiple Seemingly Unrelated Alarms Including: PK16-22, PK17-22 or PK18-22 PK19-19	Section A, Loss of Vital Instrument Panel
Loss of Indication on Both MFP Startup Stations MSR Valve Position Indicators Dead (VB3) PPC and Westronics Chart Recorders Dead (VB3 & VB4) Multiple Seemingly Unrelated Alarms Including: PK01-08 PK10-07 PK15-20 PK15-22	Section B, Loss of PY-15 (25)
CCW Surge Tank LIs Fail Low Multiple Seemingly Unrelated Alarms Including: PK01-07 PK11-04 PK12-02 PK15-18	Section C, Loss of PY-16 (26)
Generator Indications Fail Low (H_2 Pressure, Density, Seal Oil ΔP) Valve Position Lights on DEH Panel Lost Condensate Booster Pump Set Autostart Multiple Seemingly Unrelated Alarms Including: PK12-20 PK14-16 PK14-18 PK14-19	Section D, Loss of PY-17 (27)

RO QUESTION *02*

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	APE 008 AA2.20	
	Importance	3.4	3.6

Proposed Question:

An automatic reactor trip and safety injection has occurred on Unit 1.

The following plant conditions currently exist:

- All RCPs are running
- Pressurizer level is 48% and INCREASING
- RCS pressure is 1700 psig and DECREASING

Which of the following leak locations is consistent with the current plant conditions?

A. A charging header flange.

B. A pressurizer PORV.

C. CRDM Canopy Seal.

D. A weld failure of an RCS flow sensing line.

Proposed Answer:

B. A partially open pressurizer PORV.

Explanation:

A, C and D incorrect, pressurizer level would decrease.
B correct, Increasing pressurizer level and decreasing RCS pressure are the symptoms of a vapor space leak in the pressurizer.

Technical Reference(s): LMCDFRC – Mitigating Core Damage, Core Cooling

Proposed references to be provided to applicants during examination: none

Learning Objective: 3772 Identify the parameter that helps to discriminate between a vapor space and non-vapor space LOCA

Question Source: Bank # P-27764, INPO 19184

Question History: Last NRC Exam Braidwood 1 - 2000

Question Cognitive Level:

Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5

55.43

Comments: K/A: APE 008 AA2.20 - Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: The effect of an open PORV on (or?) code safety, based on observation of plant parameters

Steam Space versus Liquid Space LOCA

Differences
Obj 18,20,22

Pressurizer level as an indication of inventory can be misleading if:

- RCS subcooling does not exist.
- A steam vent path is established from the Pzr vapor space.

A vapor space LOCA would depressurize the RCS and quickly transfer the bubble to the reactor vessel.

- Steam generated in the reactor vessel may pass through the Pzr surge line and prevent the water inventory of the Pzr from draining into the RCS loops.
- This holdup of water can result in a stable or even increasing indicated Pzr level while RCS water inventory is actually decreasing.
- Pzr level should be relied on only with hot leg or core exit subcooling present.

The parameter that helps to discriminate between a vapor space and non-vapor space LOCA is the Pzr level indication.

- A vapor space LOCA provides the steam vent path from the Pzr which will cause a depressurization in the Pzr causing an indicated level increase while water flashes to steam and out the vent path.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	EPE 011 EK1.01	
	Importance	4.1	4.4

Proposed Question:

PLANT CONDITIONS:
- An RCS cold leg break has occurred on Unit 1.
- RCS pressure is 800 psig and decreasing slowly.
- RCS level is beginning to drop below the top of the steam generator U-tubes.
- Steam generator pressures are approximately 1000 psig and decreasing slowly.

Which of the following describes the heat removal mechanism currently occurring?

A. Break flow only.

B. Break flow and reflux cooling.

C. Break flow and natural circulation.

D. Break flow and radiative heat transfer.

Proposed Answer:

A. Break flow only.

Explanation:

A correct, true for large break LOCAs.
B incorrect, this is for a small break, heat removal by the break is insufficient, RCS pressure will remain above steam generator pressure and as the RCS is drained heat removal will be reflux cooling.
C incorrect, this is for a small break, heat removal by the break is insufficient, RCS pressure will remain above steam generator pressure and heat removal will be thru two phase natural circulation or as the RCS is drained, reflux cooling.
D incorrect, the core is still covered.

Technical Reference(s): LMCDFRC - MITIGATING CORE DAMAGE -CORE COOLING page 16
Proposed references to be provided to applicants during examination: none

Learning Objective:
E598 - Describe the following concepts or conditions as they apply to a LOCA: a. Mechanisms for removing decay heat from the core during a LOCA.

Question Source:
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 41.14
55.43 _____

Comments: K/A: EPE 011 EK1.01 - Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA : Natural circulation and cooling, including reflux boiling

Error! Style not defined., Continued

Example
FIGURE -04
FIGURE -05
FIGURE -06

An example follows.

The conditions:

- a 2″ equivalent diameter hole in cold leg
- minimum safeguards SI
- loss of offsite power assumed at reactor trip
- SG safety valves only means of venting steam on secondary side
- minimum AFW available one minute after reactor trip

Analysis:

- rapid depressurization to 1200 psig at 5 minutes
- from 5 to 30 minutes:
  - ♦ system repressurizes due to inability to vent steam.
  - ♦ SG primary side and crossover leg drain.
- at 30 minutes crossover leg level reaches break and steam passes.
  - ♦ rapid decrease in steam flow.
- as steam escapes through break, differential pressure adjusts in reactor vessel such that core is recovered at 32 minutes.
- SI flow becomes greater than break flow.
- from 32 to 50 minutes:
  - ♦ slow repressurization
  - ♦ secondary side pressure drops below S/G safety valve setpoint and continues to drop
- after 65 minutes:
  - ♦ *break and subcooled SI remove all decay heat*
  - ♦ *secondary pressure above primary*
  - ♦ system now in stable mode

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	______
Group #	1	______
K/A #	APE 015 AA1.10	
Importance	2.7	2.6

<u>Proposed Question</u>:

A locked rotor occurs on a running Unit 1 RCP.

Which of the following describes what you would observe from the time the locked rotor occurs until the RCP trips and the RED light goes out and amps fall to zero?

A. Amp indication pegs high and after a time delay the breaker trips on overcurrent - the green and blue lights are lit.

B. Almost immediately, before amp indication can peg high, the breaker trips on overcurrent - the green and blue lights are lit.

C. Amp indication pegs high and after a time delay the breaker trips on overcurrent - the green light is lit, the blue light remains out.

D. Almost immediately, before amp indication can peg high, the breaker trips on overcurrent - the green light is lit, the blue light remains out.

Proposed Answer:

A. Amp indication pegs high and after a time delay the breaker trips on overcurrent - the green and blue lights are lit.

Explanation:

A correct, For a locked rotor, amps will peg high (several times higher than normal running amps). The breaker will trip on overcurrent when the inverse time constant is picked up. The blue light indicating overcurrent on the primary or backup breaker will be lit.
B incorrect, amps will peg high and the breaker does not trip immediately.
C incorrect, blue light will light.
D incorrect, blue light will light.

Technical Reference(s):
STG A6, Reactor Coolant Pump

Proposed references to be provided to applicants during examination: none

Learning Objective:
32652 - Describe the following terms/ characterizes as relates to AC induction motor operation:

- Slip
- Locked rotor
- Sheared shaft

6056 - Analyze the control logic for the RCPs.

Question Source:
New X

Question History: Last NRC Exam ____________
Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: K/A: APE 015 AA1.10 - Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCP ammeter and trip alarm

Need to know

Rotor Slip, continued

- When a motor is running unloaded, there is very little slip (typically less than one percent).
- As the motor is loaded, the rotor slows down slightly. Slip increases, and the rate at which the rotor windings are cut by the rotating magnetic field also increases. This action results in an increase in rotor voltage and current. The increase in rotor current causes an increase in the force exerted on the rotor, and motor torque increases.
- A very small drop in rotor speed produces a considerable increase in torque.
- At full load, slip is about 6 percent of the rotating magnetic field speed for a typical motor (rotor speed is approximately 6 percent less than field speed).

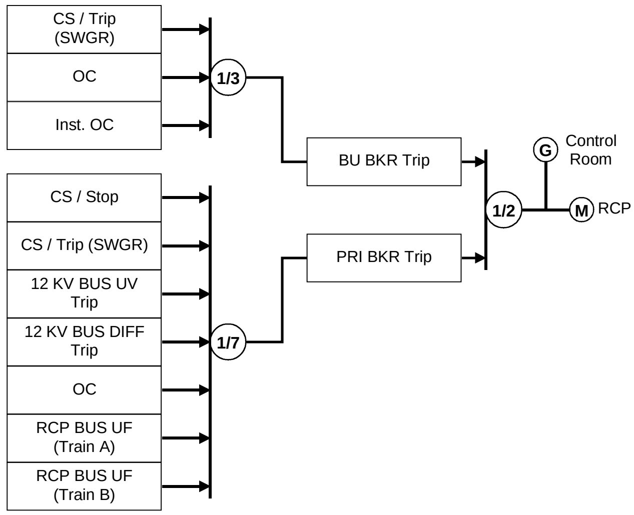
Locked Rotor

- Stopping of the motor rotor due to an excessive load such as the binding of a pump shaft.
- Indications of a locked rotor

➢ Instantaneous increase in pump current attempting to supply demanded torque
➢ Instantaneous decrease in pump discharge pressure
➢ Instantaneous decrease in system flow rate
➢ Instantaneous increase in motor winding temperatures
➢ Possible breaker trip

Trip logic
Obj 15, 16

The tripping logic for an RCP is shown below.



RCP-12

Notes:

- Control room indication indicates that either the primary or backup breaker is tripped.
- Undervoltage trip at <8050 V (70% of nominal)
- Underfrequency trip at < 54 Hz
- Trip testing is done through the SSPS system[TRP1]

Indication
Obj 7, 14

The following indications are available for each RCP motor.

VB-1 START/STOP switch		
Indicating light	**Meaning**	**Normal status**
Red	Pump motor primary <u>and</u> backup breaker closed	ON
Green	Pump motor primary <u>or</u> backup breaker open	OFF
White	• Potential on 12 kV bus • Primary and backup breaker cell interlocks sense breaker in connected position • Primary and backup breaker have 125VDC power[TRP2]	ON
Blue	Pump primary <u>or</u> backup breaker tripped on overcurrent	OFF

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	APE 022 G2.1.25	
	Importance Rating	2.8	3.1

<u>Proposed Question</u>:

PLANT CONDITIONS:
- Unit 1 has tripped from full power.
- The crew has entered E-0.1, Reactor Trip Response, and addressing 2 rods failing to insert using OP AP-6, Emergency Boration.
- Attempts to establish adequate flow through the normal boration flowpath have failed.
- The crew has established the minimum required flow through the next preferred path

At the minimum flow rate, how long must the boration be in progress?

A. 10 minutes

B. 20 minutes

C. 30 minutes

D. 60 minutes

Proposed Answer:

D. 60 minutes

Explanation:

A incorrect, If the RWST at minimum flow is used (90 gpm) and candidate uses 1 rod, (900 gallons).
B incorrect, this is 1800 gallons at 90 gpm (RWST)
C incorrect, this is 30 gpm for 900 gallons (1 rod)
D correct, Per AP-6, if normal boration is unsuccessful, the next preferred path is thru CVCS-8104 with a minimum flow of 30 gpm. Per attachment A, the requirement is to borate 900 gallons, PER ROD. At 30 gpm, to borate 1800 gallons would take 60 minutes as a minimum.

Technical Reference(s): OP AP-6 att. A

Proposed references to be provided to applicants during examination: OP AP-6

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

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 41.10
55.43 43.5

Comments: K/A: APE 022 G2.1.25 - Loss of Reactor Coolant Makeup - Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.

TITLE: Emergency Boration

08/26/04
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

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

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

NOTE: Emergency boration flowrate is not specifically defined. TS 3.1.1 Bases states that the operator should borate with the best source available for plant conditions.

2. SYMPTOMS

Any one of the following conditions requires emergency boration.

2.1 Control rods inserted below the low-low insertion limit when critical. Commence boration within 15 minutes to restore rods to above the Lo-Lo Insertion Limit.

ROD LO LO INSERTION LIMIT (PK03-14)

2.2 Failure of any 2 control rods to fully insert following a reactor trip as indicated by rod position indication and rod bottom lights. Commence boration when directed by the Emergency Procedures.

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

2.4 Uncontrolled or unexplained reactivity increase as indicated by:

2.4.1 Unexplained control rod insertion.

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

2.4.3 Unexpected increasing count rate when shutdown.

2.5 When boration is required and normal boration through the VCT makeup system is not possible. Borate as required per SFM direction.

2.6 Shutdown margin less than acceptable minimum limits per ITS 3.1.1 and 3.9.1. Commence boration within 15 minutes to restore rods to restore shutdown margin.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE 1: 900 gallons of 4% boric acid provides - 100 ppm INCREASE TO THE RCS, BOL. Calculated values may be used in place of this thumbrule.	
NOTE 2: If Letdown is NOT in service, then it will be necessary to cool down 50°F per hour while injecting Boric Acid at 30 gpm in order to maintain a constant pressurizer level.	
1. **INITIATE Emergency Boration**:	
a. Verify charging is in service	a. GO TO OP AP-17, LOSS OF CHARGING.
b. Place VCT make up control in BORATE position	
c. Set boron flow controller HC-110 pot setting to 9.0 turns	c. Increase demand manually to 100% on HC-110.
d. Enter the desired gallons of boric acid using the BATCH function and the data entry keys. Refer to Appendix A for boration requirements.	
e. Press RESET and START keys to enable the integrator.	
f. Place M/U controller 1/MU in START position - Adjust HC-110 pot setting to obtain approximately 30 GPM of boric acid flow	f. Perform the following: 1) Verify BA Transfer Pp - HIGH SPEED 2) Close HCV-104 (BATP 1-1/2-2) OR HCV-105 (BATP 1-2/2-1) 3) IF VCT pressure GREATER THAN 30 PSIG, THEN Vent the VCT by opening CVCS-8101 until LESS THAN 30 PSIG 4) IF Unable to obtain adequate Boric acid flow, THEN GO TO Step 2.
g. GO TO Step 3	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

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

2. **INITIATE Alternate Boration Method**

   a. OPEN CVCS-8104 and verify approximately 30 GPM or greater Emergency Boration Flow

   Response Not Obtained:

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

      1) Swap Charging Pp suction to the RWST.

         a. OPEN 8805A AND 8805B.

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

         c. VERIFY GREATER THAN 90 GPM charging flow.

         OR

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

3. **CHECK Sufficient Boric Acid Available:**

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

   Response Not Obtained:

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

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

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

APPENDIX A

BORATION REQUIREMENTS

Borate prescribed amount according to entry symptom:

	Symptom		Boration Requirement
1.	Rods below RIL while critical.	1.	Within 15 minutes, commence and maintain boration until rods above RIL.
2.	Two or more stuck rods following reactor trip.	2.	Borate 100 ppm (900 gallons, or calculated addition) per stuck rod.
3.	Uncontrolled cooldown following Rx Trip without ESF Actuation.	3.	Borate until adequate shutdown margin is attained.
4.	Uncontrolled or unexplained reactivity increase.	4.	Borate until control regained. Refer to EOP FR-S.1 Appendix D to isolate dilution flowpaths, if required.
5.	When normal boration methods unavailable.	5.	Borate as required to maintain proper boron concentration.
6.	SDM less than acceptable.	6.	Within 15 minutes, commence and maintain boration until adequate SDM is attained.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1	_____
	K/A #	APE 026 AA2.04	
	Importance Rating	2.5	2.9*

Proposed Question:

The Unit 1 core is being offloaded to the Spent Fuel Pool (SFP). Spent Fuel Pool temperature is 90°F.

A running CCW pump trips and flow to the SFP heat exchanger decreases to 1800 gpm. SFP temperature begins to rise at a rate of 4.0°F /hour.

Without operator action, how long before PK11-04, Spent Fuel Pool Lvl/Temp alarms due to the increasing temperature?

A. 0 to 1 hour.

B. 7 to 8 hours.

C. 8 to 9 hours.

D. 12 to 13 hours.

Proposed Answer:

A. 0 to 1 hour.

Explanation:

A correct, rate of increase is greater than 2.0 degrees. The alarm will occur as soon as the program executes and determines the change is greater than setpoint.
B incorrect, this would be an increase to 120F, a limit to prevent damage to SFP demins.
C incorrect, this corresponds to the alarm setpoint of 125 degrees.
D incorrect, this corresponds to 140 degrees, a value that requires action in AP-22, Spent Fuel Pool Low Level/High Temp/High Rad.

Technical Reference(s): PK11-04

Proposed references to be provided to applicants during examination: none

Learning Objective: 5262 - State the Spent Fuel Pool Cooling system parameters that produce alarms.

Question Source:

New X

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 41.5
55.43

Comments:
K/A: APE: 026 AA2.04 - Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW

TITLE: **SPENT FUEL POOL LVL/TEMP**

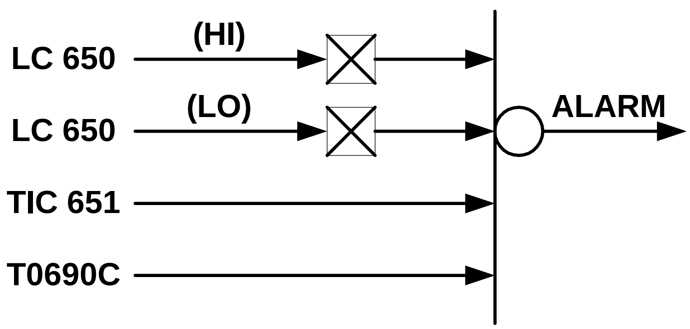
1

07/09/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
LC 650	1063	Spent Fuel Pool Lvl Hi	GT 139'0" Elev
LC 650	1064	Spent Fuel Pool Lvl Lo	LT 137'4" Elev
TIC 651	838	Spent Fuel Pool Temp Hi	GT 125°F
T0690C	0903	SFP Temperature Rate of Change High	GT 2°F / hour and GT 80°F

3. PROBABLE CAUSE

3.1 Spent fuel pool level high due to:

3.1.1 Makeup or leakage into spent fuel pool.

3.1.2 Loading of fuel elements or cask into pool.

3.1.3 Tube leak in spent fuel pool heat exchanger.

3.1.4 High pressure in containment with transfer tube open and canal flooded.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	EPE 029 EA1.01	
	Importance Rating	3.4*	3.1

Proposed Question:

Which of the following is the expected action to be performed to accomplish RCS boration per EOP FR-S.1, "Response to Nuclear Power Generation/ATWS"?

A. Align a CCP to RWST.

B. Initiate normal boration.

C. Actuate Safety Injection.

D. Initiate emergency boration per OP AP-6.

Proposed Answer:

A. Align a CCP to RWST.

Explanation:
A correct, per step 4 of FR-S.1,
a. Perform the following:
   1) OPEN 8805 A OR B (RWST suction)
   2) CLOSE LCV-112 B OR C (VCT suction)
   3) Verify AT LEAST 90 GPM charging flow
B incorrect, this is not an option in S.1
C incorrect, this is the option of last resort, because SI trips the MFP, it is not desireable.
D incorrect, this is the option if unable to align to the RWST.

Technical Reference(s): FR-S.1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 9948 - Explain the operator actions on ATWS

Question Source:
Modified Bank # DCPP P-47631

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 41.10
55.43 _____

Comments:

K/A: EPE 029 EA1.01 - Ability to operate and monitor the following as they apply to a ATWS: Charging pumps

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
4. **INITIATE Emergency Boration of RCS:**	

****

CAUTION: An SI will trip the Feed Pps and should be avoided if power is high enough to require Feed Pps to supply the heat sink.

****

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
a. Perform the following: EMERGENCY BORATION 1) OPEN 8805 A OR B 2) CLOSE LCV-112 B OR C 3) Verify AT LEAST 90 GPM charging flow	a. IMPLEMENT OP AP-6, OR Initiate SI -
b. Check PZR Pressure - LESS THAN 2335 PSIG	b. Perform the following: 1) Verify PZR PORVs and Block Valves Open IF NOT, THEN Open PZR PORVs and Block Valves as necessary until PZR Pressure LESS THAN 2135 PSIG. -
5. **VERIFY Containment Vent Isol:**	
a. CVI portion of Monitor Light Box B: o Red Activated Light- ON o White Status Lights- OFF	a. Manually Close All CVI alves. -

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	EPE 038 EA2.06	
	Importance Rating	3.8	4.4

Proposed Question:

UNIT 1 PLANT CONDITIONS:

- The crew is performing recovery actions for a steam generator tube rupture using EOP E-3.1, Post-SGTR Cooldown Using Backfill.
- RCS temperature is 500ºF
- SI is blocked
- Chemistry reports current boron concentration is 500 ppm.
- Core burnup is 19,000 MWD/MTU
- All rods are inserted

At step 4, the crew is instructed to “Verify Adequate Shutdown Margin”.

Without taking credit for Xenon or Samarium, which of the following describes whether adequate SDM has been established?

A. Yes, adequate SDM currently exists. Current boron concentration is approximately 23 ppm more than required.

B. No, SDM is inadequate. Boron concentration must be raised 77 ppm.

C. No, SDM is inadequate. Boron concentration must be raised 257 ppm.

D. No, SDM is inadequate. Boron concentration must be raised 357 ppm.

Proposed Answer:

D. No, SDM is inadequate. Boron concentration must be raised 357 ppm.

Explanation:

A incorrect, required boron concentration is 857 ppm. It would be adequate if the value for 500F from Table R19-1T-2 was used.
B incorrect, required boron concentration is 857 ppm. It would be adequate if the value for 500ºF from Table R19-1T-2 was used and corrected by adding 100 ppm as required by attachment 9.2 of STP R-19

C incorrect, this is the answer if the 100 ppm correction is not applied.
D correct, the value from R19-1T-2 is 757 ppm. Per STP R-19, attachment 9.2, 100 ppm for conservatism is added, making required boron concentration 857 ppm. Therefore, to meet the required SDM, boron must be raised 357 ppm.

Technical Reference(s): R19-1T-2
Per STP R-19, with attachments

Proposed references to be provided to applicants during examination:

- R19-1T-2
- STP R-19, attachment 9.2

Learning Objective: 10382 - PERFORM a shutdown margin (SDM) calculation using STP R-19, Data Sheet 2, Volume 9 Curves, and appropriate data

Question Source:
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 41.5
55.43

Comments:

K/A: EPE 038 EA2.06 - Ability to determine or interpret the following as they apply to a SGTR: Shutdown margins and required boron concentrations

DIABLO CANYON POWER PLANT OPERATION DATA
TABLE R19-1T-2
Cycle 13

MINIMUM REQUIRED BORON CONCENTRATION (ppm) FOR SHUTDOWN MARGIN AS A FUNCTION OF TEMPERATURE AND BURNUP

BURNUP (MWD/MTU)	CORE AVERAGE TEMPERATURE (°F)												
	68	80	100	120	140	160	180	200	220	240	260	280	300
0	1352	1347	1341	1337	1336	1336	1337	1339	1341	1344	1347	1349	1350
500	1382	1377	1371	1367	1365	1365	1366	1368	1371	1374	1376	1378	1380
1000	1408	1403	1397	1393	1391	1391	1392	1394	1397	1399	1402	1404	1406
1750	1441	1435	1429	1425	1423	1423	1424	1426	1429	1431	1434	1437	1438
2500	1465	1460	1454	1450	1448	1447	1448	1450	1453	1456	1458	1461	1462
3250	1483	1478	1471	1467	1465	1465	1465	1467	1470	1473	1475	1477	1479
4000	1494	1489	1482	1478	1475	1475	1476	1477	1480	1482	1485	1487	1488
4750	1498	1493	1486	1482	1479	1479	1479	1481	1483	1485	1488	1489	1490
5500	1496	1491	1484	1479	1477	1476	1477	1478	1480	1482	1484	1486	1486
6250	1488	1483	1476	1471	1469	1468	1468	1469	1471	1473	1475	1476	1476
7000	1474	1469	1462	1458	1455	1454	1454	1455	1457	1458	1459	1460	1460
7750	1456	1451	1444	1439	1436	1435	1435	1436	1437	1438	1439	1439	1439
8500	1432	1427	1420	1416	1413	1411	1411	1411	1412	1413	1414	1413	1412
9250	1405	1399	1393	1388	1385	1383	1383	1383	1383	1383	1384	1383	1381
10000	1373	1367	1361	1356	1353	1351	1350	1350	1350	1350	1349	1348	1346
10750	1337	1332	1325	1320	1317	1315	1313	1313	1312	1312	1311	1310	1307
11500	1298	1293	1286	1281	1277	1275	1273	1273	1272	1271	1270	1268	1264

Note: Boron Concentration includes an allowance for B^{10} depletion in the main coolant but does not include a 100 ppm Factor of Safety.
No Xenon or Samarium
N-1 Control Rods
A 25 ppm conservatism has been included.
Shutdown = 1.6%$\Delta\rho$
SOURCE: WCAP-16186, Rev 1, Table D.1-1 (1 of 4)

DIABLO CANYON POWER PLANT
STP R-19
ATTACHMENT 9.2

1 AND 2

TITLE: Data Sheet 2, SHUTDOWN MARGIN in MODES 3, 4, and 5, All Rods In or Stuck Rods

A. UNIT CYCLE MODE DATE TIME

WARNING : 1. With stuck* rod(s), perform this calculation within ONE hour and every 24 hours thereafter.
2. If credit is taken for Xenon (Step C.2), repeat this calculation at least every FOUR hours.

B. CURRENT CORE CONDITIONS
(At the time of this SDM Calculation)
1. RCS Temperature (TAVG
(Enter 200 ºF if in MODE 3 or 4 with SI blocked) __________ºF
2. RCS Boron Concentration __________ PPM
3. Core Avg Burnup (PEP R-5) __________ MWD/MTU
4. Time Since Plant Shutdown [ ] N/A
(N/A startup after refueling) __________ HRS

C. XENON AND SAMARIUM WORTH [ ] N/A
(If no credit is taken for Xe/Sm, check N/A, and go to Step D.)

	YES	NO	N/A
1. Power history determined and attached	[ ]	[ ]	[ ]

2. If at equilibrium, estimated Xenon worth from R17-1T-2 or R17-2T-2 (–)__________ PCM
Circle Table per B.3: BOL MOL EOL
3. If at equilibrium, Samarium worth from R17-1T-3 or R17-2T-3 (–)__________PCM
4. Total Xe & Sm worth (C.2 + C.3) or computer output or zero (–)__________ PCM

D. WITHDRAWN ROD/BANK WORTH (N/A if computer or all rods fully inserted) [ ] N/A
(If any rods not FULLY inserted, do not enter Ø pcm in Step D.5.)
1. Most reactive rod worth from R19-1T-1 or R19-2T-1 (+) __________ PCM
2. Number of stuck rods* __________
3. Total stuck rod worth (D.1 x D.2) (+) __________ PCM
4. Worth of withdrawn rod/bank during physics tests N/A [ ](+) __________ PCM
5. Total stuck and withdrawn rod worth (D.3 + D.4) (not zero if any rods out) (+) __________ PCM

E. REQUIRED BORON CONCENTRATION
1. Minimum required boron concentration for Temp B.1 and burnup B.3 from R19-1T-2 or R19-2T-2 __________ PPM
2. Add 100 PPM to concentration E.1 __________ PPM
NOTE: If Steps C & D = N/A, skip to E.8 and enter concentration E.2
3. Boron worth for concentration E.2 and Temp B.1 from R19-1T-5 or R19-2T-5 (–)__________ PCM
Circle Table per B.3: BOL MOL EOL
4. Boron worth multiplier for worth C.4 from R17-1F-1 or R17-2F-1 __________
(If Step C.4 = Ø pcm, enter 1)
5. Boron worth corrected for Xe/Sm (E.3 x E.4) (–)__________ PCM
6. Total worth corrections (C.4 + D.5) ( )__________ PCM
7. Net required boron worth (E.5 - E.6) ( )__________ PCM
If positive, enter Ø pcm
8. Required boron concentration for worth E.7, and Temp B.1 from R19-1T-5 or R19-2T-5 or if steps C & D = N/A concentration from E.2. __________PPM
Select Table per B.3: BOL MOL EOL

F. ACCEPTANCE CRITERIA
1.Is the actual boron concentration B.2 greater than or equal to the required concentration E.8?
[ ] YES (ACCEPTABLE) [ ] NO (NOT ACCEPTABLE)
If this SDM is NOT ACCEPTABLE immediately follow the appropriate Tech Spec Action Statement, notify the SFM, and submit an Action Request.

G. OPTIONAL EXPLICIT SDM CALCULATION
1. Boron worth for concentration B.2 and Temp. B.1 from R19-1T-5 or R19-2T-5 (–)__________ PCM
Select Table per B.3: BOL MOL EOL
2. Total Present Worth (G.1 + C.4 + D.5 - 1600 pcm) (–)__________ PCM
3. Actual SHUTDOWN MARGIN (G.2 - E.3)/(-1000 x E.4) ( )__________ %

* Stuck is defined as untrippable or immovable due to excessive friction or mechanical interference.

UHP DUNV

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1_	_____
	K/A #	APE 054 AA1.02	
	Importance Rating	4.4	4.4

<u>Proposed Question</u>:

The crew is responding to a complete loss of secondary heat sink.

It is determined that AFW pump 1-2 did not start because the control power fuses have blown.

For the current plant conditions, if an operator locally closes the breaker, what indications, if any, should there be in the control room for AFW pump 1-2?

A. No lights, amp, or flow, the pump remains shutdown.

B. Red light, pump amps and flow to steam generators 1 and 2.

C. No lights, but pump amps and flow to steam generators 1 and 2.

D. No pump amps or lights, but indication of flow to all steam generators .

Proposed Answer:

C. No lights, but pump amps and flow to steam generators 1 and 2.

Explanation:
A incorrect, the pump can be started locally.
B incorrect, this is normal indication.
C correct, Due to the loss of control power, all lights on VB-3 will be out and auto starts are defeated. However, the breaker can be closed locally. When the breaker is closed, amps and flow will inform the operator the breaker is closed and the pump is running.
D incorrect, amps will be indicated.

Technical Reference(s): Drawing 437583 – AFW Pump Schematic

Proposed references to be provided to applicants during examination:N/A

Learning Objective: 69130 – Describe Auxiliary Feedwater System instrumentation and controls, including symptoms of failure modes.

Question Source:

New X

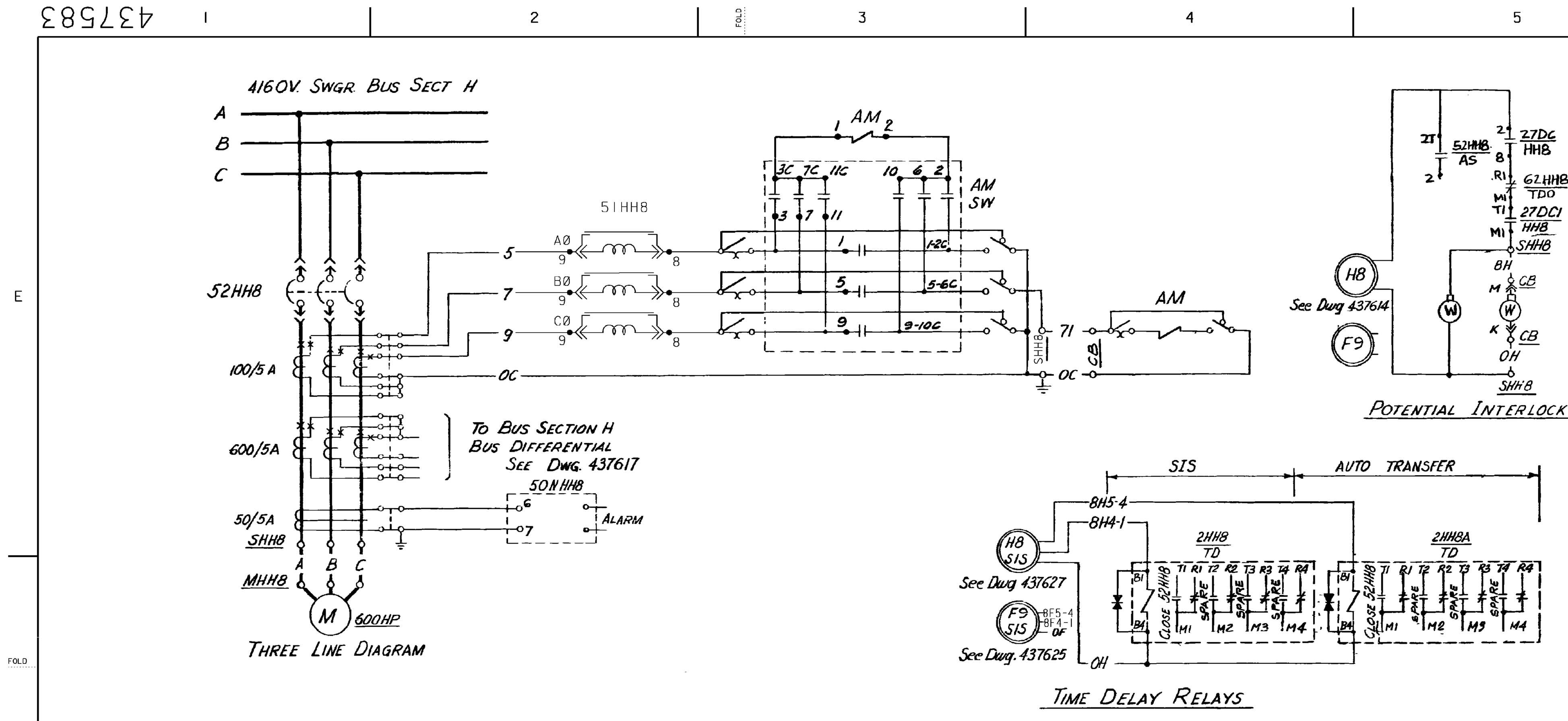
Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: K/A: APE 054 AA1.02 - Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): Manual startup of electric and steam-driven AFW pumps

437583



CONTACTS				A	INT	B	INT.	C
1–1-2C–2	1				X	X	X	X
3–3C	2			X	X			
	3			X	X			
5–5-6C–6	5			X	X		X	X
	6				X	X	X	
7–7C	7				X	X	X	
9–9-10C–10	9			X	X	X	X	
	10						X	X
11–11C	11						X	X

AMMETER SWITCH AM/SW
Used at location SHH8, SHF9

CONTACTS		STOP	NEUT	START	REMARKS
–‖–	A	X			This Dwg.
–‖–	B			X	This Dwg.
–‖–	C		X	X	This Dwg.
–‖–	D	X	X		This Dwg.
–‖–	E		X	X	This Dwg.
–‖–	F			X	Spare

CONTROL SWITCH 52HH8/CS, 52HF9/CS
Used at location CB, HØ1
Spring Return to NEUTRAL

CONTACTS	TRIP	AFTER TRIP	AFTER CLOSE	CLOSE	REMARKS
1 –‖– 1C				X	THIS DWG
2C –‖– 2	X				THIS DWG
3 –‖– 3C	X			X	SPARE
4C –‖– 4				X	SPARE
5 –‖– 5C		X	X		SPARE
5C–					
7 –‖– 7C			X	X	SPARE
7C–					

CONTROL SWITCH 52HH8/CS, 52HF9/CS
Used at location SHH8, SHF9
Spring Return to Neutral

CONTACTS		AUTO	MANUAL	REMARKS
–‖–	A		X	SPARE
–‖–	B	X		THIS DWG
–‖–	C		X	SPARE
–‖–	D	X		SPARE
–‖–	E		X	SPARE
–‖–	F		X	SPARE
–‖–	G		X	SPARE
–‖–	H		X	SPARE

MODE SELECTOR SWITCH 43HH8, 43HF9
USED AT LOCATION HØ1
NO SPRING RETURN

TABLE of DEVICES

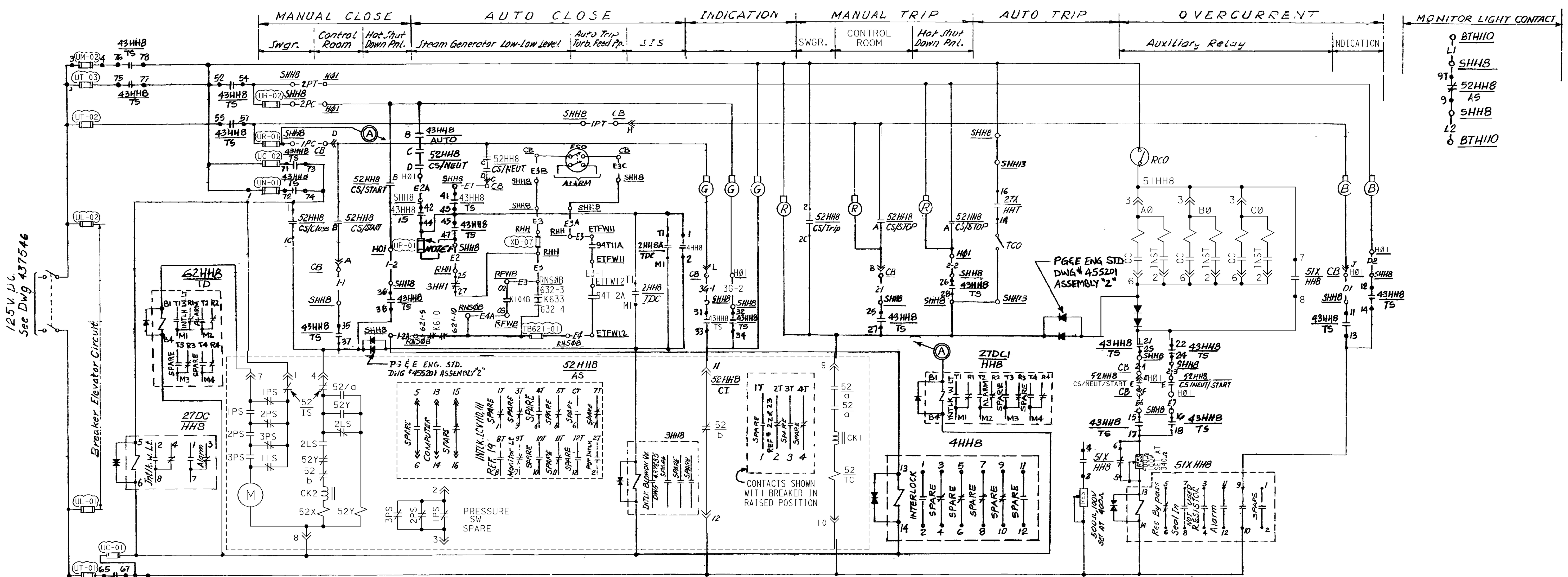
DEVICE Nº	FUNCTION	RATING	MFR	TYPE	CAT./DWG Nº	REMARKS
2HH8 2HF9	Safety Injection Signal Timing Rel.	120V. AC	AGASTAT	ETR	ETR14I3D	Time Delay 2-60 Sec
2HH8A 2HF9A	Auto Transfer Timing Relay	120 V. AC	AGASTAT	ETR	ETR14I3D	Time Delay 2-60 Secs
3HH1, 3HF1	Diesel Auto Transfer Interlock Rel.	—	—	—	(5) (9)	
3HH8, 3HF9	Aux Feedwater Pump Auto Start Rel.	125V DC	G.E. Co.	IC2820	IC2820-A100BB3L	
27X-HHT, -HFT	4KV. Bus U.V. Trip Auxiliary Rel.	—	—	—	(5) (9)	Trips On Bus Undervoltage And Transfer To Diesel
27DC-HH8, HF9	Control Bus DC Undervoltage	125V DC	G.E. Co.	HGA		
27DC1-HH8, HF9	CONTROL BUS DC UNDERVOLTAGE	125VDC	AGASTAT	EGPD		ALARM ONLY
62HH8, HF9	BKR CLOSING SPRING TIME DELAY RELAY	125VDC	AGASTAT	ETR	ETR14D3C	T.D. 1-30 SEC
50NHH8, HF9	Ground Overcurrent Sensor	—	—	—	(2)	Alarm Only
51 HH8, HF9	Overcurrent Relay	—	—	—	(2)	
51XHH8, HF9	Overcurrent Auxiliary Relay	24V. DC	G.E. Co.	HFA	12HFA51A16H	
52HH8, HF9	Air Circuit Breaker	—	—	—	(2)	Rec. No. 663332
52HH8 CI, 52HF9 CI	Air Ckt. Bkr. Cell Interlock	—	—	—		Part Of Switchgear
K633	Steam Generator Lo-Lo Level Aux Rel	—	—	—	(16) (17)(20)	
43HF9/TS	CONTROL TRANSFER SW.	125VDC	ELECTRO SW.	24	249078-2	
43HH8/TS	↓	↓	↓	↓	↓	
4HH8, 4HF9	CONTROL CKT. INTERPOSING RELAY	125VDC	G.E. CO.	HFA	12HFA51A42F	
94T11A, 94T12A	Main Feedwater Pump Tripping Rel.	—	—	—	(7)	
K610 SISA, K610 SISB	Safety Injection Interlock Blk. Rel.	—	—	--	(16)(17)(20)	
K104A	AMSAC SYSTEM (TRAIN A)	—	W.E.	—	(14)	
K104B	AMSAC SYSTEM (TRAIN B)	—	W.E.	—	(14)	

NOTE

1. FUSE PROVIDES ISOLATION FOR CABLE SPREADING ROOM FIRE WHEN TRANSFERED TO HSD PANEL

EQUIPMENT LOCATION NUMBERS

- CB Control Board Condensate & Feedwater
- HØ1 Hot Shut Down Remote Control Panel 1
- MHH8 Auxiliary Feedwater Pump Nº 12
- SHH8 Switchgear 4KV Bus H Cubicle 8
- RHH Engineer Safeguards Relay Board Bus H
- ETFW11 Equipment Cabinet FWP Turbine Nº 11
- RNSØB Nuclear Safeguard Output Rack - Train B
- BTH110 Terminal Box Area H Nº 110



125 V DC CONTROL SCHEMATIC
AUXILIARY FEEDWATER PUMP Nº 12 CIRCUIT Nº H08H00
AUXILIARY FEEDWATER PUMP NO. 13 SIMILAR EXCEPT AS SHOWN IN TABLE

REFERENCES

1. Description of Elec. Schem., Diag., Symbols & Ckt. Designations — Dwg. Nº 050003
2. Single Line Meter & Relay 4160V System Sh 2 (Bus F, G, H) — Dwg. Nº 437533
3. Schematic Diagram Potential & Synchronizing 4160V Sys. — Dwg. Nº 437614
4. Schematic Diagram 4160V. Bus F Automatic Transfer — Dwg. Nº 437625
5. Schematic Diagram 4160V. Bus G Automatic Transfer — Dwg Nº 437626
6. Schematic Diagram 4160V Bus Differential Sh. 2 — Dwg Nº 437617
7. Schematic Diagram Feedwater Turbine Pump — Dwg Nº 437567
8. Single Line Meter & Relay 125 V DC SYSTEM — DWG Nº 437546, 445076
9. Schematic Diagram 4160 V. Bus H Automatic Transfer — Dwg. No. 437627
10. Equipment Location No.s — DWG. NO. 101900
11. Piping Schematic Feedwater System — Dwg. Nº 102003
12. Logic Diagram Auxiliary Feedwater Pumps — Dwg Nº 19409
13. Schematic Diagram Main Annunciator Sh 1 — Dwg. Nº 501121
14. (W) AMSAC SYSTEM — REC. NO. 6008434
15. Schematic Diagram Steam Gen Blowdown Valves — Dwg. Nº 437685
16. (W) Function Diagrams — Rec. No. 663195
17. (W) Solid State Protection System — Rec. No. 663231
18. Schematic Diagram Monitor Light Box C — Dwg. Nº 437698
19. SCHEMATIC DIAG. AUX FW MOV'S — DWG. NO. 437507
20. SCHEMATIC DIAG., SSPS OUTPUT RELAYS, TR. A, TR. B ___ DWGS. 501869, 503088
21. SCHEMATIC DIAG MAIN ANNUNCIATOR — DWG 437100
22. S/D MAIN ANNUNCIATOR SH. 16 — DWG # 501136
23. S/D MAIN ANNUNCIATOR SH 17 — DWG # 501138
22. P.D.S. CONTROL CIRCUITS SCHEMATIC DIAGRAM _______ REC NO. 6009838-69

CONT RM / HSD PNL

DECK		CONTACTS	CONT RM	INTER	HSD PNL	REMARKS
1	1	11–13	X			THIS DWG
	2	12–14			X	THIS DWG
	3	15–17	X			THIS DWG
	4	16–18			X	THIS DWG
2	5	21–23	X			THIS DWG
	6	22–24			X	THIS DWG
	7	25–27	X			THIS DWG
	8	26–28			X	THIS DWG
3	9	31–33	X			THIS DWG
	10	32–34			X	THIS DWG
	11	35–37	X			THIS DWG
	12	36–38			X	THIS DWG
4	13	41–43	X			THIS DWG
	14	42–44			X	THIS DWG
	15	45–47	X			THIS DWG
	16	46–48			X	SPARE
5	17	51–53	X			SPARE
	18	52–54			X	THIS DWG
	19	55–57	X			THIS DWG
	20	56–58			X	SPARE
6	21	61–63	X			SPARE
	22	62–64			X	ALARM, REF 21
	23	65–67	X			THIS DWG
	24	66–68			X	THIS DWG
7	25	71–73	X			THIS DWG
	26	72–74			X	THIS DWG
	27	75–77	X			THIS DWG
	28	76–78			X	THIS DWG

TRANSFER SWITCH: 43HH8/TS, 43HF9/TS
LOC: SHH8, SHF9

MOTOR	AMSAC SYS DEV	AMSAC SYS LOC	SIS BLOCK DEVICE	SIS BLOCK LOC	SIS BLOCK CONTACT INTERLOCK	AUX RELAY DEVICE Nº	INTERLOCK CONTACT CKT NO.	INTLK. AUX. FW. MOV'S 8 –‖– 8T
Nº 12	K104B	RFWB	K610	RNSØB	621-9 –‖– 621-10	3HH8	H08H14	LCV110, 111
Nº 13	K104A	RFWA	K610	RNSØA	621-9 –‖– 621-10	3HF9	F09H14	LCV113, 115

MOTOR	MOTOR LOC.	CKT BKR Nº	CIRCUIT Nº	BUS SECT	125V DC CKT Nº	BUS UNDERVOLTAGE RELAY	BUS UNDERVOLTAGE LOC	BUS UNDERVOLTAGE CONTACT TRIP	AUTO TRANSFER BLOCKING RELAY	AUTO TRANSFER BLOCKING LOC	AUTO TRANSFER BLOCKING CONTACT INTER-LOCK	FEEDWATER TURBINE NO. 11 TRIP RELAY	LOC	CONTACT INTLK	FEEDWATER TURBINE NO. 12 TRIP RELAY	LOC	CONTACT INTLK	STM. GEN. LO-LO LEVEL TRAIN A DEVICE	LOC	CONTACT	STMGEN LO-LO LEVEL TRAIN B DEVICE	LOC	CONTACT	ALARM CIRCUIT Nºs AUTO TRIP TURB FWP CUT OUT	OVERCURRENT	GROUND OVERCURRENT	TRANSFER OF CONTROL	CONT. BUS DC UNDERVOLT.	COMPUTER CKT Nºs RUNNING
Nº 12	MHH8	52HH8	H08H00	H	314D	27X HHT	SHH8	6 –‖– 14	3HH1	RHH	25 –‖– 27	94T11A	ETFW11	–‖–	94T12A	ETFW12	–‖–	—	—	—	K633	RNSØB	632-...	AY2442DH	A1 2415DH	A1 2417DH	AY5072DH	AY4207DH	CY2845DH
Nº 13	MHF9	52HF9	F09H00	F	113D	27X HFT	SHF9	6 –‖– 28	3HF1	RHF	25 –‖– 27	94T11A	ETFW11	–‖–	94T12A	ETFW12	–‖–	K633	RNSØA	632-...	—	—	—	AY2441DF	A1 2416DF	A1 2418DF	AY5070DF	AY4407DF	CY2846DF

P.G.&E. SYSTEM PHASE SEQUENCE A-C-B

NUCLEAR SAFETY RELATED

KEY DWG. SECTION 6

DATE	1/15/93	DESCRIPTION	COORDINATION
DWN.	[illegible]	Revised per FCT 027082	NUCL.
R.E.	[illegible]		MECH.
INDEP. VERIFIER			ELEC.
P.E.	[illegible]		CIVIL
			PIPING
			I&C
		REVISION INFORMATION	

UNIT 1
ELECTRICAL
SCHEMATIC DIAGRAM
AUXILIARY FEEDWATER PUMPS
DIABLO CANYON POWER PLANT
PACIFIC GAS AND ELECTRIC COMPANY
SAN FRANCISCO, CALIFORNIA

DWG SCALE
BILL OF MATL
SUPSDS
SUPSD BY
SHEET NO. 1 OF 1

DRAWING NUMBER 437583 REV 24

U2-441302

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1	_____
	K/A #	EPE 055 EK3.01	
	Importance Rating	2.7	3.4

Proposed Question:

A vital 125 VDC battery is designed to supply DC loads following a loss of its battery charger with no DC load shedding for a minimum of…

A. 1 hour.

B. 2 hours.

C. 4 hours.

D. 8 hours.

Proposed Answer:

B. 2 hours

Explanation: Per DCM No. S-67, The vital battery is sized to supply power to its load for **2 hours** for a design basis accident combined with a loss of a charger or 480V source. Minimum voltage is designed to be at least 112.1V DC to provide sufficient voltage to connected loads.

Technical Reference(s): STG J9 page 1-8, DCM No. S-67

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 7119 - State the design basis of battery capacity.

Question Source:

DCPP P-1368

Question History: Last NRC Exam DCPP 2002

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

10 CFR Part 55 Content: 55.41 41.8
55.43 _____

Comments:

K/A: EPE 055 EK3.01 - Knowledge of the reasons for the following responses as they apply to the Station Blackout: Length of time for which battery capacity is designed

1 P-1368 Points: 1.00 Multiple Choice

WHICH ONE (1) of the following indicates how long the vital 125 VDC batteries can supply DC loads following a loss of all AC with no DC load shedding?

A. 1 hour.

B. 2 hours.

C. 4 hours.

D. 8 hours.

Answer: B

ASSOCIATED INFORMATION:

Associated objective(s):

7119	State the design basis of battery capacity

Reference Id: P-1368
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 1.00
Time to complete: 2
Topic: How long can station batteries supply DC loads.
Cross Reference: STG J-9
Comment: Seabrook exam 1994.

STG J9 page 1-8

Vital 125 VDC Systems	
Designed to ...	**Reason**
be redundant.	A single active or passive failure in the DC system will not prevent safety systems from functioning during emergency core cooling.
allow periodic inspection and maintenance of major components.	To ensure DC system components are maintained in an operational status.
allow periodic testing.	To periodically ensure DC system integrity and operability.
be operable during and after fires, tsunamis, and high winds.	Ensure system remains capable of continuously supplying needed loads during tsunamis, high winds, and fires when emergency and backup loads may be required.
provide necessary DC loads from the vital batteries for 2 hours during a design basis accident coincident with loss of battery charger.	This is the design basis for battery capacity. Batteries must provide needed loads during emergency core cooling until power can be returned to the battery chargers.
perform its safety function during: • loss of main generator • loss of offsite power • degraded offsite power • loss of battery chargers • loss or start failure of diesel generators.	These are high usage events that place large demands on battery systems.

e. The Vital DC System for each unit is designed with three electrically independent and physically separated buses. To meet the single failure criterion, the Vital loads on any two of the three Vital buses are designed to meet the safe shutdown requirements. [1971GDC17]

f. Each Vital primary battery charger is designed to be fed from an independent 480V Vital bus. Use of the Vital backup battery charger may result in entering an Improved Technical Specification Action Statement to LCO 3.8.4. [1971GDC17,1967GDC39]

g. The Vital DC System nominal battery float voltage, when fed from the battery chargers, is 135V DC. Bus voltage limits range from a minimum allowable battery voltage of 112.1V DC (1.9V/cell) to a maximum equalization voltage of 139.8V DC (2.33V/cell). Utilization voltage ranges for power and control is contained in DCM T-23, Miscellaneous Electrical Devices. [Calc 235A-DC thru 235F-DC].

Note: Initial Battery Sizing calculations took credit for 60 cell and found no voltage drop concerns with battery final voltage of 114V DC (1.9VDC X 60 cells). With the new DC System Database Module (DCSDM), battery calculations are in place to take credit for a 59 cell bank and a minimum allowable battery voltage of 112.1VDC (1.9VDC x 59 cells). Although there are 60 cells installed, the battery sizing calculation can now allow one cell to be taken out of service (jumpered out) and still meet the design basis requirements.

h. The Vital DC System is designed to perform its safety function under the following abnormal conditions:

1. Loss of the Main Generator [1971GDC17] (DCM S-63)
2. Loss of offsite power [1971GDC17] (DCM S-63)
3. Degraded offsite power [SER 9] (DCM S-63)
4. Loss of battery chargers or 480V power to battery chargers. [IEEE Std. 308-1971]
5. Loss or failure to start of the Diesel Generators

i. The Vital battery is sized to supply power to its loads for 2 hours for a design basis accident combined with the loss of a charger or 480V source. Minimum voltage is designed to be at least 112.1V DC to provide sufficient voltage to the connected loads.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	APE 056 AK1.01	
	Importance Rating	3.7	4.2

Proposed Question:

Unit 1 trips from full power due to a loss of offsite power. The crew has entered E-0.1 and verifying natural circulation.

Current plant conditions:

- Highest Core Exit Thermocouple = 590ºF and stable
- RCS pressure = 1450 psig
- Steam Generator pressures (all 4) = 985 psig and stable
- RCS hot leg (all loops) = 585ºF and stable
- RCS cold legs:
  1 and 2 = 545ºF and stable
  3 and 4 = 535ºF and stable

Based on the current plant conditions, natural circulation…

A. exists in all loops.

B. exists in loops 1 and 2 only.

C. exists in loops 3 and 4 only.

D. does not exist in any loop.

Proposed Answer:

D. does not exist in any loop.

Explanation:
Per E0.1,
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

A.Incorrect. Less than 20 degrees of subcooling exists. If subcooling was adequate it could be assumed to be occurring in loops 1 and 2.
B. Incorrect. Tcold is appropriate for SG pressure, however, insufficient subcooling currently exists.
C.incorrect. Tcold does not correspond to current SG pressure (Tsat for 1000 psia = 544 F)
D correct. 20 degrees of subcooling required, currently RCS is close to saturation.

Technical Reference(s): E-0.1 step 10

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective: 5432 – Explain the conditions that affect natural circulation cooldown.

Question Source: Modified Bank # DCPP C4P6-5432-1

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 41.5
55.43 _____

Comments: K/A: APE 056 AK1.01 - Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Principle of cooling by natural convection

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. -

1 C4P6-5432-1 Points: 1.00 Multiple Choice

A reactor trip from full power occurred about 15 minutes ago. Off-site power is not available.

Highest core exit thermocouples	590°F and stable
RCS pressure	2235 psig
S/G pressure (all four S/G)	985 psig and stable
RCS hotleg temps (all loops)	580°F and stable
RCS coldleg temps (loops 1&2)	545°F and stable
RCS coldleg temps (loops 3&4)	535°F and stable

Based on this information fully developed natural circulation flow:>>

<QQ 36653(1416)><<A. exists in loops 1 and 2 only.

B. exists in all loops.

C. exists in loops 3 and 4 only.

D. does not exist in any loop.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

5432	Explain the conditions that affect natural circulation cooldown

Reference Id: C4P6-5432-1
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 3.00
Time to complete: 3
Topic: indications of natural circulation
Cross Reference: WOGMCD OR1.3.3
Comment: used for r985
6/9/04 CNH4 - See question M-0012, which was made inactive because "cold trapping" was not adequated explained.

RO QUESTION *12*

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	______
Group #	1	______
K/A #	APE 057 G2.1.32	
Importance	3.4	3.8

Proposed Question:

Unit 1 is at full power.

A loss of Vital 120 VAC bus 12 occurs. Troubleshooting is in progress but the bus is unable to be reenergized.

How long can the unit remain at power before it must be in MODE 3?

A. 2 hours.

B. 6 hours.

C. 8 hours.

D. 30 hours.

Proposed Answer:

C. 8 hours.

Explanation:
A incorrect. This is the allowed outage time for the bus at power.
B incorrect, this is the time allowed to get to mode 3.
C correct, this is the time allowed (2 hours) and the time to be in mode 3 (6 hours_.
D incorrect, this would be correct if only the inverter was inoperable.

Technical Reference(s):

Technical Specification 3.8.7, Inverters-Operating
Technical Specification 3.8.9, Distribution Systems-Operating

Proposed references to be provided to applicants during examination:
Technical Specification 3.8.7, Inverters-Operating
Technical Specification 3.8.9, Distribution Systems-Operating

Learning Objective: 9697H - Identify 3.8 Technical Specification LCOs

Question Source: Bank #
Modified Bank # ________
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 41.10
55.43

Comments:

K/A: APE 057 G2.1.32 - Ability to explain and apply all system limits and precautions.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters-Operating

LCO 3.8.7 Four Class 1E Vital 120 V UPS inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 -----------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital 120 V AC bus de-energized. ---------------------------------- Restore inverter to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage and alignment to required AC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems-Operating

LCO 3.8.9 The required Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution subsystem inoperable.	A.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One 120 VAC vital bus subsystem inoperable.	B.1 Restore 120 VAC vital bus subsystem to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours
E. Two required Class 1E AC, DC, or 120 VAC vital buses with inoperable distribution subsystems that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	7 days

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	APE 058 AK3.02	
	Importance Rating	4.0	4.2

Proposed Question:

A loss of vital DC occurs.

If the main feedwater pumps do not trip, they are runback to minimum speed and locally tripped to prevent which of the following?

A. Pump runout.

B. Pump cavitation.

C. Overpressurizing the pump discharge.

D. Running the pump without lubricating oil.

Proposed Answer:

C. Overpressurizing the pump discharge.

Explanation:

C correct, Caution in OP AP-23 states, "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 overpressurization.

Technical Reference(s): OP AP-23

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 7116 – Explain the consequences of loss of DC vital bus.

Question Source: Bank # P-26240

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 41.4
55.43 _____

Comments:

K/A: APE 058 AK3.02 - Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of dc power

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	Initiate manual unit trip.
b. Either 86G1 (86G2) or 86G11 (86G21) tripped, (VB4)	

****

CAUTION: Personnel shall wear protective flash gear if actuating the Exciter Field Breaker as described in the following step.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
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.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. Verify Both MFPs Tripped:	Remove MFP from service: 1. Verify MFP turbine speed is at minimum. 2. Locally trip MFP.

RO QUESTION *14*

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1	_____
	K/A #	APE 065 G2.1.32	
	Importance	3.4	3.8

Proposed Question:

With Unit 1 at 100% power, a rupture occurs at Air Receiver 0-1.

As instrument air pressure decreases, the reactor will trip due to….

A. MSIVs failing closed.

B. Pressurizer spray valves failing open.

C. Main Feedwater Regulating valves failing closed.

D. Condensate Demineralizer outlet valves failing closed.

Proposed Answer:

C. Main Feedwater Regulating valves failing closed.

Explanation:
Per AP-9, NOTE: FCV-584 may begin to close at 85 psig, and the Main Feed Reg Valves may begin to close at 75 psig.
A incorrect, MSIVs have backup air and are held open by steam flow.
B incorrect, FCV-584 fails closed, but sprays fail closed.
C correct, FRVs closing results in loss of SG level control.
D incorrect, fail as is.

Technical Reference(s): OP AP-9. OIM K-1-1

Proposed references to be provided to applicants during examination: none

Learning Objective: 3541 List the effects that a loss of Instrument Air would have on the plant

Question Source: DCPP A-0741

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 41.10
55.43

Comments: K/A: APE 065 G2.1.32 – Loss of Instrument Air, Ability to explain and apply all system limits and precautions.

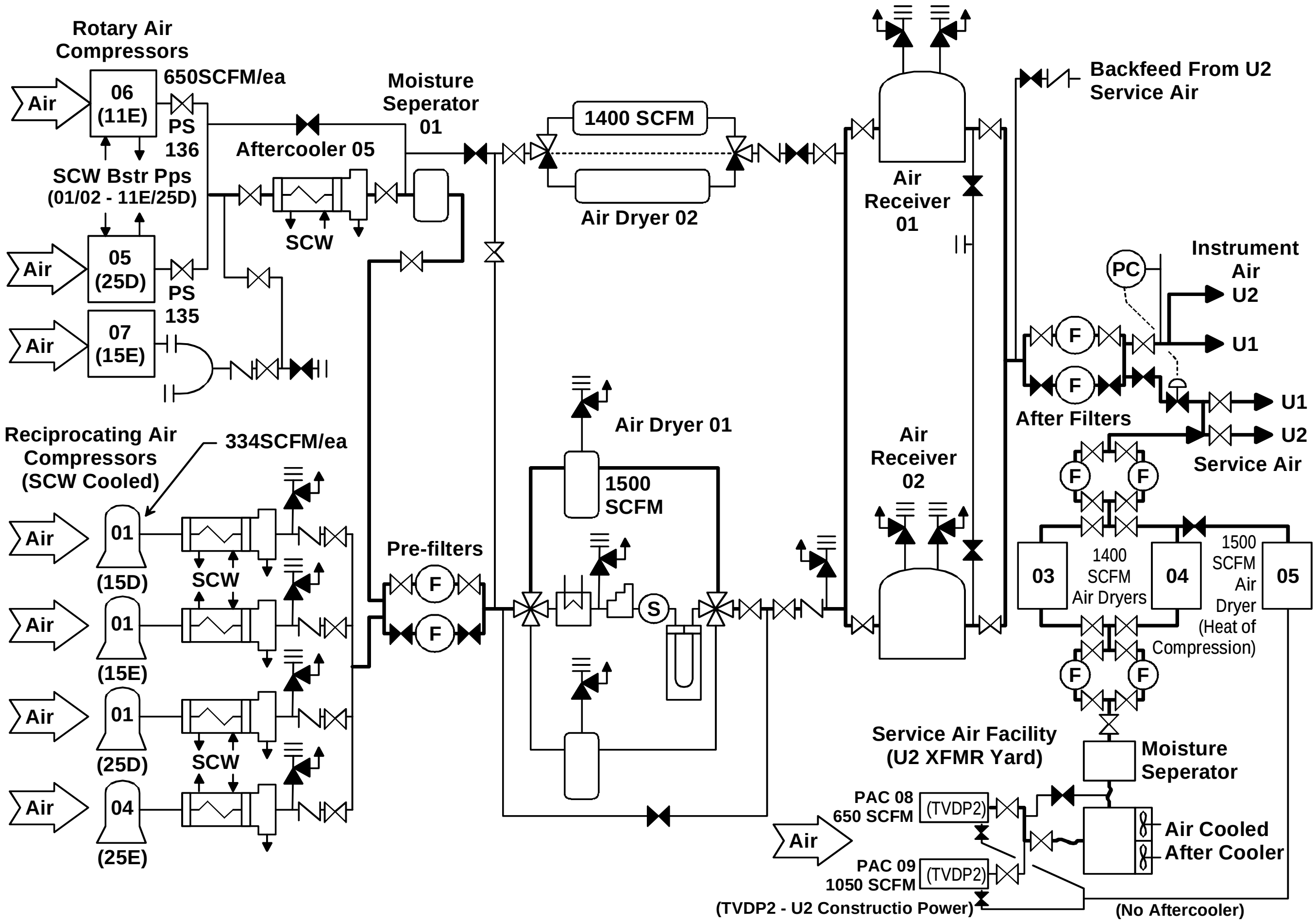
ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: FCV-584 may begin to close at 85 psig, and the Main Feed Reg Valves may begin to close at 75 psig.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **CHECK Control of Plant Systems:** • S/G Levels, PZR Levels, PZR Pressure can be maintained within their normal bands • Reactor Trip NOT Initiated • PK04-11 - OFF • PK04-12 - OFF	a. VERIFY Reactor Tripped. b. IMPLEMENT this procedure in parallel with EOP E-0. c. IMPLEMENT Appendix B (Page 36) of this procedure in parallel with appropriate steps of additional Emergency Procedures (normally EOP E-0.1) d. GO TO EOP E-0.

2. **STABILIZE the Plant:**
   a. Suspend load changes
   b. Suspend other plant activities such as fuel movement, Rx vessel level changes, radwaste discharges, etc.
3. **VERIFY Air Compressors 0-1 thru 0-4 - RUNNING**
4. **MAKE Plant P.A. Announcement:**
   a. Plant is experiencing a loss of instrument air pressure
   b. All plant personnel using air are to stop
   c. Plant personnel inspect their areas for air leaks. If found, report to the Control Room
   d. If any air leaks are reported, GO TO Step 13 (page 7)

Compressed Air System



K-1-1

Rev 19

A-0741 Points: 1.00 Multiple Choice

With the unit at 100% power, a rupture occurs at Air Dryer 0-1. The reactor will trip due to the:

A. Feedwater Regulating valves failing closed.

B. MSIVs failing closed.

C. Condensate Demineralizer outlet valves failing closed.

D. Pressurizer Spray valves failing open.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):
3541 List the effects that a loss of Instrument Air would have on the plant

Reference Id: A-0741
Must appear: No
Status: Active
User Text: 3541.080513
User Number 1: 3.10
User Number 2: 3.40
Difficulty: 4.00
Time to complete: 3
Topic: Loss of instrument air resulting in a reactor trip
Cross Reference: DUTY AREA 51
Comment: Revised stem to allow inclusion of a more plausible distractor. Should be validated. 1/11/97 MAP2
Copied from S-1356; 11/11/96; GES1
Validated question IAW TQ2.ID3, increased difficulty to 4.0 1/23/97 RCWf
Taken active following review, 1/25/97 JMH1
Checked as part of question review for biennial exam 12/22/98 MTC6
Reviewed for 00 biennial exam 1/17/01 mtc6
Reviewed for 02 biennial exam 1/7/03 mtc6
Reviewed for 04 biennial exam 12/21/04 mtc6

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	W/E 04 EK2.1	
	Importance	3.5	3.9

Proposed Question:

The crew is performing the steps of ECA-1.2, LOCA Outside Containment.

As part of the attempt to find the leak, valve 8809A, RHR cold legs 1 and 2, is closed. The operator is instructed to place the series contactor to “CUTIN”.

What purpose does the series contactor in “CUTIN” serve?

A. Applies DC control power to the circuit.

B. Allows the valve to torque out, ensuring positive seating.

C. Prevents the valve from being stopped by breakaway torque as it begins to move.

D. Allows the valve motor to be energized when the control switch is taken to open or close.

Proposed Answer:

D. Allows the valve motor to be energized when the control switch is taken to open or close.

Explanation:

A incorrect, control power comes from the AC source.
B incorrect, the full closed contact is jumpered out, accomplishing this.
C incorrect, this is the purpose of the torque switch bypass contact.
D correct, one contact on each phase closes, which allows AC to be applied to the motor when the switch is taken to the open or close position.

Technical Reference(s): STG B2, RHR, page 2-33
ECA-1.2, step 2

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 7053 – Explain the RHR system design features

Question Source:
New X

Question History: Last NRC Exam N/A

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

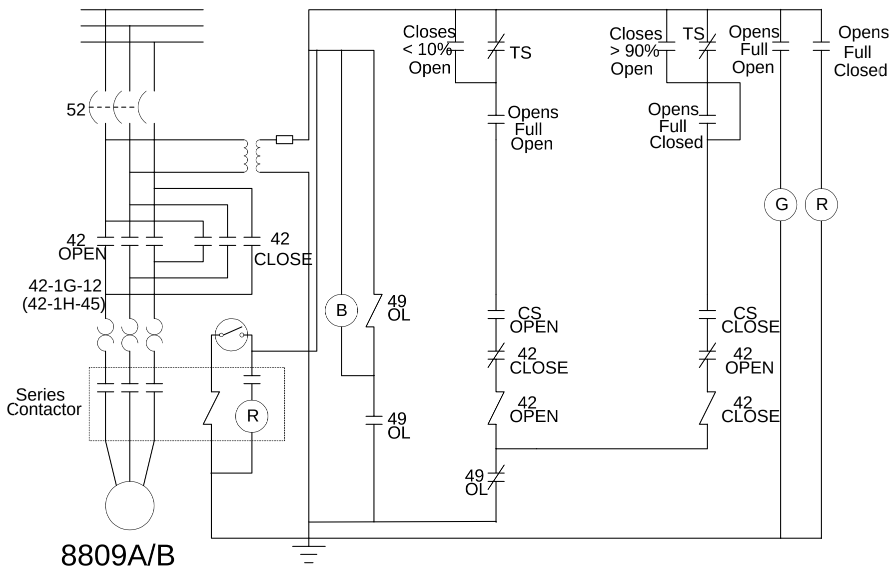
10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: K/A: W/E 04 EK2.1 – LOCA Outside Containment - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

RHR to Cold Leg Isolations, 8809A/B, Continued

Diagram

The contact diagram for valve 8809A/B is shown below.



RHR-21

Indication

The following indications are available for 8809A/B.

8809A/B control switches		
Indicating Light	**Meaning**	**Normal Status**
Red	Valve is full OPEN.	ON
Green	Valve is full CLOSED.	OFF

Note:
- If BOTH Indicating lights are LIT simultaneously, then the valve is in an intermediate position.
- Both valves are normally OPEN.

8809A/B Monitor Light Box A on VB1		
Indicating Light	**Meaning**	**Normal Status**
White	Valve not fully OPEN	OFF

Continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY Proper Valve Alignment:**	Manually close bkrs and valves.
Verify the following valves - CLOSED o 8702, RCS RHR Suct LP4 HL o 8701, RCS RHR Suct LP4 HL o 8703, RHR to hot legs 1 and 2 o 8802A, SI to hot legs 1 and 2 o 8802B, SI to hot legs 3 and 4	IF Valves CANNOT be manually closed, THEN Locally close valves as necessary. -
2. **TRY To Identify And Isolate Break:**	
a. Operate the following valves:	
1) 8809A, RHR to cold legs 1 and 2	
(a) Series contactor CUTIN	
(b) Close 8809A	(b) Locally Close 8809A. -
(c) Check RCS Pressure - STABLE OR INCREASING	(c) Open 8809A AND GO TO Step 2a2. -
(d) GO TO Step 3 (Next Page)	
2) 8809B, RHR to Cold Legs 3 and 4	
(a) Series contactor CUTIN	
(b) Close 8809B	(b) Locally Close 8809B. -
(c) Check RCS Pressure - STABLE OR INCREASING	(c) Open 8809B. -

RO QUESTION *16*

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1	_____
	K/A #	W/E 05 EK3.1	
	Importance	3.4	3.8

Proposed Question:

Why is it critical to quickly establish bleed and feed once wide range steam generator levels are less than 23%?

A. To maintain the steam generator U-tubes “wet”.

B. To ensure adequate ECCS flow to remove decay heat.

C. To extend the time until steam generator dryout occurs.

D. To prevent the steam generators from becoming a heat source.

Proposed Answer:

B. To ensure adequate ECCS flow to remove decay heat.

Explanation:
If Bleed and Feed is not established quickly, and RCS pressure is allowed to rise to the PORV setpoint and reach saturation, ECCS flow will be inadequate to remove decay heat and there will be a deeper and more prolonged core uncovery.

A incorrect, this is why flow is maintained to all steam generators if all are faulted (ECA-2.1)
B Correct. An early initiation of bleed and feed permits a maximum depressurization of the RCS, greater SI flowrate, and ensures effective heat removal. The further the transient is allowed to advance into Period 5 (a period when subcooling is being reduced) before bleed and feed is initiated, the smaller the initial depressurization will be. This results in lower SI flowrate, greater repressurization, and higher net inventory losses.
C incorrect, this is why the RCPs are stopped.
D Incorrect, steam generator temperature is still less than RCS temperature.

Technical Reference(s):

Background, FR-H.1

Proposed references to be provided to applicants during examination: none

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:
Modified Bank # DCPP P-1397

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 41.5/41.10
55.43 _____

Comments: K/A: EPE WE05 EK3.1 - Knowledge of the reasons for the following responses as they apply to the (Loss of Secondary Heat Sink) - Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

P-1397 Points: 1.00 Multiple Choice

EOP FR-H.1, "Response to Loss of Secondary Heat Sink," cautions the operators to initiate RCS bleed and feed if certain conditions are reached. Which one of the following indicates why it is vital that the operators not delay these steps?

A. To minimize core uncovery and prevent an inadequate core cooling condition.

B. To prevent a tube rupture due to excessive primary to secondary differential pressure if the S/Gs boil dry.

C. To prevent lifting pressurizer safeties.

D. To prevent caustic stress corrosion from chemical precipitation on uncovered (dry) tubes.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):
7920 Explain basis of emergency procedure step

Reference Id: P-1397
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 1.00
Time to complete: 2
Topic: LPEH - Basis to not delay actions to BLEED & FEED
Cross Reference: FR-H.1 BKGRND DOC
Comment: IP3 RO exam 7/93
LPE-H Page 20

FR-H.1 Background

2.2.2 RCS Bleed and Feed Heat Removal Transient Description

Bleed and feed is initiated by starting all high pressure SI pumps, verifying SI delivery, and manually opening and holding open all pressurizer PORVs. This will result in rapid RCS depressurization (Figures 2C, 3CB, 4C) as the pressurizer steam bubble and saturated liquid are quickly vented, the pressurizer fills (Figures 2D, 3D, 4D), and a large subcooled liquid flow is established through the pressurizer PORVs. The core exit fluid temperature when the pressurizer PORVs are opened will govern the degree of depressurization since the RCS pressure will decrease until saturation is reached at the hottest point in the system.

Once the saturation pressure is reached in the core, the RCS fluid will begin to heat up (Figures 2A, 3A, 4A and 2B, 3B, 4B) since initially the energy addition and volume swell due to core decay heat generation will exceed the energy and volume removal capability of the pressurizer PORVs and the SG liquid mass inventory remaining (Figures 2G, 3G, 4G). The flow of saturated liquid through the pressurizer PORVs will not remove enough volume to make up for the RCS fluid swell such that RCS pressure will continue to rise until a balance between pressurizer PORV volumetric flow rate and RCS fluid swell plus SI addition is reached. At that point, the RCS pressure will stabilize and remain relatively stable until either a change to all steam flow out the pressurizer PORVs increases the volumetric removal rate or the core partially uncovers reducing the core heat transfer and steam generation rate. The magnitude of the RCS repressurization and the pressure stabilization point will depend upon RCS fluid temperature and core decay heat level at the time bleed and feed is initiated along with pressurizer PORV flow capacity and SI delivery rates. Therefore, all pressurizer PORVs must be held open to minimize both the RCS repressurization and RCS pressure stabilization point in order that SI flow into the RCS may be maximized. During the pressure stabilization period, all available pressurizer PORVs should be maintained open and all available high pressure SI pumps should continue to run to maximize RCS feed flow.

Even with SI flow maximized, RCS inventory will continue to decrease resulting in an eventual emptying of the reactor vessel upper head and a drain down to the hot leg elevation (Figures 2E, 3E, 4E). At that time, steam will begin to be vented out through the hot leg to the pressurizer and pressurizer level may decrease. When the PORV flow becomes a large fraction of steam, the RCS pressure will begin to decrease steadily. This pressure decrease will permit an important increase in SI flow to prevent or minimize core uncovery. As core decay heat generation continues to decrease with time and SI flow increases, the volume removal capability of the pressurizer PORVs will start to exceed the volume addition due to steam generation from core decay heat and water addition from SI. This will be accompanied by increasing net inventory in the RCS, since SI flow will now exceed PORV flow.
An early initiation of bleed and feed permits a maximum depressurization of the RCS, greater SI flowrate, and ensures effective heat removal. The further the transient is allowed to advance into Period 5 (a period when subcooling is being reduced) before bleed and feed is initiated, the smaller the initial depressurization will be. This results in lower SI flowrate, greater repressurization, and higher net inventory losses.

If bleed and feed is initiated earlier than Period 5, then SG liquid mass will still be available to remove a limited amount of energy. This liquid mass can help reduce the extent of repressurization. For plants with smaller PORV flow to power ratios, the liquid mass remaining in the SG is important to the eventual success of bleed and feed in preventing significant core uncovery.

If action is withheld until the start of Period 6, establishing bleed and feed will not prevent significant core uncovery. This is a result of the steam generation rate within the system due to boiling in the core. The volumetric generation of steam and the resultant pressurization of the RCS will fully open the PORVs and will hold them continuously open. The RCS will remain in a high pressure condition until the core uncovers enough to reduce the steam generation rate. The mass flow rate out the PORVs during Period 6 is 50-100 lbm/sec which exceeds the pumped SI capacity of about 40 lbm/sec (290 gpm) for the reference plant. The only possible means for preventing core uncovery once the transient enters into Period 6 is to restore feedwater to the steam generator secondaries (see Reference 1).

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1	_____
	K/A #	WE11 EK2.2	
	Importance	3.9	4.3

Proposed Question:

During the performance of ECA-1.1, "Loss of Emergency Coolant Recirculation Capability" one train of SI flow is established.

What is the establishment of one train of SI flow designed to accomplish?

A. Decrease break flow.

B. Delay the time to RWST depletion.

C. Reduce RCS pressure to allow accumulator injection.

D. Maintain one train of ECCS available for use later, if needed.

Proposed Answer:

B. Delay the time to RWST depletion.

Explanation:

Per ECA-1.1 background: This step instructs the operator to establish one train of SI flow which is one charging/SI, one high-head SI and one low-head SI pump for the reference plant, in order to delay RWST depletion.

The quantified benefit of reducing SI flow is illustrated below in terms of time available before depleting 200,000 gallons of RWST water. These times are typical and are based on containment spray flow of 5600 gpm and SI pumps injecting at zero pressure.

- 12 minutes - Based on maximum safeguards flow
- 17 minutes - Based on minimum safeguards flow
- 150 minutes - Based on decay heat flow and containment spray off

A incorrect. This is why depressurization is performed later.
C incorrect, this is a result of the depressurization performed later.

D incorrect. This is true in procedures such as C.2 when RCPs are stopped to preserve them for use later.

Technical Reference(s): ECA-1.1 step 11, ECA-1.1 background

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 7920 - Explain basis of emergency procedure step.

Question Source: Bank #
Modified Bank # INPO Bank #22432
New ______

Question History: Last NRC Exam DCPP 2002
Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.8
55.43 _____

Comments:
K/A: WE11 EK2.2 - Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Facility*s heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
10. **Check if ECCS Is In Service**	GO TO Step 21 (Page 11)
o SI Pps - ANY RUNNING OR o Charging Injection - NOT ISOLATED OR o RHR Pps - ANY RUNNING	- -

NOTE: The CCP and SI Pps should be stopped in alternate trains when possible.

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
11. **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 12 (Next Page). -
e. Verify RHR Pps - ONLY ONE RUNNING	

Step 12

STEP: Establish One Train Of SI Flow

PURPOSE: To establish one train of SI flow

BASIS:

This step instructs the operator to establish one train of SI flow which is one charging/SI, one high-head SI and one low-head SI pump for the reference plant, in order to delay RWST depletion.

The quantified benefit of reducing SI flow is illustrated below in terms of time available before depleting 200,000 gallons of RWST water. These times are typical and are based on containment spray flow of 5600 gpm and SI pumps injecting at zero pressure.

- o 12 minutes - Based on maximum safeguards flow
- o 17 minutes - Based on minimum safeguards flow
- o 150 minutes - Based on decay heat flow and containment spray off

ACTIONS:

- o Determine if only one charging/SI pump is running
- o Determine if only one high-head SI pump is running
- o Determine if RCS pressure is less than (B.07) psig [(B.08) psig for adverse containment]
- o Determine if only one low-head SI pump is running
- o Stop low-head SI pumps
- o Start or stop charging/SI pumps to establish only one pump running
- o Start or stop high-head SI pumps to establish only one pump running
- o Start or stop low-head SI pumps to establish only one pump running

INSTRUMENTATION:

- o RCS pressure indication
- o Charging/SI pumps status indication
- o High-head SI pumps status indication
- o Low-head SI pumps status indication

STEP DESCRIPTION TABLE FOR ECA-1.1

CONTROL/EQUIPMENT:

Switches for:
- o Charging/SI pumps
- o High-head SI pumps
- o Low-head SI pumps

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

- o (B.07) Shutoff head pressure of the low-head SI pumps, including allowances for normal channel accuracy.
- o (B.08) Shutoff head pressure of the low-head SI pumps, including allowances for normal channel accuracy and post accident transmitter errors.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1	_____
	K/A #	WE 12 EK1.2	
	Importance	3.5	3.8

Proposed Question:

Which one of the following describes a major difference in strategies between EOP E-2, "Faulted Steam Generator Isolation," and EOP ECA-2.1, "Uncontrolled Depressurization of All Steam Generators?"

A. EOP E-2 attempts to prevent thermal shock of the S/Gs. EOP ECA-2.1 does NOT.

B. EOP E-2 maintains the faulted S/G(s) as a heat sink. EOP ECA-2.1 does NOT.

C. EOP E-2 is concerned with excessive RCS cooldown. EOP ECA-2.1 is NOT.

D. EOP E-2 isolates feed flow to the faulted S/Gs. EOP ECA-2.1 does NOT.

Proposed Answer:

D. EOP E-2 isolates feed flow to the faulted S/Gs. EOP ECA-2.1 does NOT.

Explanation:
A incorrect, this why attempts are made to isolate the faulted steam generators.
B incorrect, attempts are made to restore at least one steam generator as a heat sink.
C incorrect, this one reason feed flow is controlled to faulted steam generators.
D correct, in E-2 faulted steam generators are isolated, including feed flow, in ECA-2.1, 25 gpm is maintained to prevent the internals from drying out.

Technical Reference(s): ECA-2.1 background

Proposed references to be provided to applicants during examination:

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source: Bank # P-33611

Question History: Last NRC Exam DCPP RO EXAM 10/94

Question Cognitive Level:

Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.8
55.43 _____

Comments: K/A: WE12 EK1.2 - Knowledge of the operational implications of the following concepts as they apply to the (Uncontrolled Depressurization of all Steam Generators): Normal, abnormal and emergency operating procedures associated with (Uncontrolled Depressurization of all Steam Generators).

3. RECOVERY/RESTORATION TECHNIQUE

An uncontrolled depressurization of all steam generators provides a unique situation to the operator which requires a recovery technique different from that presented in E-2, FAULTED STEAM GENERATOR ISOLATION. The situation is unique since there is no controllable secondary heat sink available to the operator. The objective of the recovery technique contained in ECA-2.1 is to reestablish any secondary pressure boundary, control feed flow, terminate SI and cooldown/depressurize the RCS.

The following subsections provide a summary of the major categories of operator actions and the key utility decision points for guideline ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.

3.1 High Level Action Summary

A high level summary of the actions performed in ECA-2.1 is given on the following page in the form of major action categories. These are discussed below in more detail.

o Reestablish Any Secondary Pressure Boundary

An attempt is made to restore a secondary pressure boundary to the steam generators. If this attempt fails, an operator is dispatched to locally close valves, one loop at a time, while the guideline is continued.

MAJOR ACTION CATEGORIES IN ECA-2.1

- o Reestablish Any Secondary Pressure Boundary

- o Control Feed Flow

- o Terminate SI Flow

- o Cool Down and Place RHR System in Operation

- o Cool Down to Cold Shutdown Conditions

RO QUESTION *19*

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	1	_____
	K/A #	APE 001 AK2.05	
	Importance	2.9	3.1

Proposed Question:

PLANT CONDITIONS:

- Unit 1 is at 50% power.
- Control Bank D is at 200 steps.
- Rods are in automatic.
- Tave/Tref deviation is +3.0ºF.

Control Bank D begins to move. The RO notes the rods are moving and that the RED indicating light is lit on CC1.

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

A. Trip the reactor.

B. Take rods to manual.

C. Verify rods stop when Tave/Tref deviation is less than 0.5ºF.

D. Verify rods stop when Tave/Tref deviation is less than 1.5ºF.

Proposed Answer: B. Take rods to manual.

Explanation: With a +3 Tave/Tref deviation, rods should be moving IN. Given indication is the red light is lit, so rods are moving OUT. Therefore, rods should be placed in manual per OP AP-12A, step 1.

A incorrect, this done if rods do not stop once in manual.
B correct.
C incorrect, this would be true if rods were moving in the correct direction.
D incorrect, this would be considered if operator believed dead band was 1.5 vice 0.5 degrees and rods were moving correctly.

Technical Reference(s): OP AP-12A

ro tier 1 group 2_19.doc

Proposed references to be provided to applicants during examination: None

Learning Objective: 5001 - Identify the normal state of Rod Control system indicating lights during system operation

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 41.6
55.43 _____

Comments: K/A: APE 001 AK2.05 - Knowledge of the interrelations between the Continuous Rod Withdrawal and the following: Rod motion lights

TITLE: Continuous Withdrawal or Insertion of a Control Rod Bank

1 AND 2

07/30/02
EFFECTIVE DATE

SUR FHGXUH#FODVVLILFDWLRQ=##TXDOLW\#UHODWHG#

#

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).

#

#

2.4.2 For continuous rod insertion

a. ROD LO INSERTION LIMIT (PK03-13).

b. ROD LO LO INSERTION LIMIT (PK03-14).

c. PPC RX ALM AXIAL FLUX/ROD POS (PK03-25).

d. TAVG DEVIATION FROM REF (PK04-03).

e. PZR PRESS LOW (PK05-17).

f. PZR LEVEL HI-LO (PK05-21).

g. PZR LEVEL HI-LO CONTROL (PK05-22).

#

#

ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **PLACE Rods In Manual**	
2. **CHECK Rod Motion Stopped**	Manually trip reactor. GO TO EOP E-0, REACTOR TRIP OR SAFETY INJECTION
3. **MATCH TAVG AND TREF:**	
a. Adjust control rods to match TAVG AND TREF	a. Adjust turbine load to match TAVG AND TREF 1) Refer to Technical Specification 3.1.3.1.c (ITS 3.1.4) 2) Have MS initiate troubleshooting and repairs to Rod Control System
b. Adjust turbine load AND boron concentration as necessary to maintain: 1) Control rods ABOVE ROD INSERTION LIMIT 2) Axial flux difference within Tech Spec limits	
c. Notify MS of Rod Control failure	
4. **CONTINUE Normal Plant Operations:**	
a. Return rod control to auto when automatic rod control is restored to normal operation	
5. **RETURN To Procedure and Step in Effect**	

- END -

#

3. APPENDICES

None

4. ATTACHMENTS

None

RO QUESTION *20*

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	2	______
	K/A #	APE 003 AK1.21	
	Importance	2.7	3.2

Proposed Question:

PLANT CONDITIONS:

- Unit 1 is at 100% power
- All rods are at 228 steps.
- AFD is 0.00 on all quadrants.

Control rod C-3 drops to an indicated position of 108 steps.

Which of the following describes how AFD in the area closest to rod C-3 will be affected?

A. AFD on channel N-41 will be more negative.

B. AFD on channel N-43 will be more negative.

C. AFD on channel N-41 will be more positive.

D. AFD on channel N-43 will be more positive.

Proposed Answer: B. AFD on channel N-43 will be more negative.

Explanation:
A incorrect. This is the location if the alignment of the rod orientation to the NI's is not done properly.
B correct. Rod C-3 is on the periphery between 90 and 180 degrees. That location is closest to NI-43. Flux in the area of the dropped rod will be depressed and AFD will go in the negative direction.
C incorrect. AFD will go negative.
D incorrect. AFD will go negative.

Technical Reference(s): OP AP-12C step 6.
DC Unit 1 Cycle 12 NDR figure 2-9
Drawing 1020007, Neutron Detector and Temperature Monitor Locations

Proposed references to be provided to applicants during examination:

- DC Unit 1 Cycle 12 NDR figure 2-9
- Drawing 1020007, Neutron Detector and Temperature Monitor Locations

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

Question Source:
New X

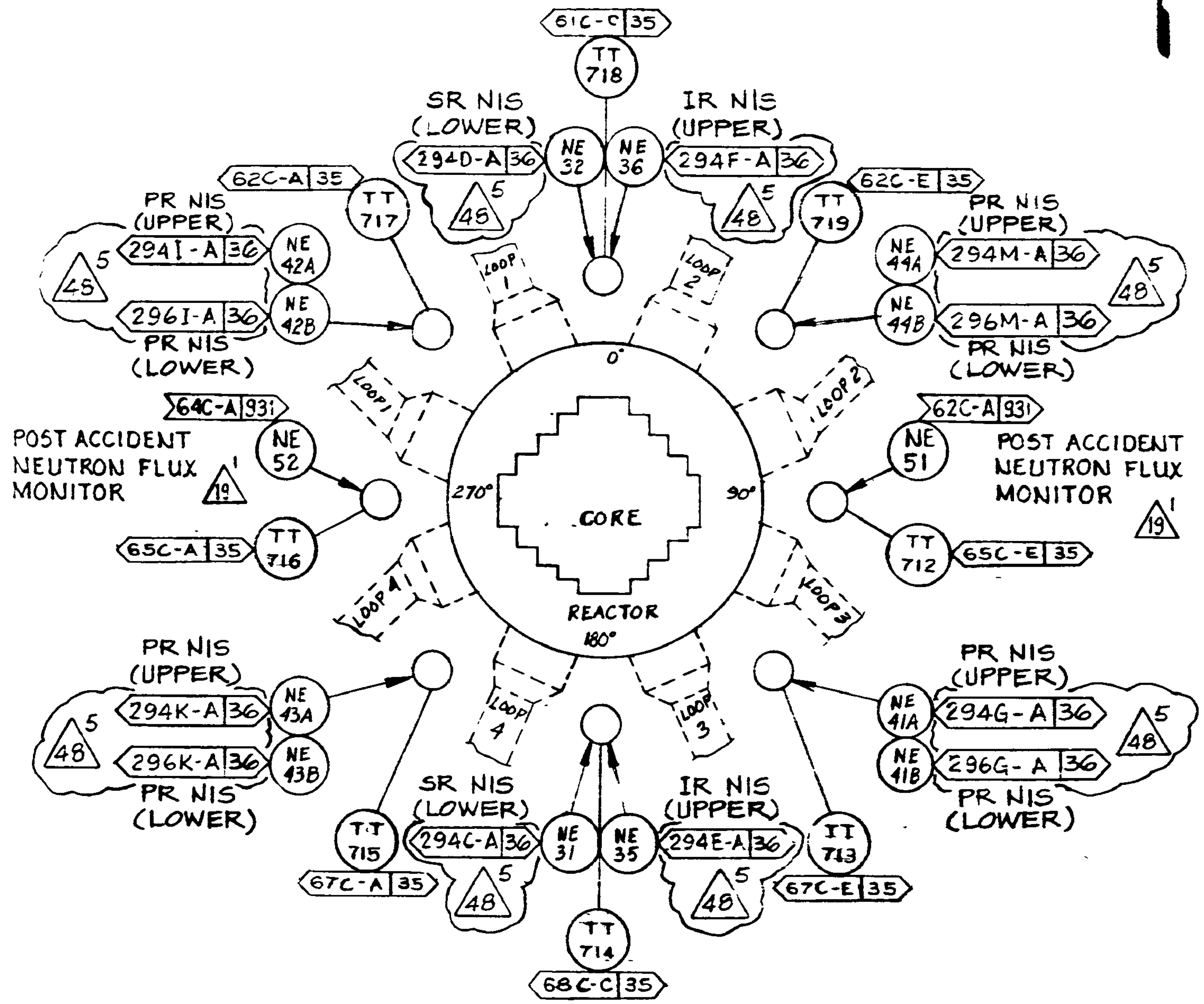
Question History: Last NRC Exam N/A

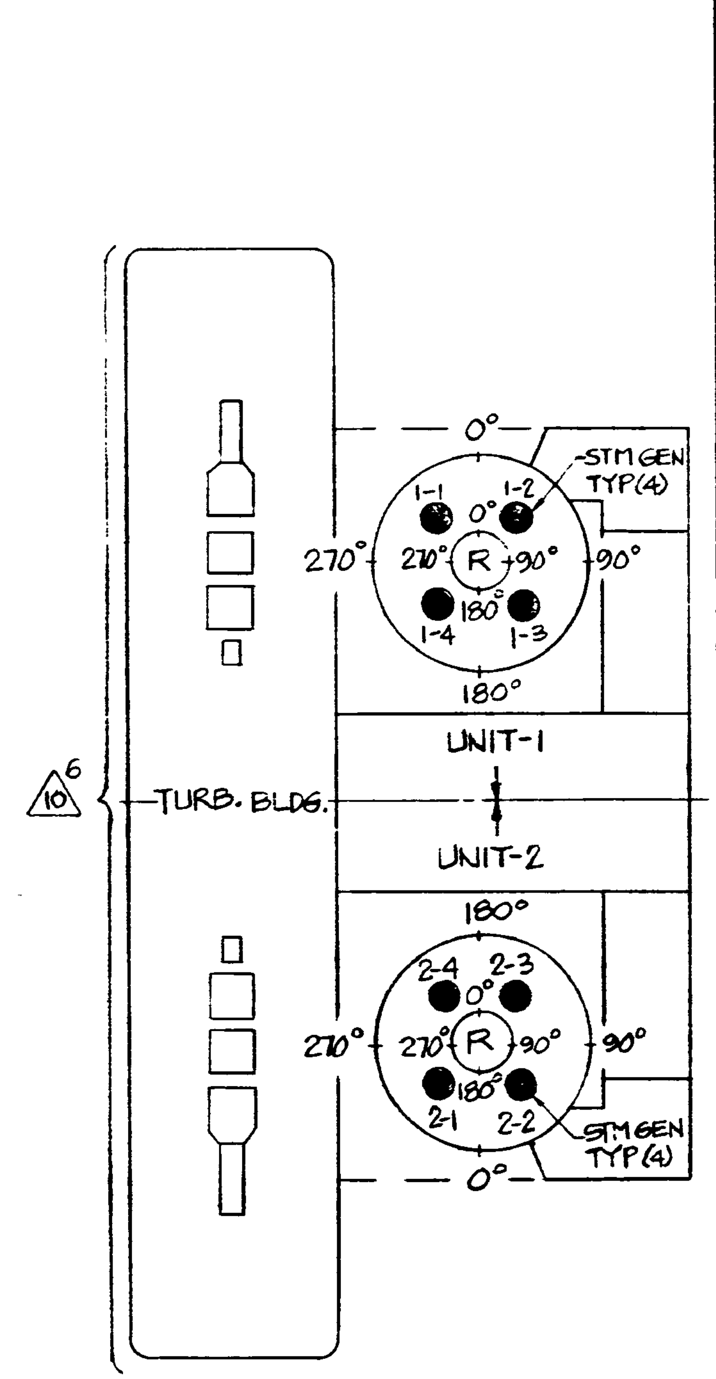
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.6
55.43 _____

Comments: K/A:
APE 003 AK1.21 - Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: Delta flux (Δ I)

60 61 62 63 64 65 66 67 68 69





NEUTRON DETECTOR & TEMPERATURE MONITOR LOCATIONS

REF: DC-663229-31

INDEXED REV 14

UNIT - 1

SCAN 48

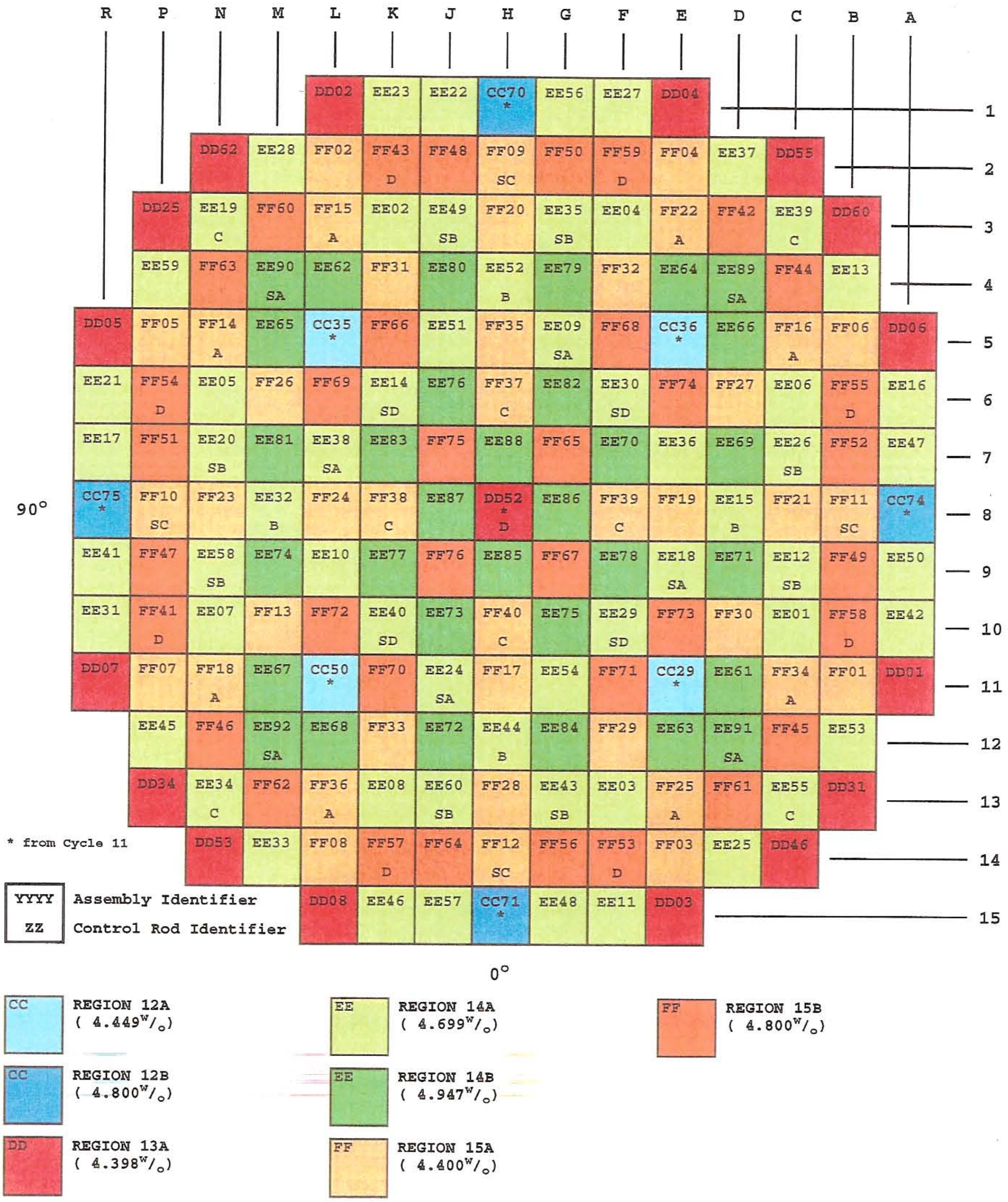
35 M/M NEG

APPD. I&C 38

SCANNED 38

P.G.&E.CO.	SHEET NO. 6 OF SHEETS	
	DRAWING NUMBER 102007	CHANGE 48

Figure 2-1. Diablo Canyon Unit 1, Cycle 13 Core Loading Pattern



Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	2	______
	K/A #	APE 033 AK1.01	
	Importance	2.7	3.0

<u>Proposed Question</u>:

Unit 1 shutdown is being performed.

Intermediate range channel N35 is overcompensated.

Which of the following describes the effect of the overcompensation on P-6 operation?

A. P-6 will energize both source ranges prematurely.

B. P-6 will require manual action to energize both source ranges.

C. P-6 will automatically energize one of the source ranges when N36 reaches the setpoint.

D. P-6 will automatically energize both of the source ranges when N36 reaches the setpoint.

Proposed Answer:

D. P-6 will automatically energize both of the source ranges when N36 reaches the setpoint.

Explanation:
A incorrect. This would be true if the coincidence was 1/2.
B incorrect. This would be true is N35 was UNDERCOMPENSATED.
C incorrect. The SR channels are not “train” dependent.
D correct. Due to overcompensation, N35 will decrease below the P-6 setpoint ahead of N36, however, both IR channels must be less than $5x10^{-11}$ to energize either (both) source ranges. Therefore, when N36 reaches the setpoint, both SR will energize.

Technical Reference(s): OIM pages B-4-3 and B-6-2a

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 5229 - Explain the effect of Intermediate Range channel failures in the NIS

Question Source:

Modified Bank # INPO 23213

Question History: Last NRC Exam IP2, 2003

Question Cognitive Level:

Memory or Fundamental Knowledge ____

Comprehension or Analysis X

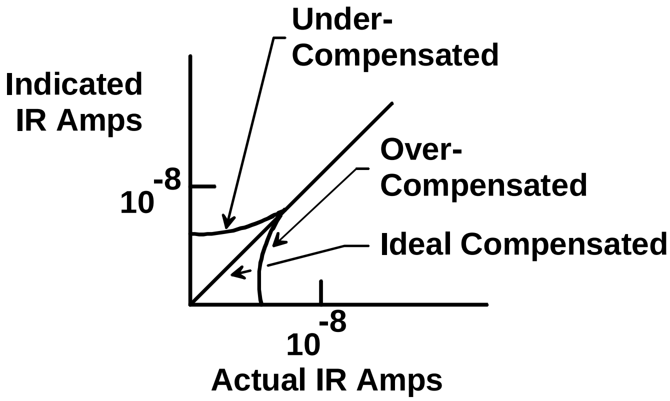
10 CFR Part 55 Content: 55.41 41.7

55.43 _____

Comments:

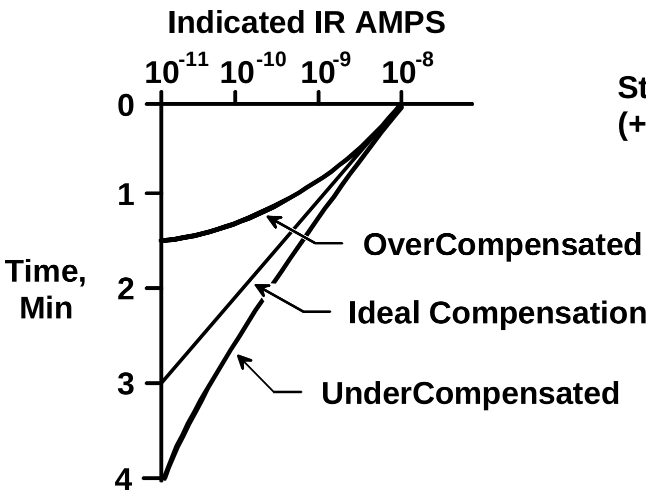
K/A: APE 033 AK1.01 - Knowledge of the operational implications of the following concepts as they apply to Loss of Intermediate Range Nuclear Instrumentation: Effects of voltage changes on performance

Intermediate Range Compensation



If the IR detector is undercompensated, the indicated IR neutron level will be greater than the actual level.

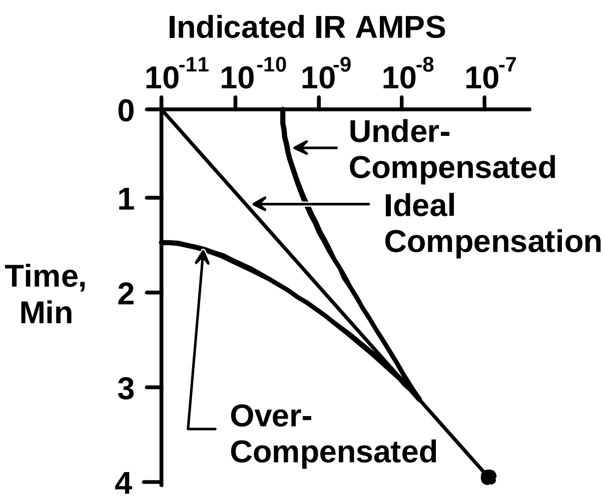
An overcompensated IR detector the indicated IR neutron level will be less than the actual level.



Stable (+) SUR

If power increases on stable +1 DPM startup rate (SUR), the indicated SUR will be less than the actual SUR if the detector is undercompensated.

Overcompensation has the opposite effect.

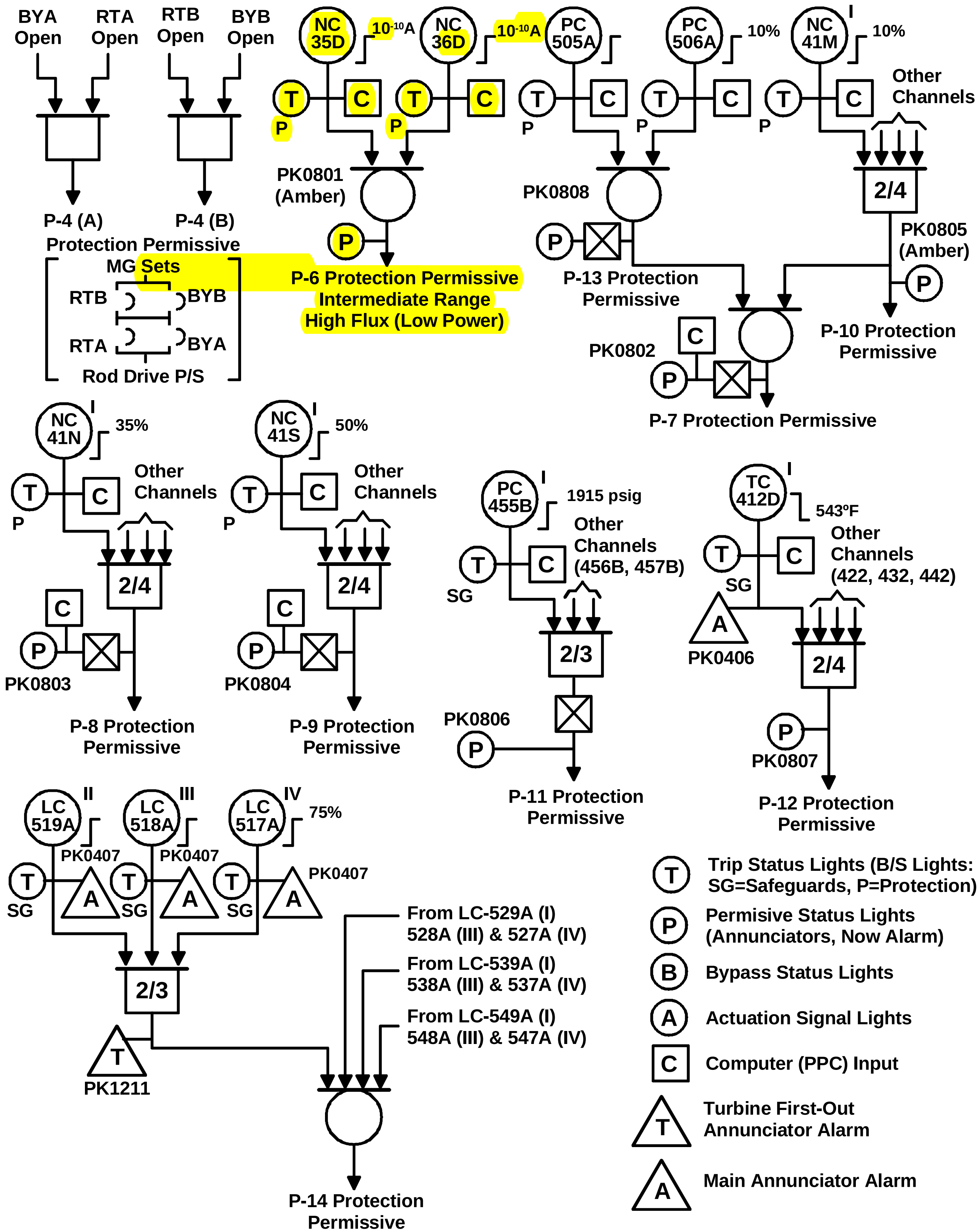


Stable (-) SUR

If power decreases on a stable -.3 DPM startup rate (SUR), the indicated SUR will be less than the actual SUR if the detector is undercompensated.

Overcompensation has the opposite effect.

Protection Permissives Logics



RO QUESTION 22

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	2	_____
	K/A #	APE 036 AK3.02	
	Importance	2.9	3.6

Proposed Question:

The crew secures refueling operation and closes Fuel Transfer Gate valve SFS-50 due to a high radiation alarm in the Spent Fuel Building.

With the gate still closed, and the PIT position selected, the operator on the reactor side accidentally presses the conveyor start pushbutton on Fuel Transfer Control Panel 1-1.

Which of the following is the expected conveyor system response?

A. The transfer cart moves until it contacts SFS-50, it will then stop due to overload or winch over temperature interlock.

B. The transfer cart moves until it contacts SFS-50, it will then stop due to cable tension interlock.

C. The transfer cart will start but almost immediately stop due to cable tension interlock.

D. The transfer cart will not move due to the gate valve open interlock.

Proposed Answer:

D. The transfer cart will not move due to the gate valve open interlock.

Explanation: Interlock with SFS-50 prevents transfer cart movement until the valve is open.
A, B and C all involve actual conveyor system interlocks which could actuate if the system were in operation.

Technical Reference(s): B8, Fuel Handling System, pages 2.2-13, 14 and 15

Proposed references to be provided to applicants during examination: None

Learning Objective: 6586 - Analyze the interlocks associated with refueling equipment controls.

Question Source:

New X

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 41.13

55.43 _____

Comments:

K/A: APE 036 AK3.02 - Knowledge of the reasons for the following responses as they apply to Fuel Handling Incidents: Interlocks associated with fuel handling equipment

Interlocks
Obj 13, 18, 24

Several interlocks exist that must be satisfied to allow conveyor motion. The table below summarizes the interlocks on the conveyor operation.

Interlock (GL is GEMCO switch)	**Is active when..**	**Action**
Conveyor at Reactor	conveyor is not at the reactor end of the tracks	• Stops the conveyor when it reaches its destination. • Controls the speed of the conveyor to a: ♦ Slow speed of 7 fpm as it approaches one foot from the ends ♦ Fast speed of 20 fpm for normal travel
Conveyor at Pit	conveyor is not at the pit end of the tracks	
Pit side frame down * (GL-5, pit)	pit side frame is down	Prevents conveyor operation unless frames are down.
Reactor side frame down * (GL-3, reactor)	reactor side frame is down	
Cable tension LS-2, LS-4	either reactor or pit cable is under an overload condition	Stops conveyor operation to prevent component damage.
Winch over temp	winch motor is hot	
Under voltage	an undervoltage condition exists	
Motor overload	motor thermal overloads have tripped	
Gate valve open, POS-162	fuel transfer gate valve is not full open	Prevents conveyor from striking the valve disk.

* This interlock may be bypassed at the control panel.

Continued on next page

Logic
Obj 9, 16, 24

The operating logic of the Containment conveyor is described in the following table.

At the Fuel Transfer Control Panel 1-1 (Reactor Side):

If the .... switch	is ...	Then the Conveyor will ....
CONVEYOR START	depressed	start if all interlocks are satisfied.
CONVEYOR STOP		stop.
REACTOR	in REACTOR position	go to the reactor when the START button is pushed.
PIT	in PIT position	go to the pit when the START button is pushed.

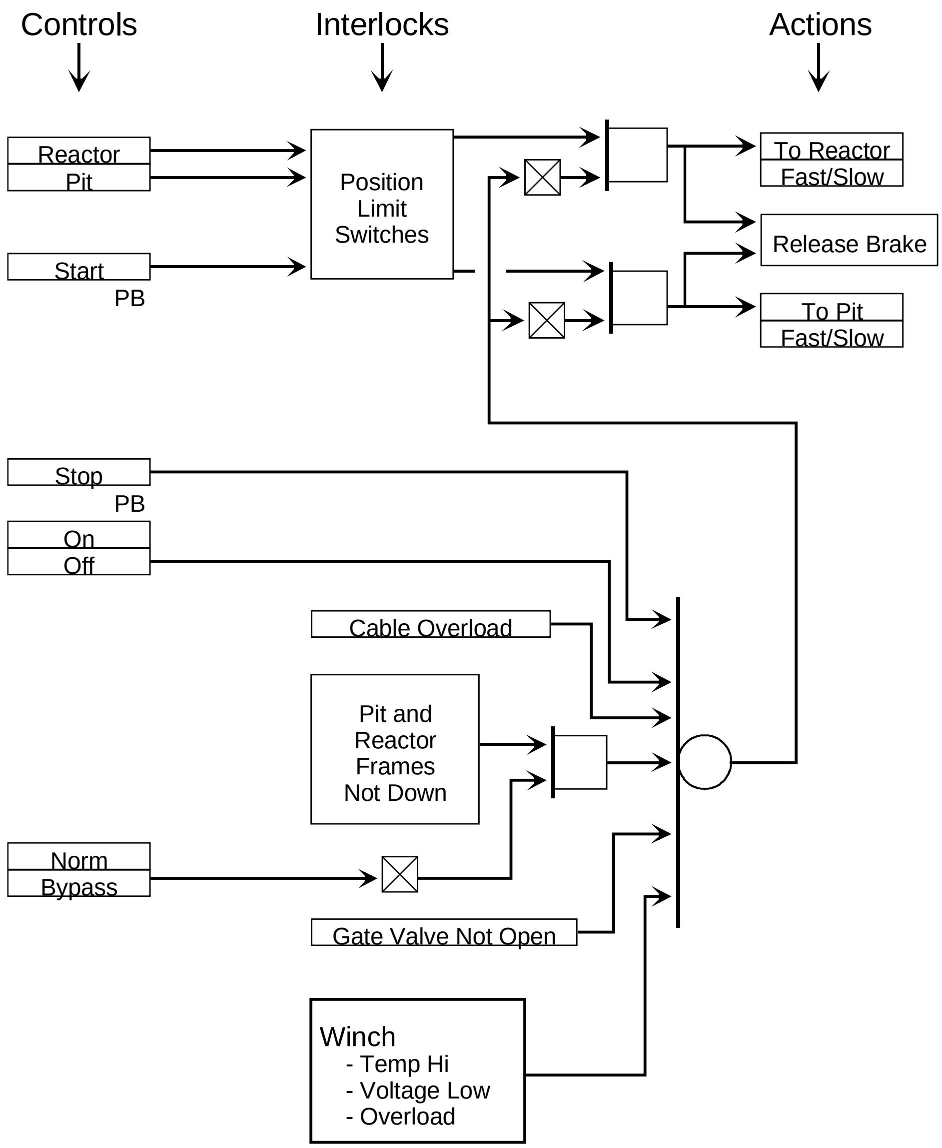
At the Fuel Transfer Control Panel 1-2 (Pit Side):

If the .... switch	is ...	Then the Conveyor will ....
CONVEYER CONTROL	ON	start if all interlocks are satisfied.
	OFF	stop.

Continued on next page

Logic diagram
Obj 17

The diagram below shows the control logic of the conveyor controls.



FHS-42

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	2	_____
	K/A #	APE 060 AA1.02	
	Importance	2.9	3.1

Proposed Question:

Several Auxiliary Building radiation alarms are received. It is confirmed that a Waste Gas Decay Tank has ruptured, and is depressurizing into the Auxiliary Building.

What action must be taken to prevent the offsite release of radioactive particulate and iodine?

A. Push "Status Reset" at POV1 and POV2, and reset the "S" signal.

B. Locally close dampers that isolate the Waste Gas Decay Tank rooms.

C. Stop all Aux Bldg supply and exhaust fans, and energize charcoal heaters.

D. Select "S" signal test, secure one Aux Bldg Ventilation train, and energize charcoal heaters.

Proposed Answer:

D. Select "S" signal test, secure one Aux Bldg Ventilation train, and energize charcoal heaters.

Explanation:
A incorrect, this is done to return the Aux Bldg ventilation to BLDG and SFGDS mode.
B incorrect, this would not prevent off site release.
C incorrect, this will not completely isolate the release
D correct, per OP AP-14 step 3

Technical Reference(s):
OP AP-14, Tank Ruptures step 3A
LPA-14, Tank Ruptures, page 7 of 24

Proposed references to be provided to applicants during examination: None

Learning Objective: 3477 - Describe the major actions of OP AP-14, Tank Ruptures.
5512 State the alignments for Auxiliary Building Ventilation system

Question Source: Bank # DCPP P-0764

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 41.13
55.43 _____

Comments:
K/A: APE 060 AA1.02 - Ability to operate and / or monitor the following as they apply to the Accidental Gaseous Radwaste: Ventilation System

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION: With Aux Bldg ventilation in SFGDS Only Mode, airborne activity will increase. Respiratory protection may be required.

****

3. SAMPLE Aux Bldg for Airborne Activity:

- When radiation protection confirms airborne activity is LESS THAN 10 FR 20 Appendix B limits, THEN return Aux Bldg ventilation to BLDG and SFGDS Mode:

  a. Restore control power to Aux Bldg Exhaust Fan at RCV1/RCV2

  b. Push "Status Reset" at POV1 and POV2

  c. Reset "S" Signal at POV1 and POV2

****

CAUTION: The SWP for recovery activities must incorporate clothing and respiratory protection requirements from EP RB-2.

****

4. IMPLEMENT Applicable Radiological Emergency Procedures:

- RB-2 EMERGENCY EXPOSURE GUIDES
- CP M-13 PERSONNEL INJURY OR ILLNESS WITH RADIOACTIVE CONTAMINATION OR PERSONNEL OVEREXPOSURE
- R-3 RELEASE OF RADIOACTIVE LIQUIDS

Minimize Dose Steps, Continued

Steps 1-4
Obj 6

The major actions of steps 1 through 4 are shown in the table below.

Step	Action	Bases/Discussion
1	Alert Plant Personnel Commence EP R-2	Provide an initial off-[CNH41]dose assessment, to determine protective action guidelines and emergency classification
2	Evacuate Personnel	Minimize dose
3	Place Aux Bldg Ventilation in the Safeguards Only Mode, discharging through the Charcoal filters.	Minimize release of radioactive iodine to the environment • "S" Signal Test at POV1 and 2 puts the system in Bldg and Safeguards mode and puts the charcoal filter in service • Securing a fan at RCV1 or 2 forces the system in Safeguards Only mode • Safeguards only mode does not provide ventilation to the general Aux Bldg areas
4	Sample Aux Bldg for airborne activity. *When* Radiation Protection confirms that airborne activity is < 10FR20, appendix B limits, *then* Return Aux Bldg Ventilation to a normal Building and Safeguards lineup.	Determine respiratory protection requirements. • The CAUTION serves as a reminder when personnel are subsequently sent to identify and isolate the ruptured [CNH42] Maximizing ventilation will reduce airborne levels within the Aux Building.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE 061 AA2.04	
	Importance	3.1	3.5

Proposed Question:

PK11-10, FHB High Radiation RE-58 and 59, alarms in the Unit 1 Control Room.

If the alarming monitor is functioning properly, in addition to the GREEN power available light, what should be the indication on PAM2?

A. Red and Amber lights and audible alarm.

B. Red and Amber lights, only.

C. Red light, only.

D. Red or Amber light lit, either will cause the alarm.

Proposed Answer:

B. Red and Amber lights, only.

Explanation:
A incorrect, the audible alarm is at the local panel.
B correct, as reading exceeds the trip 1 setpoint, the amber light comes on and at trip 2, the red light comes on and the alarm actuates.
C incorrect, amber light remains on.
D incorrect, only trip 2 gives the alarm.

Technical Reference(s): G4A, Radiation Monitoring, pages 2.2-22 and 3-11

Proposed references to be provided to applicants during examination: None

Learning Objective: 8504 - Identify the location of Radiation Monitoring system alarm indications in Control Room.

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 41.11

55.43

Comments:

K/A: APE 061 AA2.04 - Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Whether an alarm channel is functioning properly

RM-58, 59 Fuel Handling Building Monitors, Continued

Indications
Obj 8, 9, 10, 17

The following indications are available on RM-58 and 59 control modules.

Control Module

Lamp Indication	Meaning	Normal Status
RED	The measured radiation has exceed the high setpoint.	OFF
AMBER	The measured radiation has exceed the alert setpoint.	
GREEN	Indicates that the Operate Selector switch is in the OPERATE position.	ON

Local Indication Panel

Lamp Indication	Meaning	Normal status
RED	Measured radiation > high setpoint	OFF
AMBER	Measured radiation > alert setpoint	
GREEN	The following conditions are met: • power is available, and • meter reading is greater than 0.1 mR/hr.	ON

Indication	Can be read on ...	Normal reading
RM-58	Control Module meter Local Indication Panel meter Data Logger PPC	0.2 mR/hr
RM-59		0.1 mR/hr

Alarms
Obj 18

The following alarms exist for RM-58 and 59.

Parameter	Source	Location
FHB High Radiation	• RM-58 • RM-59	
Rad Monitor System failure/CVI Bypass	• 74HRP2G • 74HRP2H	Control Room

Abnormal Operations, Continued

Alarm input
Obj 36, 37

The annunciators associated with the Rad Monitoring System are listed below.

Control Room

Location	Title	Parameter	Setpoint
PK11-09	RE-11 AND RE-12 LOW FLOW	Low flow on: • RE-11	< normal
PK11-10	FHB HIGH RADIATION RE-58 AND 59	High radiation alarm on either: • RE-58 • RE-59	
PK11-13	POST ACDNT SMPL ROOM RAD MON	Either of the following on RM-48: • High radiation alarm • Failure alarm	
PK11-17	S.G. BLOWDOWN HI RAD	High radiation alarm on either: • RE-19 • RE-23	
PK11-18	MAIN STM LINE HIGH RAD	High radiation alarm on any of the following: • RE-71 • RE-72 • RE-73 • RE-74	See the I&C RMS Data Book in the Control Room for setpoints
PK11-19	CONTMT RADIATION (HI LVL RAD MON SYSTEM)	High alarm on either of the following: • RE-30 • RE-31	
PK11-21	HIGH RADIATION	High radiation alarm on any of the following: RE-01 RE-12 RE-02 RE-13 RE-03 RE-17A RE-04 RE-17B RE-06 RE-18 RE-10 RE-22 RE-11 RE-44A* RE-44B* RE-07	

*These are digital radiation monitors and are covered in STG G-4b.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	2	_____
	K/A #	EPE E01 EK3.3	
	Importance	3.5	3.3

Proposed Question:

PLANT CONDITIONS:

- The crew is performing the actions of E-0, Reactor Trip or Safety Injection following SI actuation from full power.
- All systems operated as designed.
- Tave is being maintained by condenser steam dumps.
- No operator actions have been taken, except to throttle AFW for temperature control
- At the diagnosis steps in E-0, the crew fails to positively identify the cause of the SI but suspect a tube rupture has occurred.

The SFM directs the BOPCO to open the Blowdown Inside Containment isolations (FCV-760, 761, 762, 763).

The BOPCO reports that when the OPEN/AUTO/CLOSE switch for each isolation valve was taken to OPEN, the valves did not move.

Which of the following explains why the valves will not open?

A. Instrument air to containment is not restored.

B. High Radiation on RE-19 or RE-23.

C. Main Steam isolation.

D. SI not reset.

Proposed Answer:

A. Instrument air to containment is not restored.

Explanation: The IC valves close on Main Steam Isolation or loss of air. Because MSI has not occurred and the valves are closed, a loss of air must have occurred. Restoring IA to containment has not been done, so the valves will not open.

B incorrect, this isolates for the OC isolation or sample valves.

C incorrect, temperature control is with the condenser steam dumps, so MSI has not occurred.

D incorrect, resetting SI has no effect on operation of the valves

Technical Reference(s):
D-2, Steam Generator Blowdown System.

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

Learning Objective: 8755 State Steam Generator Blowdown system automatic isolation signals

Question Source: Bank #
Modified Bank # S-71159
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:
K/A: EPE E01 EK3.3 - Knowledge of the reasons for the following responses as they apply to the (Reactor Trip or Safety Injection/Rediagnosis) - Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

1 S-71159 Points: 1.00 Multiple Choice

Unit 2 has experienced a Hi Rad signal on RE-23 and a Phase A Isolation Signal.
The Shift Chem Tech wishes to sample the Steam Generators.
The Control Operator has all the valves open except the IC Blowdown Isolation Valves.

What is the most likely problem?

A. Air to containment NOT Open

B. Phase A NOT RESET

C. RE-19/23 CUTOUT Switch is in the CUTOUT position.

D. AFW AUTO start signal NOT RESET.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8755	State Steam Generator Blowdown system automatic isolation signals

Reference Id: S-71159
Must appear: No
Status: Active
User Text: 8724.13049E
User Number 1: 3.10
User Number 2: 3.40
Difficulty: 3.00
Time to complete: 2
Topic: D2 SGBD Islation Signal
Cross Reference: STGD2
Comment: This question was developed for L031, System Phase, rlf1,10/17/03

Containment Isolation Valves, Continued

Logic
Obj 9,10,11

The logic for Containment Isolation Valve operation is described below.

For FCV-760, 761, 762, and 763

IF the OPEN/CLOSE switch is in...	THEN the valve will...
Neutral after OPEN (spring return from OPEN)	CLOSE on: • Main Steam Isolation. • loss of air.
OPEN	OPEN if closed... • OPEN will override close signal as long as switch is held in OPEN and air is available.
CLOSE	CLOSE.

For FCV-151, 154, 157, and 160

IF the OPEN/AUTO/CLOSE switch is in...	THEN the valve will...
AUTO (spring return from OPEN)	CLOSE when it receives any of the following: • Containment Isolation Signal Phase A. • AFW pump auto start signal. • SGBD HIGH radiation signal from either RE-19 or RE-23. • AMSAC (Actually causes AFW pump auto start which in turn causes isolation.) • loss of power to RE-19 or RE-23.
OPEN	OPEN if closed. • overrides the close signal as long as switch is held in OPEN
CLOSE	CLOSE

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	2	_____
	K/A #	E03 G2.1.23	
	Importance	3.9	4.0

Proposed Question:

PLANT CONDITIONS:

- A LOCA and loss of offsite power has occurred on Unit 1.
- The crew has entered EOP E-1.2, Post-LOCA Cooldown and Depressurization.
- All AFW pumps are running
- Steam generator narrow range levels:
    - o 1-1 = 22%
    - o 1-2 = 18%
    - o 1-3 = 20%
    - o 1-4 = 6%

The crew is checking if the TDAFW pump is required for heat removal.

Which of the following actions should be taken with the TDAFW pump?

A. Leave the TDAFW pump running, it is needed for heat removal.

B. Shutdown the TDAFW pump by momentarily placing the control switch for FCV-95 in CLOSE.

C. Shutdown the TDAFW pump by placing the control switches for FCV-37 and FCV-38 in CLOSE.

D. Dispatch an operator to open the breaker for FCV-95.

Proposed Answer:

C. Shutdown the TDAFW pump by placing the control switches for FCV-37 and FCV-38 in CLOSE.

Explanation:
A incorrect, 16% necessary in 3 of 4 steam generators (incorporates adverse containment values)
B incorrect. Due to the loss of offsite power, closing FCV-95 will not shutdown the pump. When the switch is released, FCV-95 will reopen. This is how it would normally be shutdown.
C correct, to shutdown the pump, both steam supplies must be closed.
D incorrect, opening the breaker removes power from the solenoid but the valve would also have to be manually closed.

Technical Reference(s):
E-1.2, step 8.
D-1, Auxiliary Feedwater System, page 2-18, 2-23, 2-28

Proposed references to be provided to applicants during examination: none

Learning Objective: · 3891 - Explain the operation of turbine-driven AFW pump.

Question Source: New

Question History: Last NRC Exam N/A
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

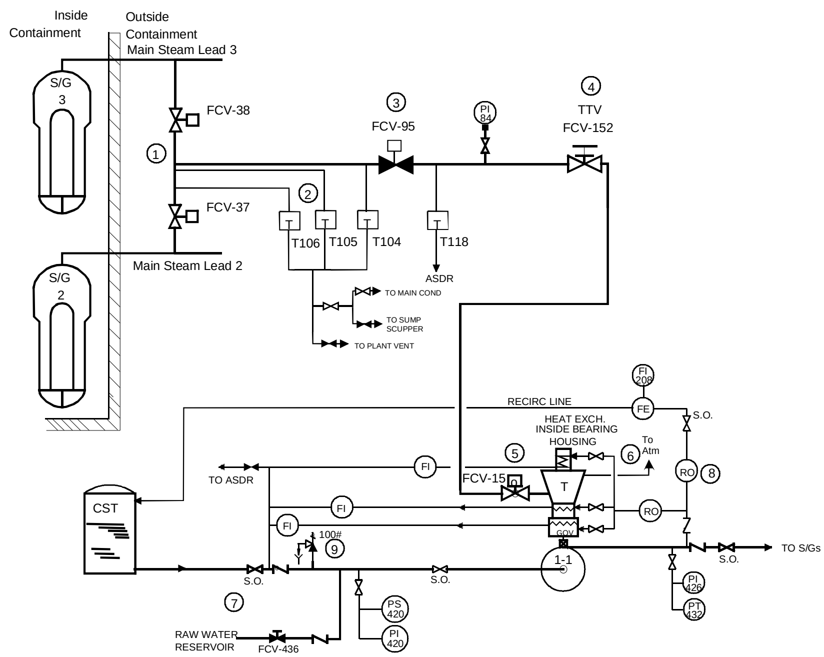
10 CFR Part 55 Content: 55.41 41.8
55.43 _____

Comments:
K/A: E03 G2.1.23 - LOCA Cooldown and Depressurization, Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Turbine Driven AFW Pump 1, Continued

Flowpath diagram
Obj 7, 13

The following is a diagram of the steam supply and exhaust flowpath for the TDAFW pump:



AFW-10

Obj 38

Stage	Description
1	Steam can be supplied by #2 and/or #3 S/G(s) through motor operated isolation valves FCV-37 and 38.
2	• Steam traps on the supply lines remove moisture from the system. • Traps 104, 105, and 106 must be operable to declare the pump operable.
3	The pump is started and stopped with steam supply flow control valve FCV-95.

Continued on next page

Turbine Steam Supply Control Valves (FCV-37/38), Continued

Controls

Obj 9, 10, 25

FCV-37 and 38 have control capability from the
- Control Room on VB-3 and
- local handwheel.

In the Control Room:

Control	Operation
CLOSE/STOP/OPEN	3 position, maintained

Indications

The following indications are available for FCV-37 and 38:

Obj 9, 10, 11, 23

FCV-37 and 38 control switches		
Indicating Lights	**Meaning**	**Normal Status**
Red	Valve is full OPEN.	ON
Green	Valve is full CLOSED.	OFF
NOTE: If BOTH Indicating lights are LIT simultaneously, then the valve is in an intermediate position.		

Alarms

The following alarms exist for FCV-37 and 38:

Obj 12, 23

Parameter	Source	Location
CLOSED valve position.	POS 425, 433	Control Room

TDAFW Pump Steam Supply Valve FCV-95

Purpose
Obj 13

The purpose of FCV-95 is to provide isolation of the steam supply to the TDAFW pump and maintain it in a standby condition.

Location
Obj 8

FCV-95 is located in the 115′ penetration area.

See figure on page 2-22

Power supplies

FCV-95 is powered from DC Pnl 12.

Description

FCV-95 is a DC powered MOV.
- It is designed to operate during a loss of all AC.

Controls

Obj 10, 25

- FCV-95 has control capability from the
  - ♦ Control Room on VB-3
  - ♦ HSD Pnl
  - ♦ local handwheel
- FCV-95 transfer relay has control capability from 480V Bus G.

In the Control Room:

Control for valve:	**Operation**
CLOSE/OPEN	3 position, spring return to neutral.

At the HSD Pnl:

Control for valve:	**Operation**
CLOSE/OPEN	3 position, spring return to neutral.
CONT RM/LOCAL	2 position, maintained.

At 480 V Bus G:

Control for transfer relay:	**Operation**
CUTIN/CUTOUT	2 position, maintained.
TRIP/RESET	2 position, maintained.

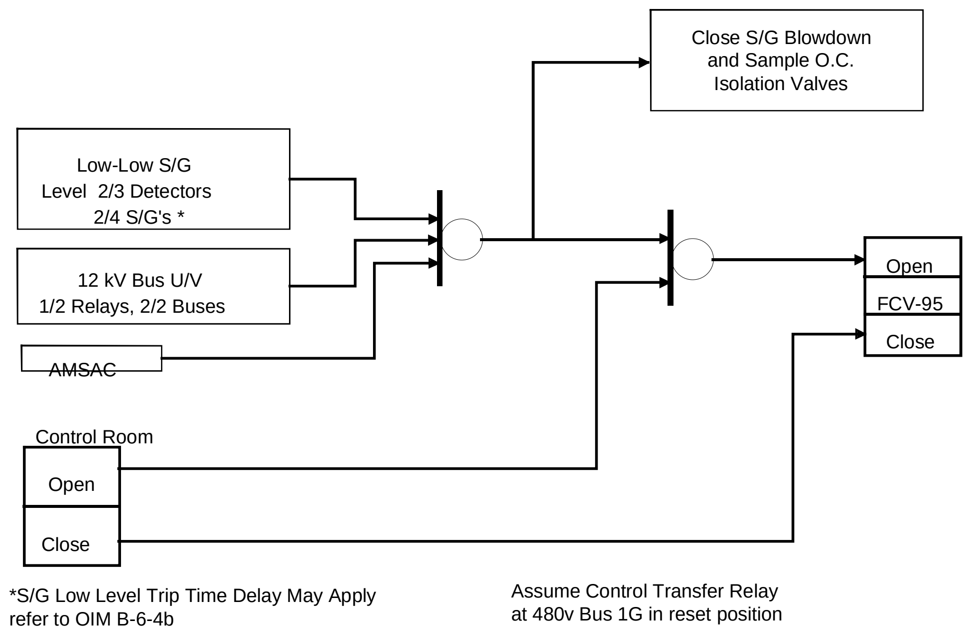
Continued on next page

TDAFW Pump Steam Supply Valve FCV-95, Continued

Logic (continued)

When controlling FCV-95 from the HSD Pnl,
- all automatic trips continue to function normally.
- all FCV-95 lights in the Control Room are extinguished.

Logic Diagram The following logic diagram assumes control from the Control Room:



AFW-06

Continued on next page

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. CHECK Intact S/G Levels:	
a. Check S/G NR Level - GREATER THAN 6% [16%]	a. Maintain TOTAL feedflow GREATER THAN 435 GPM until S/G NR Level is GREATER THAN 6% [16%] in at least one S/G.
b. Control feedflow to maintain S/G NR Level between 6% [16%] and 44%	b. IF NR Level in any intact S/G continues to increase in an uncontrolled manner, THEN GO TO EOP E-3, STEAM GENERATOR TUBE RUPTURE.
c. Check TD AFW Pp - REQUIRED FOR HEAT REMOVAL	c. WHEN NR Level in at least 3 S/Gs is GREATER THAN 16 % THEN Shutdown TD AFW Pp
d. Throttle TD AFW LCVs as necessary to maintain S/G Level.	

RO QUESTION 27

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	_____
Group #	2	_____
K/A #	W/E 16 G2.4.19	
Importance	2.7	3.7

Proposed Question:

The crew is about to exit E-0 and enter E-1.

The following containment conditions exist:

- Current containment pressure, 12 psig
- Peak containment pressure, 25 psig
- Containment Recirc Sump level (VB1), 94 feet
- Containment Wide Range Sump level (Pam 1), 91 feet
- Containment Radiation, RM-30 & 31, 8 R/hr

Assuming all other status trees are GREEN, which of the following actions should be taken as the crew exits E-0?

A. Do not enter E-1, go to FR-Z.1.

B. Do not enter E-1, go to FR-Z.2.

C. Enter E-1 or go to FR-Z.2 at the discretion of the SFM.

D. Enter E-1 or go to FR-Z.3 at the discretion of the SFM.

Proposed Answer:

D. Enter E-1 or go to FR-Z.3 at the discretion of the SFM.

Explanation:

A incorrect, the criteria for entry to Z.1 has cleared.
B incorrect, the criteria for Z.2 is WR sump level, not recirc sump level
C incorrect, Z.2 is a magenta path, which if satisfied would have to be entered.
D correct, Z.3 is a yellow path, the option to perform the procedure is up to the operator.

Technical Reference(s):
ro tier 1 group 2_27 rev1.doc

EOP F-0, page 2 and attachment 6 (F-0.5, Containment)

Proposed references to be provided to applicants during examination: F-0.5

Learning Objective: 3855 - Explain the priorities of the CSFSTs.
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 41.10
55.43 _____

Comments: K/A: W/E 16 G2.4.19 – High Containment Radiation, Knowledge of EOP layout, symbols and icons.

3.0 RULES OF USAGE

3.1 The Critical Safety Function Status Trees shall be monitored in the following order of priority:

3.1.1 Subcriticality using Status Tree F-0.1.

3.1.2 Core Cooling using Status Tree F-0.2.

3.1.3 Heat Sink using Status Tree F-0.3.

3.1.4 RCS Integrity using Status Tree F-0.4.

3.1.5 Containment using Status Tree F-0.5.

3.1.6 Inventory: Using Status Tree F-0.6.

3.2 IF an extreme challenge (RED PATH) is diagnosed, THEN the operator shall IMMEDIATELY stop procedure in effect and initiate functional restoration to restore the critical safety function under extreme challenge.

3.3 IF a severe challenge (MAGENTA PATH) is diagnosed, THEN the operator shall continue to check the status of all remaining critical safety functions. IF no extreme challenges exist, THEN the operator shall stop procedure in effect and initiate functional restoration to restore the highest priority critical safety function under severe challenge.

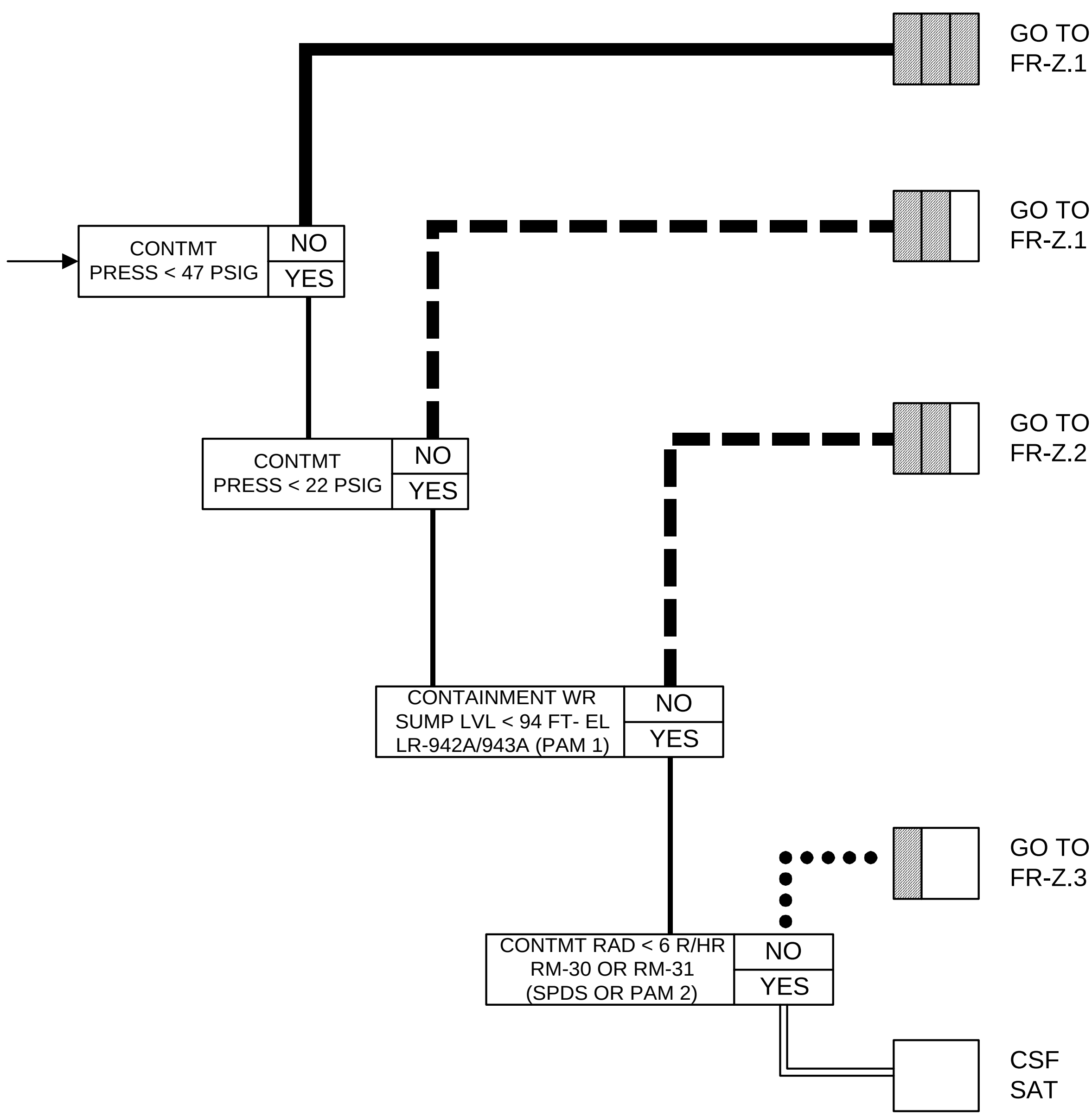
3.4 IF a not satisfied condition (YELLOW PATH) is diagnosed, THEN the operator shall continue to check the status of all remaining critical safety functions. IF no extreme or severe challenges exist, THEN it is the operator's option to continue optimal recovery procedures or to initiate functional restoration of the affected critical safety function challenge.

3.5 IF a satisfied condition (GREEN PATH) is diagnosed, THEN no challenge exists for the affected critical safety function. The operator shall continue to check the status of all remaining critical safety functions.

3.6 IF during function restoration to address a critical safety function challenge, a higher priority challenge is diagnosed, THEN the operator should terminate the ongoing response and initiate function restoration to address the higher priority critical safety function challenge.

3.7 IF an extreme challenge (RED PATH) or a severe challenge (MAGENTA PATH) is diagnosed AND subsequently clears before the status trees are being implemented or during the performance of a higher priority function restoration, THEN that challenge is to be considered satisfied.

F-0.5 CONTAINMENT



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RO QUESTION 28

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	003 K4.07	
	Importance	3.2	3.4

Proposed Question:

During the response to a loss of all A/C power, a maximum rate S/G depressurization is performed to 240 PSIG.

What is the benefit of performing the secondary depressurization?

A. It reduces RCS pressure and seal delta P and thus RCS leakrate.

B. It minimizes thermal stress across the steam generator tube sheet.

C. It prevents the formation of a void in the upper head of the reactor vessel.

D. It ensures that sufficient heat transfer capability exists to remove heat from the RCS.

Proposed Answer:

A. It reduces RCS pressure and seal delta P and thus RCS leakrate.

Explanation:

A correct, the step in ECA-0.0 depressurizes the intact SGs, thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss.
B incorrect, sufficient level is maintained in the S/Gs such that they are not challenged.
C incorrect, it is assumed a void may occur during the depressurization but the depressurization continues.
D incorrect, this is why level must be maintained in the S/Gs during the depressurization.

Technical Reference(s): Background ECA-0.0, Loss of All AC Power

Proposed references to be provided to applicants during examination: none

Learning Objective: 7920 Explain basis of emergency procedure step

Question Source: Bank # B-0144

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 41.3
55.43 _____

Comments:
K/A: 003 K4.07 - Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Minimizing RCS leakage (mechanical seals)

1	B-0144	Points:	1.00	Multiple Choice

During the response to a loss of all A/C power, a maximum rate S/G depressurization is performed to 240 PSIG.

How does this secondary depressurization minimize the RCS inventory loss?

A. It reduces RCS pressure and seal delta P and thus RCS leakrate.

B. It ensures that NO RCP seal failures will occur.

C. It will cause the RCP No. 1 seal to go from a film riding to face rubbing seal thus reducing RCS leakage.

D. It will cause the No. 1 seal to "uncock" and reseat properly.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7920	Explain basis of emergency procedure step

Reference Id: B-0144
Must appear: No
Status: Active
User Text: 7920.130494
User Number 1: 3.00
User Number 2: 3.40
Difficulty: 3.00
Time to complete: 3
Topic: EP ECA-0.0, Minimize RCS inventory loss by secondary depres
Cross Reference: ECA 0.0, STEP 17
Comment: 0000060501
REF: EP ECA-0.0, "Loss of All A/C" step #17 note, and BGD
VALIDATION DATE: 2/6/916

STEP: Depressurize Intact SGs To (O.08) PSIG

PURPOSE: To depressurize the intact steam generators

BASIS:

Step 16 depressurizes the intact SGs, thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss. The advantages to performing this action, as well as restrictions that apply during the action, are detailed in Subsection 2.3.

During SG depressurization, SG level must be maintained above the top of the SG U-tubes in at least one SG. Maintaining the U-tubes covered in at least one SG will ensure that sufficient heat transfer capability exists to remove heat from the RCS via either natural circulation or reflux boiling after the RCS saturates. Step 16a requires that SG level be in the narrow range in at least one SG before SG depressurization is initiated in Step 16b. If level is not in the narrow range in at least one SG, RNO 16a instructs the operator to maintain maximum AFW flow until narrow range level is established in one SG. When narrow range level is established, SG depressurization can be started or continued via Step 16b.

Step 16b instructs the operator to reduce SG pressures by depressurizing the intact SGs. Depressurization should be accomplished by opening the PORVs on the intact SGs to establish a maximum steam dump rate, consistent with plant specific constraints. The step is structured assuming that the operator can open and control SG PORVs from the control room. This structure assumes that the PORVs are air-operated and have dc control power and pneumatic power (i.e., either air reservoirs or nitrogen bottles) available. Some plants may not have the capability to open the SG PORVs from the control room. These plants should evaluate their capability to accomplish this step locally via PORV handwheels. Such an evaluation should consider accessibility and communications necessary to accomplish local PORV operation.

Once depressurization is initiated, maintenance of a specified rate is not critical. The depressurization rate should be sufficiently fast to expeditiously reduce SG pressures, but not so fast that SG pressures cannot be controlled. It is important that the depressurization not reduce SG pressures in an uncontrolled manner that undershoots the pressure limit, thus permitting potential introduction of nitrogen from the accumulators into the RCS.

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.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	004 K2.01	
	Importance	2.9	3.1

Proposed Question:

A loss of offsite power has occurred. Emergency diesel generators 1-1, 1-2 and 1-3 are running, and supplying their respective 4 KV bus.

Diesel 1-1 trips.

Which of the following describes the effect on the Reactor Makeup System Boric Acid pumps?

A. Boric Acid pump 1-1 has lost power.

B. Boric Acid pump 1-2 has lost power.

C. Both Boric Acid pump 1-1 and 1-2 have lost power.

D. No effect, both boric acid pumps 1-1 and 1-2 still have power.

Proposed Answer:

D. No effect, both boric acid pumps 1-1 and 1-2 still have power.

Explanation:
Unit 1 Bus F, G, H – EDG 1-3, 1-2, 1-1
A incorrect, BA pump 1-1 powered from Bus F (EDG 1-3)
B incorrect, For Unit 1, EDG 1-2 supplies bus G. Bus G powers Boric Acid pump 1-2.
C incorrect, one pump per bus prevents this.
D correct, pump 1-1 is powered from bus F (EDG 1-3) and 1-2 is powered from bus G (EDG 1-2).

Technical Reference(s):
B1B – Reactor Makeup System, page 2.2-10

Proposed references to be provided to applicants during examination: none

Learning Objective:
5681 - Explain the impact on boration flowpaths if a loss of 480V Bus G occurs.

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 41.6
55.43 _____

Comments:

K/A: 004 K2.01 – Knowledge of bus power supplies to Boric Acid Makeup Pumps

Boric Acid Transfer Pumps

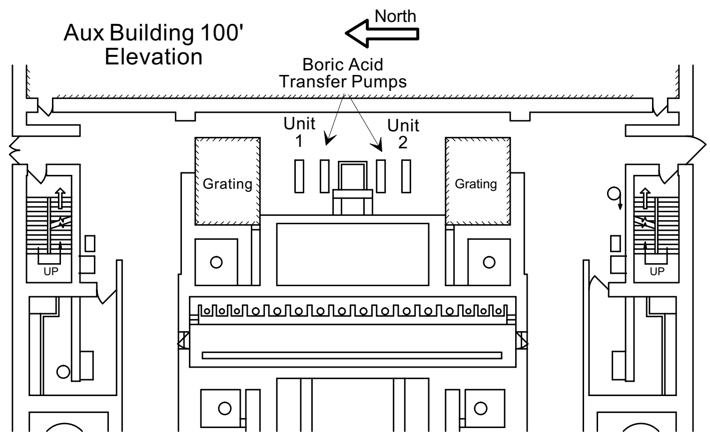
Purpose
Obj 5, 8

The purpose of the boric acid transfer pumps is to:

- Direct flow to the boric acid tanks from the boric acid batching tank.
- Direct flow from the boric acid tanks to the blender upon demand from the makeup system or to the charging pump suction for emergency boration.
- Recirculate fluid in the boric acid tank to prevent stratification.

Location
Obj 6

The pumps are located east of the filter gallery on the 100′ elevation of the Auxiliary Building.



RMU-09

Power supplies

The table below lists the power supplies to the pumps

BAT Pump	480 VAC vital bus
1-1	F
1-2	G
2-1	F
2-2	G

Continued on next page

RO QUESTION 30

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	004 A4.17	
	Importance	2.7	2.7

Proposed Question:

CVCS Cation Demin 1-1 is to be placed in service for pH control using OP B-1A:XIII, CVCS Demineralizers. The bed was last used 2 days ago.

Which of the following is verified to prevent a possible reactivity excursion?

A. Letdown temperature is normal.

B. Effluent samples of the demineralizer have been taken to verify effects on RCS chemistry.

C. No deborating demineralizer is in service.

D. Automatic makeup to the VCT is in service at the current boron concentration.

Proposed Answer:

C. No deborating demineralizer is in service.

Explanation: per OP B-1A:XIII, attachment 9.3 Place CVCS Cation Demin 1-1 in Service for RCS Lithium Removal (pH control)

A incorrect, abnormal letdown temperature can cause a reactivity excursion if flow is through a deborating demineralizer.
B incorrect, this applies to deborating demins.
C correct, Caution: Placing Cation demin 1-1 in service while the Deborating demin is in service could result in a reactivity excursion. Switch in normal and bypass light lit indicates no deborating demin in service.
D incorrect, this is a caution but not to prevent a reactivity excursion, this is to ensure makeup to the VCT available if a rinse is performed and letdown is diverted.

Technical Reference(s): OP B-1A:XIII

Proposed references to be provided to applicants during examination: None

Learning Objective: 76912 – Explain the precautions taken in the control room when placing CVCS demineralizers in service.

Question Source: Bank #
Modified Bank # _______
New X

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 41.6
55.43 _____

Comments: K/A: 004 A4.17 - Ability to manually operate and/or monitor in the control room: Deborating demineralizer

DIABLO CANYON POWER PLANT
OP B-1A:XIII
ATTACHMENT 9.3

1

TITLE: Place Cation Demin 1-1 In Service for RCS Lithium Removal (pH Control)

Duration Cation Demin 1-1 is to be in service per chemistry guidance: ________________

****
CAUTION: Placing Cation Demin 1-1 in service while the Deborating demin is in service could result in a reactivity excursion. (Ref AR A0581707)
****

SFM authorizes the use of this procedure attachment: SFM Signature: ________________ DATE: ________

1. Verify that neither Deborating demin is currently in service.
2. Check if Cation Demin 1-1, which is to be placed in service, has been in service at least once in the previous seven (7) days by reviewing the date and time on the Abnormal Status Board.

NOTE: If a date is not written on the Abnormal Status Board and cannot be verified by some other means (i.e., a log entry), it must be assumed that the Cation Demin has NOT been used within the last seven (7) days, and the rinse to the LHUT aligned for RCS letdown must be performed as specified below.

****
CAUTION: PRIOR to diverting letdown to an LHUT, ensure automatic makeup to the VCT is in service.
****

3. Notify the control operator and shift chemistry technician that Cation Demin 1-1 is now being placed in-service.
4. Align the demin as follows:
   a. If Cation Demin 1-1 has NOT been in-service within the last seven (7) days, align for a demin rinse to an LHUT as follows:
      1) Verify that automatic makeup to the VCT is in service.
      2) Notify the control operator to select "DIVERT" on CVCS-1-LCV-112A to align demin effluent to an LHUT.
   b. Verify open CVCS-1-8535, Cation Demin 1-1 Inlet.
   c. Verify open CVCS-1-8516, 1-1 Cation Demin Inlet.
   d. Open CVCS-1-8518, 1-1 Cation Demin Outlet.
   e. Close CVCS-1-8514, 1-1 Cation Demin Bypass.
   f. If rinsing demin to an LHUT, select "AUTO" on CVCS-1-LCV-112A after rinsing for a minimum of 10 minutes (based on 75 gpm letdown).

TITLE: Place Cation Demin 1-1 In Service for RCS Lithium Removal (pH Control)

5. Monitor the in-service RCS Letdown Filter D/P while Cation Demin 1-1 is in service.
6. Record date/time Cation Demin 1-1 placed in-service: __________/_________
7. When the required length of time has elapsed, perform the following:
    a. Open CVCS-1-8514, Cation Demin 1-1 Bypass.
    b. Close CVCS-1-8518, 1-1 Cation Demin Outlet.
8. Record date/time Cation Demin 1-1 removed from service: __________/_________
9. Notify the control operator and shift chemistry technician that Cation Demin 1-1 has been removed from service.
10. Update the CO's Abnormal Status Board to reflect the last in-service date/time of Cation Demin 1-1.
11. Return completed attachment to the chemistry foreman.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	005 K6.03	
	Importance	2.5	2.6

Proposed Question:

10 hours ago a design basis LOCA occurred on Unit 1. All equipment is running as designed and aligned per the emergency procedures.

CCW Cooling is lost to RHR heat exchanger 1-1.

As a result cooling is lost to RHR flow going to which of the following?

A. The suction of the safety injection pumps and hot legs 1 and 2.

B. The suction of the safety injection pumps and cold legs 1 and 2.

C. The suction of the charging pumps and hot legs 1 and 2.

D. The suction of the charging pumps and cold legs 1 and 2.

Proposed Answer:

D. The suction of the charging pumps and cold legs 1 and 2.

Explanation:
A incorrect, hot leg recirc is not established until 10.5 hours and SI pumps are supplied by HX 1-2.
B incorrect, SI pumps are supplied by HX 1-2
C incorrect, the unit is still in cold leg recirc
D correct, the charging pumps are supplied by HX 1-1 and RHR HX 1-1 cools water being injected into cold legs 1 and 2.

Technical Reference(s):
E-1, Loss of Reactor or Secondary Coolant, step 20
E-1.4, Transfer to Hot Leg Recirculation, steps 1 and 2
B2, Residual Heat Removal, page 3-18

Proposed references to be provided to applicants during examination: none

Learning Objective: 7012 - Explain the operation of RHR system valves.

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 41.3
55.43 _____

Comments:
K/A: 005 K6.03 - Knowledge of the effect of a loss or malfunction of the following will have on the RHRS: RHR heat exchanger

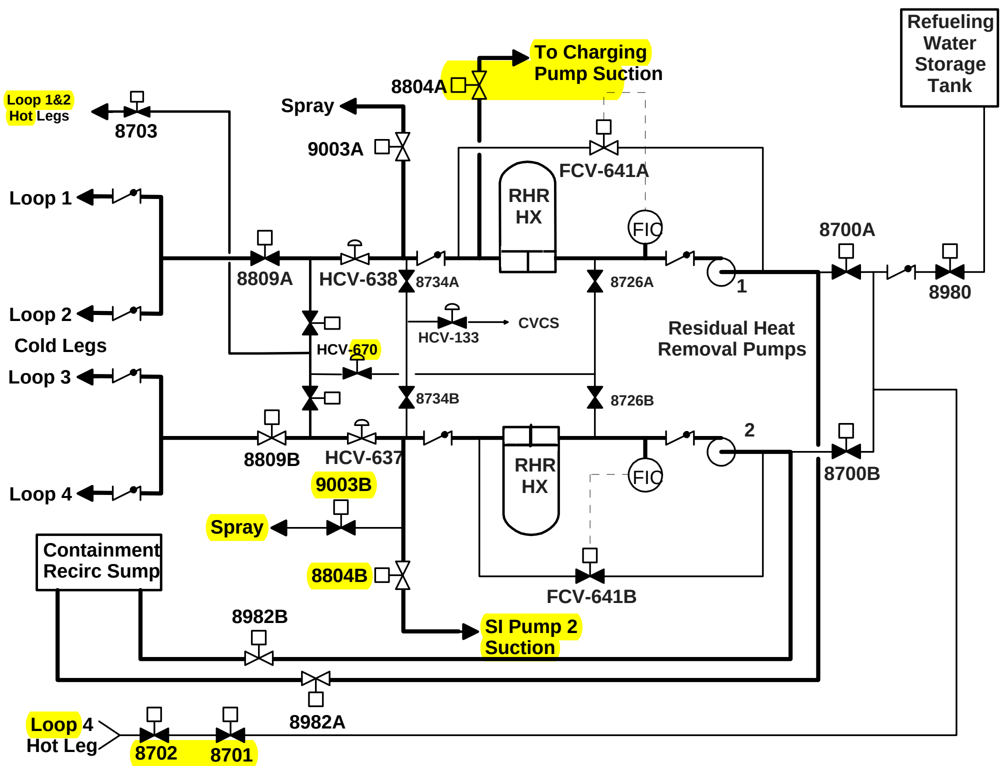
Emergency Operations, Continued

Cold Leg Recirculation Alignment
Obj 19

When the RWST level reaches [33%], the RHR pumps will trip. At this point, the RHR system is realigned to take a suction from the Containment sump and discharge to:

- RCS cold legs
- Charging pump suction
- SI Pump suction
- Containment Spray headers

The diagram below shows the valve alignment.



RHR-09

Continued on next page

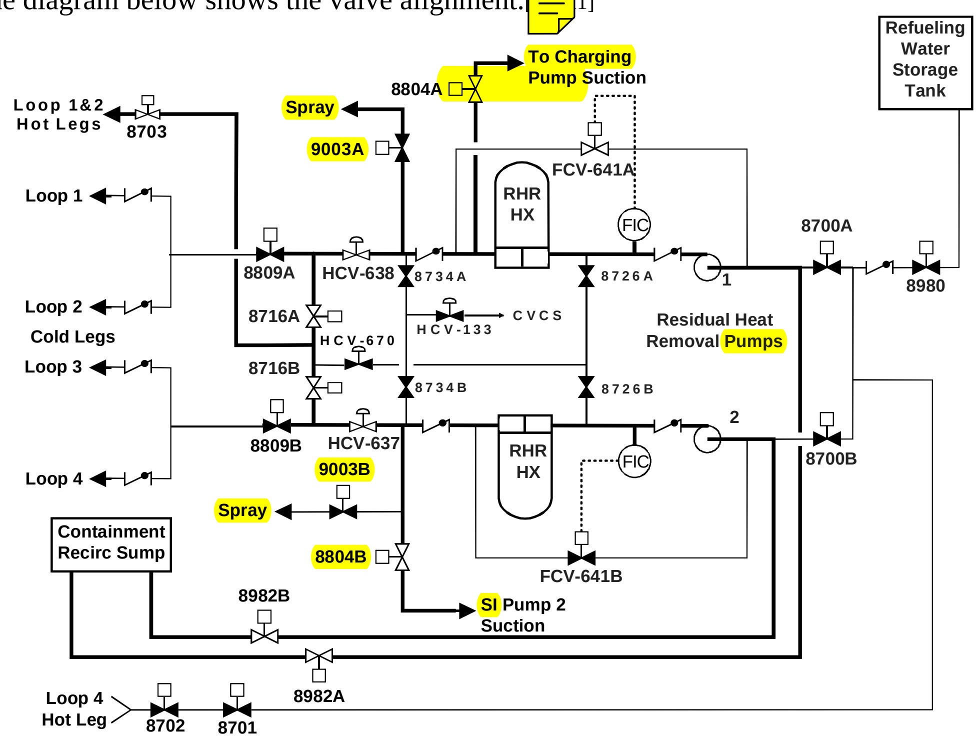
Emergency Operations, Continued

Hot Leg Recirculation Alignment
Obj 19

Reason for Hot Leg Recirculation:

- Cold leg recirculation is performed for 10-1/2 hours.
  - ♦ ECCS fluid enters the bottom of the reactor core,
  - ♦ flows upward over the core,
  - ♦ absorbs decay and sensible heat,
  - ♦ leaves the reactor vessel.
- Initially, the ECCS fluid may be boiling, leaving boric acid behind as the vapor is swept away.
  - ♦ If the boric acid is permitted to build up on the fuel rods, increased fuel temperatures could result.
  - ♦ Also, since boric acid solubility is a function of fluid temperature, boric acid may start to precipitate near the bottom of the core (the coldest portion of the core).
  - ♦ Excessive precipitation could result in blocked flow passages, which could reduce the emergency core cooling flow to the core.
- When the ECCS is shifted to hot leg recirculation;
  - ♦ the fluid enters the top of the reactor core,
  - ♦ flows downward over the core,
  - ♦ absorbs decay and sensible heat,
  - ♦ washes plated-out boric acid off the fuel rods,
  - ♦ washes the concentrated boric acid out of the lower part of the core,
  - ♦ prevents flow restriction and heat transfer problems.

The diagram below shows the valve alignment.[1]



RHR-10

Continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18. **DETERMINE If Reactor Vessel Head Should Be Vented:**	
a. Consult TSC for venting approval	a. GO TO Step 19.
b. REFER TO EOP FR-I.3, RESPONSE TO VOIDS IN REACTOR VESSEL	
19. **CHECK Containment Hydrogen Concentration:** (PAM 1)	
a. Check H_2 analyzer - IN SERVICE (PAM 1)	a. Contact the Chemistry Dept to place the Hydrogen Monitoring System in service.

****

CAUTION: Operation of the Containment Hydrogen Recombiner with Containment Hydrogen Concentrations GREATER THAN 4.0% could result in ignition of the hydrogen.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Hydrogen concentration - LESS THAN 3.5%	b. Consult Plant Engineering Staff (TSC) when manned for additional recovery actions with potential explosive H_2 mixture in containment. GO TO Step 20.
c. Hydrogen concentration - LESS THAN 0.5%	c. IMPLEMENT OP H-9, INSIDE CONTAINMENT H_2 RECOMBINATION SYSTEM, to reduce Hydrogen Concentration.
20. **At 10 hours After Event Initiation Prepare For Hot Leg Recirculation:**	
GO TO EOP E-1.4, TRANSFER TO HOT LEG RECIRCULATION	

- END -

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**

NOTE: It is important during this phase that two separate and redundant trains of recirculation outside containment are established unless an inoperable 4 KV vital bus prevents total separation.

1. **PREPARE For Hot Leg Recirculation 10 hours After Event Initiation:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check the following control switches in their required position:	a. Place the Valve Control Switches in the required position.
o 8802A - CLOSED, SI to Hot Legs 1 & 2	- -
o 8835 - OPEN, SI Pp to Cold Leg	
o 8703 - CLOSED, RHR to Hot Legs 1 & 2	
o 8802B - CLOSED, SI to Hot Legs 3 & 4	
b. Close the following 480V breakers:	
o 52-1F-48, 8802A	
o 52-1G-24, 8835	
o 52-1G-56, 8703	
o 52-1G-56R, 8703	
o 52-1H-26, 8802B	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. At 10.5 hours ALIGN SI Pp 1 For Hot Leg Recirculation:	
a. Verify both RHR Pps are running	a. Manually Start any RHR Pp NOT running. IF RHR Pp 1 is NOT Operable, THEN Close 8804A AND GO TO Step 2f. IF RHR Pp 2 is NOT Operable, THEN Continue with Step 2b. -
b. Verify 8804A, RHR Hx No. 1 to Chg and SI Pps Suction - OPEN	
c. Cutin 8809A series contactor toggle switch	
d. Close 8809A, RHR to Cold Legs 1 and 2	
e. Verify Closed 9003A, RHR Pp 1 to Spray Hdr A - CLOSED	
f. Verify SI Pp 1 - STOPPED	
g. Close 8821A, SI Pp No. 1 Disch Crosstie Vlv	
h. Open 8802A, SI to Hot Legs 1 and 2	
i. Perform the following 1) Start SI Pp 1 2) Verify operating RHR Pp motor current LESS THAN 57 AMPS	i. IF SI Pp 1 is NOT Operable, THEN GO TO Step 2k (Next Page). -

THIS STEP CONTINUED ON NEXT PAGE

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	006 K1.02	
	Importance	4.3	4.6

Proposed Question:

Some time after an accident, containment pressure indicates the following on VB1:
- PI-934 = 20 psig
- PI-935 = 23 psig
- PI-936 = 20 psig
- PI-937 = 24 psig

Containment spray does not actuate.

Which of the following would explain why spray did not actuate?

A. Safety Injection signal has been reset.

B. RWST is below the low level alarm setpoint.

C. Containment Isolation Phase A has failed to actuate.

D. Not enough channels of containment pressure are at/above the nominal setpoint.

Proposed Answer:

A. Safety Injection signal has been reset.

Explanation:
A correct, an SI signal must be present
B incorrect, RWST level would not inter with actuation.
C incorrect, no interface.
D incorrect, setpoint is 2/4 at 22 psig

Technical Reference(s):
Drawings 4014233, SSPS Functional Diagram and 498006 Containment Pressure
STG I2, Containment Spray System, page 2-9

Proposed references to be provided to applicants during examination: None

Learning Objective: 5422 Explain the consequences of SSPS failures on plant operation

Question Source: Bank #
Modified Bank # P-49415
New ______

Question History: Last NRC Exam DCPP RO Exam 1995

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

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: K/A: 006 K1.02 - Knowledge of the physical connections and/or cause effect relationships between the ECCS and the following: ESFAS

1 P-49415 Points: 1.00 Multiple Choice

An operator attempts to initiate containment spray manually by actuating the switches "Containment Isol Phase B/ Containment Spray Train A" and "Containment Isol Phase B/ Containment Spray Train B" on VB-1. The spray fails to actuate. WHICH ONE of the following is a possible cause of the failure of spray to initiate?

A. A Safety Injection signal was NOT present.

B. The two switches were operated simultaneously.

C. The Spray Additive Tank was below the low level alarm setpoint.

D. Containment Isolation Phase B failed to actuate.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

5422	Explain the consequences of SSPS failures on plant operation

Reference Id: P-49415
Must appear: No
Status: Active
User Text: 013K1.01
User Number 1: 4.20
User Number 2: 4.40
Difficulty: 3.00
Time to complete: 2
Topic: 23 ESFAS: Containment Spray Actuation requires SI signal
Cross Reference: STG B6A,REV 8,PG 2.2-25
Comment: History: DCPP RO Exam 1995

Containment Spray Pumps, Continued

Logic
Obj 10, 27

The logic associated with the Containment Spray Pump is described in the following table.

If the VB-1 pump switch is in ...	Then the pump breaker will ...
STOP	Open
NEUTRAL	Close on: • Safety Injection Actuation AND either: • HI HI Containment Pressure signal OR • MANUAL spray actuation Open on: • Overcurrent • Bus transfer to diesel (load shed).
START	Close

Continued on next page

5
6
7
8
9
10
TRAIN A
TRAIN B
LO STM LINE
PRESSURE S.I.
BLOCKED
ALTERNATE
TRAIN
PC
937B
936B
935B
934B
CONTAINMENT
HI-HI PRESSURE
2/4
HI-HI
SAFEGUARDS
ACTUATION
COORD. 5-E
DWG 4014233
CLOSE
MAIN STEAM
STOP VALVES
STEAMLINE ISOLATION
936A
935A
934A
CONTAINMENT
HIGH PRESSURE
2/3
HI
SAFEGUARDS
ACTUATION
COORD. 5-D
DWG 4014233
CONTAINMENT PRESSURE
DWG 498024
HIGH RATE
MAIN STEAM
ISOLATION
HIGH STEAM PRESSURE RATE
(LAG COMPENSATED), DWG 498020
E
D
C


5
6
7
8
9
10
CONTAINMENT PURGE EXHAUST DETECTOR
CONTAINMENT VENTILATION ISOLATION
CONTAINMENT PRESSURE COORD 10-E, DWG 498006
HI-HI
MAN. CONTMT. ISOL PHASE B AND SPRAY ACTUATION
CONTAINMENT ISOLATION PHASE A
CONTAINMENT ISOLATION PHASE B
MAN SAF INJ ACTUATION
FEEDWATER CONTROL COORD 3-C, DWG 498006
OPEN SPRAY NaOH VALVES
SPRAY PUMPS START AND DISCH VALVES OPEN
CONTAINMENT PRESSURE COORD 10-D, DWG 498006
HI
SAF. INJ.
LOW STEAM LINE PRESSURE COORD 10-B, DWG 498006
AUTO S.I. BLOCKED
P4 REACTOR TRIP
RT
MAN SAF INJ RESET
AND BLOCK
1.5-5 SEC
1-2 MIN
SAFEGUARDS ACTUATION DWG 498025
SAFETY INJECTION
MANUAL CONTAINMENT ISOLATION PHASE A
E
D
C

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	006 K6.03	
	Importance	3.6	3.9

Proposed Question:

Which of the following ECCS equipment out of service configurations would result in entry into Technical Specification 3.0.3?

A. Train A CCP and Train B RHR pump.

B. Train A SI pump and Train B SI pump.

C. Train A CCP and Train B SI pump.

D. Train A CCP and Train A SI pump.

Proposed Answer:
B. Train A SI pump and Train B SI pump.

Explanation:
A incorrect, still have 100% flow equivalent to a single train.
B correct, per 3.5.2 bases: Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. (i.e. minimum of one OPERABLE CCP, SI, and RHR pump and applicable flow paths capable of drawing from the RWST and injecting into the RCS cold legs). This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable….
With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.
C incorrect, still have 100% flow equivalent to a single train.
D correct, still have 100% flow equivalent to a single train.

Technical Reference(s): Tech Spec Bases 3.5.2

Proposed references to be provided to applicants during examination: Tech Spec 3.5.2

Learning Objective: 9694E - Discuss 3.5 Technical Specification bases

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 41.10

55.43 43.5

Comments:

Resampled KA due to inability to test original KA. No effective tie between Loss of annunciators and ECCS (006 G2.4.32)

K/A: 006 K6.03, ECCS - Knowledge of the effect of a loss or malfunction of the Safety Injection Pumps will have on the ECCS

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

---NOTE---
In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valve(s) for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status	---------NOTE------- The Completion Time may be extended to 7 days for Unit 1 cycle 12 for centrifugal charging pump 1-1 seal replacement ------------------------ 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. <u>Number</u> / <u>Position</u> / <u>Function</u> 8703 / Closed / RHR to RCS Hot Legs 8802A / Closed / Safety Injection to RCS Hot Legs 8802B / Closed / Safety Injection to RCS Hot Legs 8809A / Open / RHR to RCS Cold Legs 8809B / Open / RHR to RCS Cold Legs 8835 / Open / Safety Injection to RCS Cold Legs 8974A / Open / Safety Injection Pump Recirc. to RWST 8974B / Open / Safety Injection Pump Recirc. to RWST 8976 / Open / RWST to Safety Injection Pumps 8980 / Open / RWST to RHR Pumps 8982A / Closed / Containment Sump to RHR Pumps 8982B / Closed / Containment Sump to RHR Pumps 8992 / Open / Spray Additive Tank to Eductor 8701 / Closed / RHR Suction 8702 / Closed / RHR Suction	12 hours
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.3	Verify ECCS piping is full of water.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program.
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position. Charging Injection Throttle Valves: 8810A, 8810B, 8810C, 8810D Safety Injection Throttle Valves: 8822A, 8822B, 8822C, 8822D	24 months
SR 3.5.2.8	Verify, by visual inspection, each ECCS train containment recirculation sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	24 months

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

APPLICABILITY (continued)

requirements are addressed by 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."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available (capable of injection into the RCS, if actuated), the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their safety function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. (i.e. minimum of one OPERABLE CCP, SI, and RHR pump and applicable flow paths capable of drawing from the RWST and injecting into the RCS cold legs). This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

The intent of this Condition, to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available, applies to both the injection mode and the recirculation mode.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR cross-tie valve can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a

(continued)

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	007 A2.03	
	Importance	3.6	3.9

Proposed Question:

The plant is at full power. PRT parameters are all at normal steady state values.

The following events occur:

- The turbine trips
- The reactor fails to automatically trip but is manually tripped.
- All other systems operate as expected.
- The Emergency procedures have been performed
- The plant is stable at no load Tave and NOP
- It is noted that on the transient RCS pressure peaked at 2370 psig.
- Containment parameters are normal
- PK-05-25, PZR Relief Tank Press, Lvl and Temp is in alarm
- PRT Temperature - 140ºF,
- PRT Level - 85%
- PRT Pressure - 12 psig

Which of the following actions should be taken by the crew to address the current PRT conditions?

A. Reduce temperature by opening PRT primary water supply RCS-8030.

B. Restore level to normal by opening PRT primary water supply RCS-8030.

C. Investigate a possible failure of the N2 supply regulator.

D. Check closed PRT primary water supply RCS-8030 and drain the PRT to the RCDT.

Proposed Answer:
A. Reduce temperature by opening PRT primary water supply RCS-8030.

Explanation:
A correct, PRT temperature is high and is reduced by opening RCS-8030.
B incorrect, level is not low (56%)
C incorrect, this action is taken if temperature and level are normal.
D incorrect, this action is done if PRT level is high (89%)

Technical Reference(s): PK-05-25.

Proposed references to be provided to applicants during examination: PK05-25

Learning Objective: 4946, 93963 - State the normal readings of PRT system indications.
4946 - State the PRT parameters that produce alarms.

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 41.10
55.43

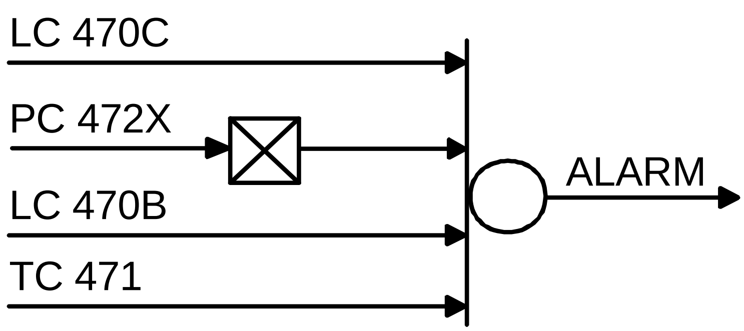
Comments:
K/A: 007 A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overpressurization of the PZR

TITLE: PZR RELIEF TANK PRESS, LVL, TEMP

01/31/03
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
LC 470A	367	PZR Relief Tk Lvl Hi	GT 89%
LC 470B	1394	PZR Relief Tk Lvl Lo	LT 56%
TC 471	318	PZR Relief Tk Temp Hi	GT 130°F
PC 472X	545	PZR Relief Tk Press Hi and Vent Hdr Isol	GT 10 Psig

3. PROBABLE CAUSE

3.1 Pressurizer power relief valves PCV-474, 455C, and/or 456 leaking through or lifting.

3.2 Pressurizer safety valves 8010A, 8010B, or 8010C leaking through or lifting.

3.3 Malfunction of pressurizer pressure control system.

3.4 Relief valve (from outside containment) lifting or leaking through:

3.4.1	Containment Spray Hdr. 9007A	260 psig
3.4.2	SIS Pump Suct. Hdr. 8858	220 psig
3.4.3	RHR Ht. Exch. 1-2 8856B	600 psig
3.4.4	Charging Pps Suct. Hdr. 8125	220 psig
3.4.5	SIS Pp. 1-2 Disch. Hdr. 8853B	1750 psig
3.4.6	SIS Pp. 1-1 Disch. Hdr. 8853A	1750 psig
3.4.7	SIS Pps. Disch. Hdr. 8851	1750 psig
3.4.8	RHR Ht. Exchg. 1-1 8856A	600 psig
3.4.9	Contmt. Spray Hdr 9007B	260 psig

3.5 Relief Valve (from inside containment) lifting or leaking through:

3.5.1	Seal Wtr. Ret. Hdr. 8121	150 psig
3.5.2	CVCS Letdown Hdr. 8117	600 psig
3.5.3	RHR Pp. Suct. Hdr. 8707	450 psig
3.5.4	RHR Pp. Inj. Loops 1 & 2	600 psig
3.5.5	RHR to RCS Hot Legs Hdr. 8708	600 psig

3.6 Valve Stem Leak-off:

3.6.1 8000 A, B, C, - PZR PORV ISO

3.6.2 PCV-455 A, B - PZR SPRAY

3.6.3 8701 and 8702 - LOOP 4 to RHR SYSTEM

3.6.4 8033 A, B, C, D - PZR SPRAY ISO VLVS

3.6.5 8076 - CVCS LTDN ISO VLV

3.6.6 8143 - EXCESS LTDN DIVERT VLV

3.6.7 HCV-123 - EXCESS LTDN FLOW CONTROL

3.7 Pressurizer safety valve loop seal drain header valve(s) leaking through.

3.8 Primary water supply valve 8030 leaking through (high PRT level).

3.9 PRT to RCDT 8031 leaking through (low PRT level) or leak in PRT or level tap below the water line.

3.10 Malfunction of N_2 supply PCV 8035.

4. AUTOMATIC ACTIONS

4.1 Vent header isolation at 10 psig (PCV-472 closed).

5. OPERATOR ACTIONS

5.1 Check annunciator typewriter printout and PPC computer alarm typewriter printout or CRT.

5.2 Check PRT level, pressure and temperature.

5.3 Monitor PORV Discharge HEADER temperature (TI-463) and Safety Valves Discharge Temperatures (TI-465, 467, or 469).

5.4 Check if PK05-08 is in alarm.

5.5 Monitor Sonic Flow Indicator, PDI-116A, 117A and 118A.

5.6 If source of discharge is a safety valve or PORV, refer to TS 3.4.11 or TS 3.4.13.

5.7 Refer to OP AP-1 Excessive RCS Leakage.

5.8 Determine and identify source of leakage. Refer to STP R-10 if PK05-08 is in alarm. Make preparations for entering containment to check valve stem leak-off indicating light panel.

5.9 High Level Alarm on PRT

5.9.1 CLOSE or check closed RCS-8030.

****
CAUTION: Overfill of the RCDT (>100% indicated) will cause filling of the LI reference legs and will require draining by MS. Due to the PRT drain RCS-1-8031 being a slow acting valve, it must be closed early to prevent overfill of the RCDT.
****

5.9.2 Drain the PRT to the RCDT.

a. Run both RCDT pumps in manual to keep up with the drain rate.

b. Maintain communication with the Aux. Senior during draining.

c. Open RCS-1-8031 and drain to approximately 84% level, closing RCS-1-8031 as necessary when RCDT level reaches 60% on LI-188 to prevent overfilling.

5.10 High Temp Alarm on PRT

5.10.1 OPEN PRT primary water supply RCS-8030.

5.10.2 Reduce PRT temperature to 120°F <u>THEN</u> CLOSE RCS-8030.

****

<u>CAUTION</u>: Overfill of the RCDT (>100% indicated) will cause filling of the LI reference legs and will require draining by MS. Due to the PRT drain RCS-1-8031 being a slow acting valve, it must be closed early to prevent overfill of the RCDT.

****

5.10.3 Drain the PRT to the RCDT.

a. Run both RCDT pumps in manual to keep up with the drain rate.

b. Maintain communication with the Aux. Senior during draining.

c. Open RCS-1-8031 and drain to approximately 84% level, closing RCS-1-8031 as necessary when RCDT level reaches 60% on LI-188 to prevent overfilling.

5.11 Low Level Alarm on PRT

5.11.1 OPEN PRT primary water supply RCS-8030 to increase level to 84%.

5.11.2 Check RCS-8031 closed. If necessary, cycle RCS-8031. If level continues to drop, check PRT and instrument taps for leaks.

5.12 High Pressure Alarm on PRT

5.12.1 If pressure has increased without increased temp and or level, <u>THEN</u> the N_2 supply regulator may have failed. If this is the case, close 8045, and either lower PRT level or remove blank flange from manual vent valve 8048 and bleed pressure off slowly to less than 10 PSIG at which time PCV-472 should be available.

5.12.2 If pressure has increased along with temp and/or level, follow appropriate steps as addressed in Step 5.7 and 5.8 above.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	008 K3.03	
	Importance	4.1	4.2

Proposed Question:

PLANT CONDITIONS:
- Unit 1 is at full power
- FCV-749 (RCP Lube Oil cooler CCW Return Valve) indicates fully closed
- Highest Motor bearing temperature is 180ºF, increasing at 2ºF/min
- Highest Stator winding temperature is 240ºF, increasing at 5ºF/min

What is the maximum amount of time the plant can remain at power?

A. 5 minutes

B. 10 minutes.

C. 12 minutes.

D. 22.5 minutes.

Proposed Answer:

A. 5 minutes.

Explanation:

A correct, per the foldout page of AP-28, for a loss of to the RCP motor coolers, the trip criteria is 5 minutes.
B incorrect, Foldout page, RCP have exceeded the limit of 200°F in 10 minutes at current rate, but must be tripped at 5.
C incorrect, this is the time until winding temperature reaches 300ºF
D incorrect, this is the time at 2ºF /min, 225ºF is reached (limit on RCP radial bearing), limit is not applicable, but candidate may remember it as a limit.

Technical Reference(s): OP AP-28

Proposed references to be provided to applicants during examination: none

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

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 _____

Comments: K/A: 008 K3.03 - Knowledge of the effect that a loss or malfunction of the CWS will have on the following: RCP

IF Any of the following criteria is met

THEN Trip the reactor **AND** trip the affected RCP

AND

Go To EOP E-0 **WHILE** completing foldout actions if applicable.

1.0

No. 1 SEAL REACTOR TRIP CRITERIA	
TRIP CRITERIA	**FOLDOUT ACTIONS**
1. RCP Seal #1 Return Flow GREATER THAN 6.0 GPM **AND** Pump Bearing **OR** Seal #1 Return Temps are INCREASING	**GO TO** step 1 of section 4 (pg 10)
2. RCP Seal #1 Return Flow is GREATER THAN 8.0 GPM	**GO TO** step 2 of section 4 (pg 11)
3. RCP Seal #1 Return Flow LESS THAN 0.8 GPM **AND** Pump Brg **OR** Seal return Temps are INCREASING	**GO TO** step 1 of section 4 (pg 10)

2.0

RCP MOTOR REACTOR TRIP CRITERIA
1. Any Motor Bearing Temperature GREATER THAN 200°F
2. Motor Stator Winding Temperature GREATER THAN 300°F
3. CCW Temperature GREATER THAN 120°F
4. Loss of CCW to RCP Motor Coolers for GREATER THAN 5 Minutes

3.0

RCP PUMP REACTOR TRIP CRITERIA
1. RCP Radial Bearing Outlet Temperature GREATER THAN 225°F
2. RCP #1 Seal Outlet Temperature GREATER THAN 225°F
3. Loss of RCP Seal Flow AND Loss of Thermal Barrier Cooling

4.0

RCP VIBRATION REACTOR TRIP CRITERIA
1. Any Valid RCP Seal flow **OR** Motor/Pump bearing temperature alarm concurrent with a vibration alarm.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	008 G2.2.13	
	Importance	3.6	3.8

Proposed Question:

While at power, CCW pump 1-1 was cleared for maintenance. As part of the clearance, several normally sealed open valves listed in OP K-10E1, Sealed Valve Checklist for Component Cooling Water Pump 1-1, are closed.

During restoration what type of seal should be reapplied to the valves?

A. YELLOW or GREEN seals.

B. YELLOW seals.

C. GREEN or RED seals.

D. RED seals.

Proposed Answer:

D. RED seals.

Explanation:

A incorrect, Yellow is flow balancing and Green indicates a valve sealed out of position.
B incorrect, YELLOW seal identifies a flow balancing throttle valve.
C incorrect, A GREEN seal is used to identify a valve sealed out of position.
D correct, Valves listed in the 10K series are considered Category I valves and as such should be sealed with a RED seal.

Technical Reference(s):
OP1.DC20, Sealed Components

Proposed references to be provided to applicants during examination: None

Learning Objective: 3598, Explain the characteristics of the Sealed Valve Program

Question Source: Bank #
Modified Bank # ________
New X

Question History: Last NRC Exam ______________
Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

X
10 CFR Part 55 Content: 55.41 41.10
55.43 _____

Comments:

K/A: 008 G2.2.13 – CCW, Knowledge of tagging and clearance procedures

2.2 Categories of Sealed Components

2.2.1 In general, when it is desired to assure non-operation of a component in an operational system, a CAUTION tag is normally used as directed by OP2.ID2. For components that are important to the proper functioning of a safeguards system, a seal shall be used rather than a CAUTION tag to assure non-operation.

2.2.2 The first category of sealed components consists of the plant components which are sealed to directly or indirectly satisfy Technical Specification requirements or to maintain reactor safety. The majority of these components are sealed to satisfy surveillance requirements where manual flowpath valves, transfer switch handles or breakers must be "locked, sealed, or otherwise secured in position." This category of sealed components is procedurally controlled by the K-10 series of operating procedures. For the purposes of this administrative procedure, all sealed components identified in the K-10 series of the plant operating procedure manual shall be considered CATEGORY 1 SEALED COMPONENTS.

2.2.3 A second broad category of sealed components consists of those components which have been selected to be sealed in position for reasons other than those discussed in Step 2.2.2. These components are sealed to indicate that, for some reason, the position of the sealed component requires administrative controls more stringent than those imposed on unsealed plant components. Typically, these components are identified as being sealed on OVID drawings. For the purpose of this procedure, these components shall be classified as CATEGORY 2 SEALED COMPONENTS.

2.2.4 A third category of sealed components consists of controlled flow balancing throttle valves. The seals on these valves are controlled by Engineering procedures as described in Section **Error! Reference source not found.**.

3. <u>RESPONSIBILITIES**{** TC "RESPONSIBILITIES" \F C \L "1" **}**</u>

3.1 The operations manager is responsible for ensuring that all plant operators are trained in the requirements of this procedure, and that the associated operating procedures and OVID drawings are in compliance with this procedure.

3.2 The manager of the appropriate engineering group is responsible for ensuring the controlled flow balancing throttle valves are sealed in accordance with the following procedures:

3.2.1 STP V-14 and V-15 for ECCS flow balancing throttle valves (NSSS Systems).

3.2.2 STP V-13A for CCW to CFCU flow balancing throttle valves (BOP Systems).

3.2.3 STP M-54 for RCP Seal Injection throttle valves (NSSS Systems).

4.1.6 This procedure applies to a variety of components having a number of different operating mechanisms (e.g., handles, levers, handwheels, chain drives, etc.). With the exception of throttled valves, an acceptable seal configuration is one that is both visible and physically impedes operation such that a broken seal will result when other than minor movement of the operating mechanism has occurred.

NOTE: Minor movement is defined as movement that will not change the process being controlled by the component.

Throttled valves should be sealed such that minor handwheel movement will break the seal. If a valve cannot be so sealed, it shall be:

a. Sealed as well as possible, and

b. The valve shall be plainly marked as being a sealed valve with the desired valve sealed position indicated.

4.1.7 Seals should be installed so that only a single seal is needed to impede component operation. When the size or configuration of the component is such that a single seal cannot effectively impede its operation, a chain, cable, wire, or other appropriate means may be used in conjunction with a seal. Usually this involves:

a. Chaining the handwheel to the bonnet for valves

b. Using a seal or other equivalent means to join the ends of the chain

4.1.8 Supplies of unused seals must be controlled. It is acceptable to maintain supplies of new seals in the areas such as the control room, operations watch office, auxiliary building control board, and intake auxiliary operator's office. It is unacceptable to store new seals in the plant areas near where they are used.

4.1.9 Discarded broken component seals shall be properly disposed of in appropriate waste containers, at no time shall used or discarded seals be left on plant equipment or on equipment area floors.

4.2 Category 1 Sealed Components**{** TC "Category 1 Sealed Components" \f C \l "2" **}**

4.2.1 All category 1 sealed components shall be identified in the form of a sealed component checklist in the K-10 series of the operating procedures.

4.2.2 Prior to changing plant operating modes, the shift foreman shall verify that all sealed component checklists required to be complete for that mode are current and complete, per OP L-O.

4.2.3 To ensure sealed component checklists are current for mode transitions, the administrative controls required for sealed components shall be complied with following issuance of the sealed component checklist for completion. Attachment **Error! Reference source not found.** provides a sheet which may be filed in the SFM sealed components binder in the associated checklist tab to signify that the checklist controls are in effect. For example, if a sealed component checklist is in progress and a component on the checklist must be repositioned for a clearance, a sealed component change form (see Step **Error! Reference source not found.**) is to be issued to track that the sealed component is out of position.

4.2.4 All category 1 sealed components should be sealed with RED plastic seals, except for the following:

a. When a component must be sealed in a position other than the normal sealed component checklist position. In this case, the component shall be sealed with a GREEN seal or lockwire, such that the abnormal position is evident until restored to its normal position by the procedure. Installation and removal of the GREEN seal or lockwire shall be controlled in a procedure, as well as restoration of the red seals if the K-10 checklist is in effect.

b. Engineering controlled throttled component may use yellow seals.

c. If, for some reason, red plastic seals are temporarily unavailable, any sealing device may be substituted providing it must be physically broken to open the seal and cannot be re-used.

1. Substitute seals, if used, must be documented in the remarks section of the sealed component checklist and shall be replaced with the appropriate red plastic seal as soon as possible.

2. An AT OPPR type action request shall be initiated on any checklist where substitute seals are utilized. This action request shall remain open until the seals are replaced with the appropriate plastic seals.

4.2.5 Category 1 seals may be broken during plant modes when the associated sealed component checklist is required to be current only when authorization by the appropriate shift foreman is received.

a. This authorization may consist of an approved sealed component change form or a plant procedure authorized by the shift foreman which clearly indicates that component seals are to be broken.

b. In cases of emergencies, the shift foreman may issue a verbal authorization to allow an immediate repositioning of the sealed component. This is acceptable providing the required forms and verifications are performed as soon as possible following the actual seal removal and component repositioning.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	010 K6.02	
	Importance	3.2	3.5

Proposed Question:

The plant is operating at 100% power. All systems are lined up and operating normally. PZR pressure transmitters PT-455/PT-456 are selected for control.

PT-455 fails low.

Which of the following describes the expected plant response? Assume no operator action.

A. PORV 474 eventually opens and plant trips on high pressurizer pressure.

B. PORV 474 eventually opens and pressure cycles on the PORV.

C. PORV 456 eventually opens and plant trips on high pressurizer pressure.

D. PORV 456 eventually opens and pressure cycles on the PORV.

Proposed Answer:
D. PORV 456 eventually opens and pressure cycles on the PORV.

Explanation:
A incorrect, PORV 474 will not open due to the failure and plant trip setpoint is above PORV open setpoint.
B incorrect, PORV 474 will not open, 456 opens.
C incorrect, PORV open and trip setpoint are not the same.
D correct, PORV 456 will operate, pressure will cycle on the PORV.

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

Proposed references to be provided to applicants during examination: none

Learning Objective: 4573 - Analyze Pressurizer control logic.

Question Source: Bank # INPO 24653
Modified Bank # ________
New ______

Question History: Last NRC Exam Seabrook 2003
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments:
K/A: 010 K6.02 - Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR

Pressurizer Pressure Channel Failures

Control/Backup Channel Selected	Channel Failure	Indications	Alarms	Plant Response
PT-455/PT-456 (normally selected)	PT-455 Fails HIGH (controlling channel)	● PI-455 fails high ● Pzr Press recorder fails high if PT-455 selected ● PORV and safety valve tailpipe temps increase	● Pzr Press High PK05-16 ● Pzr Safety/Relief Temp PK05-23 ● Prot Chan Activated PK04-06 ● Pzr Relief/Safety Vlvs Open PK05-20 ● Pzr Low Press PK05-17 (small delay)	● PCV-455C will OPEN, <u>independent</u> of controller or channel selector switch position (PC-474B intlk CLOSES it at 2185 psig) ● PCV-474 will OPEN due to high controller output (PC-474B intlk CLOSES it at 2185 psig) ● Spray valves will open (manual action needed to avoid trip and/or SI) ● Heaters will turn off (proportional htrs at min - bkr still closed) ● If press is restored to >2185 psig, PORVs will re-open and cycle near the interlock press (2185 psig)
	PT-455 Fails LOW (controlling channel)	● PI-455 fails low ● Pzr Press recorder fails low if selected to PT-455	● Pzr Low Press PK05-17 ● OTΔT C-3 Activated PK04-04 ● Prot Chan Activated PK04-06 ● Safeguard Chan Act PK02-04	● Heaters will turn on (all); Spray valves cannot open in auto ● PCV-455C will not open due to PT-455 failure (<u>independent</u> of controller) ● PCV-474 will not open due to low ctrlr signal; PCV-456 will still function ● Press will slowly increase to 2335 psig and PORV will cycle
	PT-456 Fails HIGH (backup channel)	● PI-456 fails high ● PORV and safety valve tailpipe temps increase	● Pzr Press High PK05-16 ● Pzr Safety/Relief Temp PK05-23 ● Prot Chan Activated PK04-06 ● Pzr Relief/Safety Vlvs Open PK05-20 ● Pzr Low Press PK05-17 (small delay)	● PCV-456 will OPEN <u>independent</u> of controller or channel selector switch position (PC-457E intlk CLOSES it at 2185 psig) ● Heaters will turn on due to low press from ctrl channel ● When press is restored to >2185 psig, PCV-456 will re-open and cycle near the interlock press (2185 psig)
	PT-456 Fails LOW (backup channel)	● PI-456 fails low ● PZR Press recorder fails low if selected to PT-456	● OTΔT C-3 Activated PK04-04 ● Prot Chan Activated PK04-06 ● Safeguard Chan Act PK02-04	● PCV-455C and PCV-474 will still function, but PCV-456 will not function due to PT-456 failure (<u>independent</u> of controller) ● No plant transient occurs due to this failure
PT-457/PT-456	PT-457 Fails HIGH (controlling channel)	● PI-457 fails high ● PZR Press recorder fails high if PT-457 selected ● PORV and safety valve tailpipe temps increase	● Pzr Press High PK05-16 ● Pzr Safety/Relief Temp PK05-23 ● Prot Chan Activated PK04-06 ● Pzr Relief/Safety Vlvs Open PK05-20 ● Pzr Low Press PK05-17 (small delay)	● PCV-474 will OPEN due to high controller output (PC-474B intlk CLOSES it at 2185 psig); PCV-456 intlk PC-457E is not effective (failed high) ● Spray valves will open (manual action needed to close these to avoid trip and/or SI) ● Heaters will turn off (proportional htrs at min - bkr still closed) ● If press is restored to >2185 psig, PORV will re-open and cycle near the interlock press (2185 psig)
	PT-457 Fails LOW (controlling channel)	● PI-457 fails low ● PZR Press recorder fails low if selected to PT-457	● Pzr Low Press PK05-17 ● OTΔT C-3 Activated PK04-04 ● Prot Chan Activated PK04-06 ● Safeguard Chan Act PK02-04	● Heaters will turn on (all); Spray valves cannot open in auto ● PCV-474 will not open due to controlling chan failure, and PCV-456 will not open due to interlock chan failed low; PCV-455C will still function ● Pressure will slowly increase until PORV lifts and cycles
	PT-456 Fails HIGH or LOW (backup chan)	See PT-456 response above as backup channel	See PT-456 response above as backup channel	See PT-456 response above as backup channel
PT-455/PT-474	PT-455 Fails HIGH or LOW (ctrl chan)	See PT-455 response above as ctrl channel	See PT-455 response above as ctrl channel	See PT-455 response above as controlling channel
	PT-474 Fails HIGH (backup channel)	● PI-474 fails high	● Pzr Press High PK05-16	● PCV-455C and PCV-474 intlk from PC-474B is not effective ● No plant transient occurs due to this failure
	PT-474 Fails LOW (backup channel)	● PI-474 fails low	● OTΔT C-3 Activated PK04-04 ● Prot Chan Activated PK04-06 ● Pzr Low Press PK05-17 ● Safeguard Chan Act PK02-04	● PCV-455C and PCV-474 will not open due to intlk PC-474B failed low; PCV-456 will still function ● No RCS pressure transient will occur

<u>Note</u>: If a controlling pressurizer pressur channel has failed low, ensure that the master pressure controller (HC-455K) is placed in manual <u>prior</u> to selecting an operaable control channel; not selecting manual first could

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	012 K5.01	
	Importance	3.3	3.8

Proposed Question:

Which of the following describes the type of core protection afforded by the Reactor Protection System Overtemperature DeltaT trip?

A. Power density

B. Fuel Overheating

C. KW per linear foot

D. Departure from nucleate boiling

Proposed Answer:

D. Departure from nucleate boiling

Explanation:
A incorrect, This is for Overpower DT
B incorrect, This is for Overpower DT
C incorrect, This is for Overpower DT
D correct, The inputs to the Overtemperature DT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop DT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system.

Technical Reference(s): TS 3.3.1 bases

Proposed references to be provided to applicants during examination: none

Learning Objective: 9942 Analyze the reactor trip logic, inputs and setpoints

Question Source: Bank # INPO 22988
Modified Bank # ________

New ______

Question History: Last NRC Exam Prairie Island 2, 2002

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.5

55.43 _____

Comments: K/A: 012 K5.01 - Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Source Range Neutron Flux (continued)

Above the P-6 setpoint, the NIS source range neutron flux trip may be manually blocked and the high voltage to the detectors may be de-energized. Below the P-6 setpoint, the source range neutron flux trip is automatically reinstated and the high voltage to the detectors is automatically energized. In MODES 3, 4, and 5 with the reactor shut down, but with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted, the Source Range Neutron Flux trip Function must also be OPERABLE (1-out-of-2 coincidence) to provide core protection against a rod withdrawal accident. If the Rod Control System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like an uncontrolled boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection and it protects against vessel exit bulk boiling and ensures that the exit quality is within the limits defined by the DNBR correlation. The inputs to the Overtemperature ΔT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure—the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution—f(ΔI), the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

ΔT_0, as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the measured ΔT value from the previous cycle until ΔT is once again measured at full power. Accurate determination of the loop specific ΔT values are made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distribution conditions not affected by xenon or other transient conditions). The indicated ΔT variation between loops is due to the difference between hot leg temperatures and hot leg temperature measurement biases. The hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases primarily caused by differences in hot leg temperature streaming error between loops. The loop ΔTs change with burn up which result from the change in the hot leg streaming biases as the radial power distribution changes.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. The pressure and temperature signals are used for other control functions; thus the actuation logic must be able to withstand an input failure to the

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE . Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions.

Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions. In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB (2-out-of-4 coincidence). In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions for Condition I and II events (Ref. 12). This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux—High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. The Overpower ΔT trip also provides protection to mitigate the consequences of small steamline breaks, as reported in WCAP-9226, Ref. 16, and steamline breaks with coincident control rod withdrawal (Ref. 3). It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature—including dynamic compensation for the delays between the core and the temperature measurement system.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

ΔT_0 , as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the measured ΔT value from the previous cycle until ΔT is once again measured at full power. Accurate determination of the loop specific ΔT values are made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distribution conditions not affected by xenon or other transient conditions). The indicated ΔT variation between loops is due to the difference between hot leg temperatures and hot leg temperature measurement biases. The hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases is primarily caused by differences in hot leg temperature streaming error between loops. The loop ΔTs change with burn up which result from the change in the hot leg streaming biases as the radial power distribution changes.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. The temperature signals are used for other control functions; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the trip setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE (2-out-of-4 coincidence). Note that the Overpower ΔT trip Function receives input channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure—High and —Low trips and the Overtemperature ΔT trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Pressurizer Pressure—Low

The Pressurizer Pressure—Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure—Low to be OPERABLE (2-out-of-4 coincidence).

In MODE 1, when DNB is a major concern, the Pressurizer Pressure—Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 Low Pressure Permissive interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, there is insufficient heat production to be concerned about DNB.

b. Pressurizer Pressure—High

The Pressurizer Pressure—High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires four channels of the Pressurizer Pressure—High to be OPERABLE (2-out-of-4 coincidence).

The Pressurizer Pressure—High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	013 K5.01	
	Importance	2.8	3.2

Proposed Question:

Unit 1 is at power, in a normal SSPS lineup.

PK02-02, Safety Injection Initiate, alarms.

What does this PK alarm mean to the operator?

A. At least one safeguard channel has actuated.

B. At least two safeguards channels for same parameter have actuated.

C. The minimum number of channels for SI initiation has been met and at least one Train of SI has actuated.

D. The minimum number of channels for SI initiation has been met and both Trains of SI have actuated.

Proposed Answer:

B. At least two safeguards channels for same parameter have actuated.

Explanation:
A incorrect, one channel trip actuates PK02-04, Safeguard Channel Activated.
B correct, all SI actuate signals need 2 (or more) bistables to trip. When the coincidence is met, SI Initiate signal is generated and the PK actuates.
C incorrect, this generates PK08-21, SI Actuation
D incorrect, this will generate PK08-21

Technical Reference(s): PK02-02, PK02-04 and PK08-21

Proposed references to be provided to applicants during examination: none

Learning Objective:
5410 - Identify the SSPS parameters that produce alarms

Question Source: Bank #
Modified Bank # _______
New X

Question History: Last NRC Exam _____________
Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments:

K/A: 013 K5.01 - Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Definitions of safety train and ESF channel

TITLE: SAFEGUARD CHANNEL ACTIVATED

1

08/10/99
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM

#

#

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
K0605	44	Stmline Press Low Loop 1 1/3	LT 600 PSIG
K0606	44		
K0607	44		
K0608	45	Stmline Press Low Loop 2 1/3	LT 600 PSIG
K0609	45		
K0610	45		
K0611	46	Stmline Press Low Loop 3 1/3	LT 600 PSIG
K0612	46		
K0613	46		
K0701	48	Stmline Press Low Loop 4 1/3	LT 600 PSIG
K0702	48		
K0703	48		
K0512	42	Contmt Hi Press 1/3	GT 3 PSI
K0513	42	Contmt Hi Press 1/3	GT 3 PSI
K0514	42	Contmt Hi Press 1/3	GT 3 PSI
K0601	43	Contmt Hi-Hi Press 1/4	GT 22 PSI
K0602	43	Contmt Hi-Hi Press 1/4	GT 22 PSI
K0603	43	Contmt Hi-Hi Press 1/4	GT 22 PSI
K0604	43	Contmt Hi-Hi Press 1/4	GT 22 PSI
K0907	1316	Pzr Press Lo Ch IV	LT 1850 PSIG
K0909	1317	Pzr Press Lo Ch II	LT l850 PSIG
K0908	59	Pzr Press Lo Ch I	LT l850 PSIG
K0910	1318	Pzr Press Lo Ch III	LT l850 PSIG
K1211	63	RCS Loop 1-2 Lo-Lo Tavg	LT 543°F
K1210	75	RCS Loop 1-1 Lo-Lo Tavg	LT 543°F
K1212	69	RCS Loop 1-3 Lo-Lo Tavg	LT 543°F
K1213	64	RCS Loop 1-4 Lo-Lo Tavg	LT 543°F
K1301	76	Stm Press Rate Hi 1/3 Loop 1	100 PSI
K1303	77	Stm Press Rate Hi 1/3 Loop 2	100 PSI
Kl305	78	Stm Press Rate Hi 1/3 Loop 3	100 PSI
Kl307	79	Stm Press Rate Hi 1/3 Loop 4	100 PSI

#

#

3. PROBABLE CAUSE

3.1 One or more safeguard channel Bi-stables activated due to a channel failure or a plant upset resulting in any one or more of the following conditions:

3.1.1 Steamline pressure decrease

3.1.2 Steamline Lo pressure

3.1.3 Lo-Lo T_{AVG}

3.1.4 Pressurizer Lo Pressure

3.1.5 Contmt. Hi pressure

3.1.6 Contmt. Hi-Hi pressure

4. AUTOMATIC ACTIONS

4.1 Possible safety injection actuation.

4.2 Possible reactor trip.

5. OPERATOR ACTIONS

5.1 Check main annun. printout.

5.2 Check control room status lights, main annun's., and instrumentation for plant conditions.

5.3 If a reactor trip or safety injection occurs, go to EOP E-0.

5.4 If plant conditions require a reactor trip or safety injection, initiate same and go to EOP E-0.

5.5 If alarm is due to an upset in plant conditions, determine cause of upset and restore plant conditions to the normal range.

5.6 If alarm is due to a channel failure:

5.6.1 Refer to OP AP-5 Malfunction of Protection or Control Channel.

5.6.2 Refer to the Technical Specifications 3.3.2 (ITS 3.3.2) for required restrictions on Plant Operation.

5.7 If an alarm is reflashing due to a SSPS problem in one train (as indicated by a status light flashing on and off at a constant interval), the reflashing can be stopped by placing the SSPS Train A multiplexer test switch in "NORMAL." Refer to STP I-38AB.5, Appendix 10.2 for guidance. Notify MS of the SSPS problem.

PACIFIC GAS AND ELECTRIC COMPANY		**NUMBER**	**AR PK02-02**
NUCLEAR POWER GENERATION		**REVISION**	**9**
DIABLO CANYON POWER PLANT		**PAGE**	**1 OF 2**
ANNUNCIATOR RESPONSE		**UNIT**	

TITLE: SAFETY INJECTION INITIATE (RED)

1

APPROVED: ________________________ **04/13/94** **04/18/94**

DATE **EFFECTIVE DATE**

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
1 SI/CS	1133	Manual S.I.	
2 SI/CS	1133	Manual S.I.	
K0201	15	Pzr. Lo Press. 2/4 SI	LT l850 PSIG
K0206	20	Stmline Press Low S.I. 1/4	LT 600 PSIG
K0207	20	Stmline Press Low S.I. 1/4	LT 600 PSIG
K0208	20	Stmline Press Low S.I. 1/4	LT 600 PSIG
K0209	20	Stmline Press Low S.I. 1/4	LT 600 PSIG
K0205	19	Contmt Hi Press 2/3 SI	GT 3 PSIG

3. PROBABLE CAUSE

   3.1 Pressurizer Low pressure.

   3.2 Low steamline pressure.

   3.3 Containment hi pressure.

   3.4 Manual S.I. initiation.

4. AUTOMATIC ACTIONS

   4.1 Safety injection actuation.

   4.2 Reactor trip.

   4.3 Possible steam line isolation

5. OPERATOR ACTIONS

   5.1 Go to EP E-0 Reactor Trip or Safety Injection.

PACIFIC GAS AND ELECTRIC COMPANY	**NUMBER** AR PK08-21
NUCLEAR POWER GENERATION	**REVISION** 0
DIABLO CANYON POWER PLANT	**PAGE** 1 OF 1
ANNUNCIATOR RESPONSE	**UNIT**

TITLE: SAFETY INJECTION ACTUATION (RED)

1

APPROVED: ______________________ **04/19/94** **04/21/94**
DATE **EFFECTIVE DATE**

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
K1811	1371	Safety Injection Actuation	

3. PROBABLE CAUSE

   3.1 Safety injection initiate signal.

4. AUTOMATIC ACTIONS

   4.1 Safety injection actuation.

5. OPERATOR ACTIONS

   5.1 Go to EP E-0.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	022 A1.02	
	Importance	3.6	3.8

Proposed Question:

Unit 1 is at full power.

The following events occur:

- A cold leg LOCA and complete loss of offsite power occurs
- Bus F and H become de-energized.
- All other equipment operate as designed

Which of the following describes why containment design pressure may or may not be exceeded?

A. Not exceeded - because the minimum requirement of one train of Containment spray and 2 CFCU's will be met.

B. Not exceeded - because the minimum requirement of one train of Containment spray and 3 CFCU's will be met.

C. Exceeded - because the minimum requirement of one train of Containment spray and 2 CFCU's will not be met.

D. Exceeded - because the minimum requirement of one train of Containment spray and 3 CFCU's will not be met.

Proposed Answer:
A. Not exceeded - because the minimum requirement of one train of Containment spray and 2 CFCU's will be met.

Explanation:
A correct. Minimum requirement is 1 train of CS and 2 CFCU's. Train A CS and CFCU 3 and 5 will have power from Bus G.
B incorrect. Only 2 CFCUs will be energized.
C incorrect, the minimum requirements are met.
D incorrect, this is not the minimum requirement.

Technical Reference(s): H2 – CFCU and I2 – Containment Spray

Proposed references to be provided to applicants during examination:

Learning Objective: 6141 - State the power supplies to CFCU system components.
6022 - State the power supply to Containment Spray pumps.
3417 - State the purpose of the CFCU system.

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 _____

Comments: K/A: 022 A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure

Containment Fan Cooler Units (CFCUs), Continued

Power Supplies
Obj 8

Power supplies for the CFCUs are shown below.

CFCU	480 V vital bus
1	F
2	F
3	G
4	H
5	G

Physical description

The physical description of the CFCUs are described in the following table.

CFCU Description:

Characteristic	Details
Type	Single stage, centrifugal
Rated flow (LOW)	47,000 to 57,000 CFM
Rated flow (HIGH)	99,000 to 121,000 CFM
Fan drive	480 VAC 300 hp induction motor
Other characteristics	Fan assembly is protected by a vacuum relief damper to limit the pressure differential across the CFCU enclosure.
Material	Carbon steel covered by a protective coating to prevent corrosion of the assembly.

Continued on next page

Design Information

Design basis
Obj 5

The design bases are tabulated below.

The Containment Spray system is designed to...	**Reason**
operate in conjunction with the Containment Fan Cooler Units (CFCUs). One train of Containment Spray and two of five CFCUs provide sufficient heat removal to maintain Containment pressure below its design value of 47 psig following a design basis LOCA or MSLB.	The spray is used to help condense the steam resulting from a LOCA or MSLB. The CFCUs further reduce the temperature of the air and water vapor mixture.
provide post-accident Containment heat removal, and is therefore designated Design Class I.	This design accommodates all postulated events.
provide a time delay on starting CSS pumps: • A minimum time delay of 23 seconds • A maximum delay of 41 seconds A Containment Spray piping fill time of 39 to 44 seconds is required for flow to reach the spray nozzles.	• The spray piping needs to remain dry until after the maximum seismic loading has subsided after an earthquake, • Cooling is needed in the Containment atmosphere in a LOCA. The piping in Containment is normally dry.

The Containment Spray system Non Safety-related functions during the re-circulation phase are...	**Benefit...**
• continued heat removal from the Containment atmosphere • continued removal of fission products from the Containment atmosphere • enhanced mixing of the Containment atmosphere	• The spray is used to continue to lower Containment temperature. • The spray further reduces fission products in the Containment atmosphere. • Mixing of Containment atmosphere minimizes hydrogen pockets.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	022 G2.4.34	
	Importance	3.8	3.6

Proposed Question:

The crew is performing the actions of OP AP-8A, Control Room Inaccessibility – Establishing Hot Standby.

The operator is directed to start CFCU 1-2.

What minimum condition(s) must be met for CFCU 1-2 to start when the operator takes the Control Switch to ON?

A. At the HSDP - Control Transfer Switch to LOCAL.

B. At the 480V Auxiliary Relay Panel - Control Transfer Relay to CUTIN.

C. At the HSDP - Control Transfer Switch to LOCAL.
At the HSDP - Control Transfer Relay to CUTIN.

D. At the HSDP - Control Transfer Switch to LOCAL.
At 480V Auxiliary Relay Panel - Control Transfer Relay to CUTIN.

Proposed Answer:
D. At the HSDP - Control Transfer Switch to LOCAL.
At 480V Auxiliary Relay Panel - Control Transfer Relay to CUTIN.

Explanation:
A incorrect, Control Transfer Relay must also be in CUTIN.
B incorrect, Control Transfer Switch must be CUTIN
C incorrect, Control Transfer Relay switch on 480V Aux Relay Panel.
D correct, Control switch in LOCAL at HSDP and Transfer Relay in CUTIN at 480V Aux Relay panel.

Technical Reference(s):
A8 – Remote/Hot Shutdown Panels
OP 8A, Appendix F

Proposed references to be provided to applicants during examination:
OP 8A, Appendix F

Learning Objective: 4480 Explain the operation of CFCU system at hot shutdown panel

Question Source:
New X

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
55.43

Comments:
K/A: 022 G2.4.34 – Containment Cooling - Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.

APPENDIX F

480V Bus Alignment

1. 480V Vital Bus F

   a. Place Control Transfer Cutout Switches (green Lamicoids) to CUT-IN

EQUIPMENT	SWITCH	POSITION
CFCU 1-2	43X-1F-1	CUT-IN
Letdown Orifice	43BX	CUT-IN

   b. Open the following breakers to prevent spurious operation.

FCV-430	52-1F-11	OPEN
LCV-112B	52-1F-12	OPEN
8805A	52-1F-19	OPEN
FCV-750	52-1F-23	OPEN

2. 480V Vital Bus G

   a. Place Control Transfer Cutout Switches (green Lamicoids) to CUT-IN

EQUIPMENT	SWITCH	POSITION
CFCU 1-5	43X-1G-2	CUT-IN
BATP 1-2	43X-1G-4	CUT-IN
AFW Pp 1, FCV-95	43X-12-30	CUT-IN
LCV-106	43X-1G-44	CUT-IN
Emerg Borate 8104	43X-1G-57	CUT-IN
LCV-107	43X-1G-68	CUT-IN

   b. Open the following breakers to prevent spurious operation.

LCV-112C	52-1G-11	OPEN
8805B	52-1G-14	OPEN
FCV-363	52-1G-23	OPEN
FCV-431	52-1G-28	OPEN
FCV-356	52-1G-36	OPEN
9003A	52-1G-48	OPEN
8982A	52-1G-58	OPEN

Phone Nos. HSDP x3107, x3781 DSDP x1355, x3179
D/G 1-1 x1249 D/G 1-2 x1251, D/G 1-3 x1438
12kV Rms x1247 4kV Rms x1747 480 V Rms 1372

APPENDIX F (Continued)

3. 480V Bus H

   a. Open the following breakers to prevent spurious operation.

9003B	52-1H-6	OPEN
8982B	52-1H-12	OPEN
FCV-355	52-1H-16	OPEN
FCV-357	52-1H-17	OPEN
FCV-749	52-1H-18	OPEN

4. 480V Bus Section 13D

   a. Place Control Transfer Cutout Switches (green Lamicoids) to CUT-IN (located inside panel behind control transfer relay).

EQUIPMENT	SWITCH	POSITION
PZR Htr Group 2	43X-3D-6	CUT-IN

5. Inform the HSDP Operator that 480V Bus Alignment is complete.

Phone Nos. HSDP x3107, x3781 DSDP x1355, x3179
D/G 1-1 x1249 D/G 1-2 x1251, D/G 1-3 x1438
12kV Rms x1247 4kV Rms x1747 480 V Rms 1372

CFCU Control Switches

Purpose
Obj 7

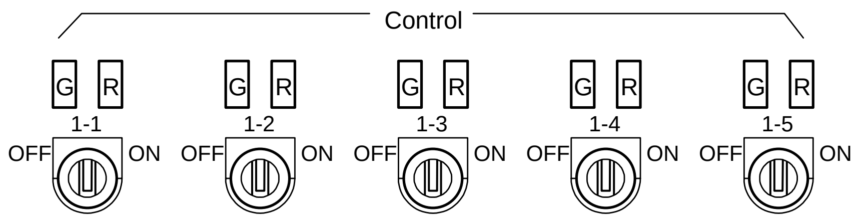
The purpose of the CFCU Control Switches is to operate the CFCUs 1 through 5 after control has been transferred to the HSDP.

Location

The CFCU Control Switches are located inside of the HSDP. Refer to the Basic Description Section for the specific location of the switch inside of the HSDP.

Controls
Obj 24

The CFCU Control Switches shown below are available at the HSDP.



RSD-27

Control	Operation
OFF/ON	2 position, spring return to neutral

Logic
Obj 9,24

The logic associated with the CFCU Control Switches at the HSDP is described in the table below:

With the CONTROL TRANSFER SWITCH in the LOCAL position and the Control Transfer Relay CUTIN	
If the control switch is in ...	**Then the CFCU will ...**
ON	Start in high speed.
OFF	Stop
Neutral	Start on a SI signal in slow speed.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	026 K4.01	
	Importance	4.2	4.3

Proposed Question:

Unit 1 was at 100% in a normal lineup when a large LOCA occurs.

While performing the actions of EOP E-1.3, “Transfer to Cold Leg Recirculation”, the crew opens 9003A, RHR Pp 1 to Spray Hdr A.

When is the logic satisfied to open 9003A?

A. When the RWST decreases to less than 4%.

B. When 8809A, RHR to Cold Legs 1 and 2 is closed.

C. When 9001A and B, Containment Spray Pp Discharge Vlvs, are closed.

D. When 8982A, RHR Pp No. 1 Suct From Contmt Recirc Sump, is opened.

Proposed Answer:

D. When 8982A, RHR Pp No. 1 Suct From Contmt Recirc Sump, is opened.

Explanation:
A incorrect, this is a prerequisite for opening the valve, but not an interlock.
B incorrect, this is a prerequisite for opening the valve, but not an interlock. The loop suctions are closed at power and 8982B is opened at step 6.
C incorrect, this is a prerequisite for opening the valve, but not an interlock.
D correct, to open 9003A, 8982A/B must be open and 8701or 8702, (loop suction valves) closed (which they are at power).

Technical Reference(s): B2 – Residual Heat Removal
I2 – Containment Spray

Proposed references to be provided to applicants during examination: none

Learning Objective: 6043 – Analyze interlocks associated with Containment Spray System valves.

Question Source:
New X

Question History: Last NRC Exam ______
Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments:

K/A: 026 K4.01 - Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Source of water for CSS, including recirculation phase after LOCA

RHR to Containment Spray Isolations, 9003A/B, Continued

Logic

The logic for 9003A/B is described below.

If the switch is in...	Then the valve will...
OPEN	open if: • 8701 or 8702 loop suction valve, is closed, and • 8982A/B, Containment suction, is open. Interlock is designed to prevent inadvertent spray of the Containment from RCS during normal operation and to insure RHR is in recirc mode during accident conditions.
STOP	will stop valve movement.
CLOSE	close.
Other features	**Description**
Closing interlock	The full closed contact is jumpered out, the valve is allowed to torque out in the closed direction to assure positive seating.
Torque switch bypasses	Each torque switch is bypassed by a contact that closes when the valve is within 10% of its opposite position, e.g., the open torque switch is bypassed by a contact that closes when the valve is $\geq$ 90% closed. • This prevents the valve from being stopped if the torque switch opens on breakaway torque as it starts to move open.

Continued on next page

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 16

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 • 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 • 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

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	039 A1.05	
	Importance	3.2*	3.3

Proposed Question:

PLANT CONDITIONS:

- Unit 1 is at 3% power, EOL
- Steam Dumps are maintaining RCS Tave in AUTOMATIC in Steam Pressure mode

Which of the following describes the effect of the Steam Dump pressure controller, HC-507 pot setting being changed from 8.38 turns to 9.10 turns?

A. Tavg would INCREASE and reactor power would DECREASE.

B. Tavg would INCREASE and reactor power would REMAIN THE SAME.

C. Tavg would DECREASE and reactor power would DECREASE.

D. Tavg would DECREASE and reactor power would REMAIN THE SAME.

Proposed Answer:

A. Tavg would INCREASE and reactor power would DECREASE.

Explanation:
A correct, raising the setpoint causes Tavg to increase the negative feedback with rods in manual would cause power to decrease.
B incorrect, power would decrease.
C incorrect, Tavg would increase.
D incorrect, Tavg would increase, power would decrease.

Technical Reference(s): C-2B – Steam Dump

Proposed references to be provided to applicants during examination: none

Learning Objective: 8006 – Explain the effects of operating Steam Dump System controls

Question Source: Bank #
Modified Bank # DCPP A-0731
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 41.5
55.43 _____

Comments:

K/A: 039 A1.05 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: RCS T-ave

1 A-0731 Points: 1.00 Multiple Choice

A reactor start up is in progress with the following plant conditions:

•Control rods	Manual
•Reactor power	5%
•IR SUR	0
•Steam Dumps	Pressure mode
•HC-507	AUTO

What would happen if the steam dump pressure controller HC-507 pot setting were to be changed to 9.10? (Normal setting is 8.38)>>

A. Tavg would increase and reactor power would decrease.

B. Tavg would remain the same and reactor power would decrease.

C. Tavg would remain the same and reactor power would increase.

D. Tavg would decrease and reactor power remain the same.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8006	Explain the effects of operating Steam Dump System controls

Reference Id:	A-0731
Must appear:	No
Status:	Active
User Text:	8006.130741
User Number 1:	3.00
User Number 2:	3.70
Difficulty:	3.00
Time to complete:	3
Topic:	SDS - Steam dump steaming rate affect on power and Tavg
Cross Reference:	DUTY AREA 74, LC-2B
Comment:	Checked OK, 1/10/97. No changes. MAP2

Copied from S-1311; 11/11/96, GES1
Validated question IAW TQ2.ID3 1/23/97 RCWf
Taken active following review, 1/25/97 JMH1
LC-2B
Reviewed by R973 test review group, reformatted stem, 8/29/97, jmb1

Checked as part of question review for biennial exam 12/22/98 MTC6
Reviewed for 00 biennial exam 1/17/01 mtc6
Reviewed for 02 biennial exam 1/7/03 mtc6
Reviewed for 04 biennial exam 12/21/04 mtc6

Major DCPP Events, Continued

LER 323-000914
Obj 32

On September 14, 2000, operators were preparing for the Diablo Canyon Unit 2 plant startup following a mode 3 forced outage. In order to place the 40 percent condenser steam dumps in service, the potentiometer (POT) setting for the steam dump controller HC-507 was required to be set at 1,005 psig (8.38 turns). With the controller in manual, the POT setting was peer-checked following adjustment by the control operator. After HC-507 was placed in automatic, the 40 percent steam dumps opened with a subsequent secondary pressure and reactor coolant system (RCS) pressure and temperature transient.

The transient was terminated quickly after HC-507 was taken back to manual and the steam dumps were closed. During the transient, steam header pressure dropped to 924 psig and the rate sensitive low steam line pressure safety injection (SI) setpoint reached 797 psig (600 psig setpoint). Two additional licensed individuals observed the HC-507 POT setting following the transient and believed it to have been set correctly.

The control room contacted technical maintenance (TM) personnel and requested HC-507 troubleshooting. Prior to the start of controller troubleshooting on the following shift, the relieved control operator decided that the HC-507 POT setting warranted another look; this time using a magnifying glass to enhance viewing of the small numbering. Operators determined that the POT setting had been set to 6.38 turns instead of the required 8.38 turns

Normal Operations, Continued

Control system alignment diagram

Refer to OIM figures C-2-3, 4 & 5 for normal control system alignment of the Steam Dump system.

Pressure Control mode
Obj 17

Refer to OIM figures C-2-3, 4 & 5 for the following discussion.
The Steam Pressure mode of Steam Dump control is normally used when reactor power is less than approximately 10%. This includes plant cooldown and stabilizing temperature on heatup.

Group 1 and group 2 Steam Dump valves only are affected in Steam Pressure mode.

In automatic pressure sensed by PT-507 is compared to the setpoint selected on the Steam Pressure controller. This is normal set at 1005 psig at Hot Standby and above. The set point could be varied below Hot Standby to control heatup/cooldown. However, the preferred method for cooldown is to place HC-507 in manual and use the up/down pushbuttons (direct control of controller output) to adjust valve position for a more constant cooldown/heatup [TRP2].

Steam Pressure mode is selected by the Mode Selector switch 43/SDI on CC-2. The other requirements for operation in Steam Pressure mode are:
- Main condenser available C-9.
- Bypass switch not in OFF/RESET.
- T_{avg} not less than 543°F P-12. P-12 can be bypassed for control of group 1 valves only.
- Manual or Auto selected at HC-507 on CC-3

Total valve demand from the Steam Pressure controller can be read on UI-500 on VB-3 or HC-507 demand with the following relationships.

Group	Controller Demand	Group Position
1	0 - 15.625%	0 - 100%
2	15.625 - 46.875%	0 - 100%
3	46.875 - 87.5%	0 - 100%
4	87.5 - 100%	0 - 100%

Continued on next page

RO QUESTION 44

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	039 A4.03	
	Importance	2.8*	2. 8*

Proposed Question:

What is the programmed auto setpoint delta-P for the Digital Feedwater Control System Feed Pump Speed Control?

A. Ramps from 74 psid to 170 psid from 0-100% load. Load is indexed from total feedwater flow.

B. Constant at 74 psid from 0-20% load, ramps from 74 psid to 170 psid from 20-100% load. Load is indexed from total feed flow.

C. Ramps from 74 psid to 170 psid from 0-100% load. Load is indexed from total steam flow.

D. Constant at 74 psid from 0-20% load, ramps from 74 psid to 170 psid from 20-100% load. Load is indexed from total steam flow.

Proposed Answer:

D. Constant at 74 psid from 0-20% load, ramps from 74 psid to 170 psid from 20-100% load. Load is indexed from total steam flow.

Explanation:
A incorrect, constant from 0 to 20%. indexed to total steam flow
B incorrect, indexed to total steam flow
C incorrect, constant from 0 to 20%
D correct, see graph

Technical Reference(s): C-8B - DFWCS

Proposed references to be provided to applicants during examination: none

Learning Objective: 4330 – Explain the operations associated with DFWCS

Question Source: Bank # P-0172

Modified Bank # ________
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 41.4
55.43 _____

Comments:

K/A: 039 A4.03 - Ability to manually operate and/or monitor in the control room: MFW pump turbines

1 P-0172 Points: 1.00 Multiple Choice

What is the programmed auto setpoint delta-P for the Digital Feedwater Control System Feed Pump Speed Control?

A. Constant at 74 psid from 0-20% load, ramps from 74 psid to 170 psid from 20-100% load. Load is indexed from total steam flow.

B. Constant at 74 PSID from 0-20% load, ramps from 74 psid to 170 PSID from 20-100% load. Load is indexed from total feedwater flow.

C. Ramps from 74 psid to 170 psid from 0-100% load. Load is indexed from total feedwater flow.

D. Ramps from 74 psid to 170 psid from 0-100% load. Load is indexed from total steam flow.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

4330	Explain the operations associated with DFWCS

Reference Id: P-0172
Must appear: No
Status: Active
User Text: 4330.050593
User Number 1: 3.00
User Number 2: 3.10
Difficulty: 1.00
Time to complete: 2
Topic: prog auto stpt of delta P for DFWCS
Cross Reference: STG C-8B
Comment:

Section 2.3

Main Feed Pump Control

Overview

MFP ΔP considerations

The main feed pumps (MFPs) are operated according to the programmed ΔP from steam pressure and feed pressure. The ΔP program is a function of total steam flow. The discharge pressure required to feed water into the S/Gs is a function of the following factors:

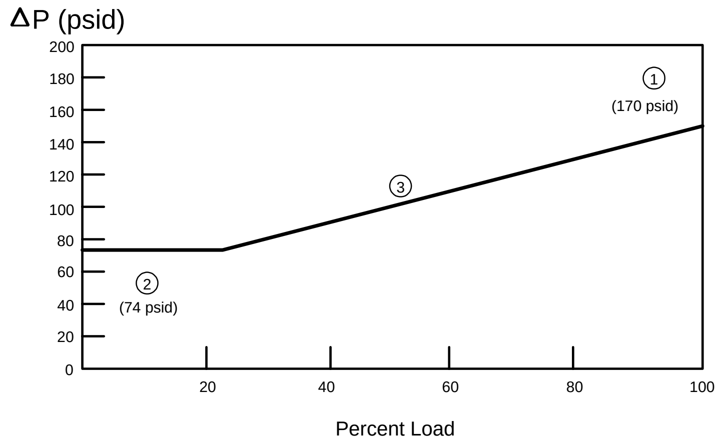
- The feedwater pump must raise the pressure of the water entering its suction to value that is at least as high as S/G pressure. Steam pressure drops with plant load so pump discharge pressure will drop as well.
- Feed pump pressure must also overcome the pressure due to the elevation difference from pump to feed ring.
- As a result of flow, there is a throttling (head) loss across the partially open feedwater control valve. The pressure drop will rise as the flow through the valve increases.
- As a result of fluid flow, there are frictional (head) losses through all the various piping and components from the feedwater system to the S/Gs.
- The ΔP program allows main feedwater reg valve position to remain within the middle band of the total valve stroke over the entire range of turbine load.
  - ♦ An advantage to this is a linear relationship between % of valve opening and % of total flow through the valve (improved throttling characteristics).
  - ♦ Another advantage is reduced erosion of valve internals and thereby improved reliability.

Continued on next page

Overview, Continued

MFP ΔP program diagram
Obj 8

The MFP ΔP program is shown below.



DFWCS-16

Stage	Description
1	The programmed ΔP for full load (170 psid), is designed to overcome all the head losses incurred between the feedwater pumps and the S/Gs.
2	The programmed ΔP for no load, (74 psid), is based on: • minimizing erosion of control valve seats and discs, • erratic steam flow measurements at low power levels.
3	The feedwater pump speed control system automatically regulates the speed of the feed pumps to maintain the programmed ΔP.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	059 K3.04	
	Importance	3.6	3.8

Proposed Question:

Unit 1 is at 75% power when MFP 1-1 trips. The plant operates as designed.

Which of the following describes the ***initial*** RCS temperature and pressurizer level response?

A. Power mismatch causes RCS temperature to decrease, which causes pressurizer level to decrease.

B. Power mismatch causes RCS temperature to increase, which causes pressurizer level to increase.

C. Program response of steam dumps and control rods causes both RCS temperature and pressurizer level to decrease.

D. Rod insertion causes RCS temperature to decrease while charging mismatch causes pressurizer level to increase.

Proposed Answer:

B. Power mismatch causes RCS temperature to increase, which causes pressurizer level to increase.

Explanation:
"…small load rejection resembles a normal ramp when looking at only the end points of the transient. During the transient, however, TAVG will rise several degrees above normal, prior to stabilizing at its final value. Pressurizer level will initially rise, from a beginning value of 42%, and then decrease to a final value of 35%. An accompanying change in pressurizer pressure will also occur."
A incorrect, the power mismatch is primary greater than secondary, the RCS will heat up.
B correct, the primary to secondary mismatch will cause an initial heat up and swell on the primary side.
C incorrect, the control systems will eventually take over but not initially.
D incorrect, RCS temperature increases.

Technical Reference(s): TH18T – Transient Analysis page 25, 26

Proposed references to be provided to applicants during examination: none

Learning Objective:
10583 - DESCRIBE the reactor, RCS and Secondary System responses to each of the following transients:
a. Partial load rejection with rods in automatic.
b. Partial load rejection with rods in manual.

Question Source: Bank #
Modified Bank # INPO 24632
New ______

Question History: Last NRC Exam Seabrook 5/2003
Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 _____

Comments:

K/A: 059 K3.04 - Knowledge of the effect that a loss or malfunction of the MFW will have on the following: RCS

TH18T PAGE 25, 26

3.1 PLANT RESPONSES TO SPECIFIC TRANSIENTS

Data for the transients analyzed in this section is listed in Section 1.1 and Section 1.2 of this chapter. Only the middle of cycle (MOC) transients are evaluated. In instances where the BOC or EOC transients are more significant, supportive descriptions are provided.

Partial Load Rejection with Rods in Automatic

The most common type of load rejection is one in which the Main Generator output breakers open (termed a Full Load Rejection). This results in electrical load being reduced to about 5% of full load, and reactor power stabilizing near 30% by dumping steam through the steam dumps. For this analysis, only a partial load rejection is considered. This simplifies the analysis and provides a range of values compatible with both automatic and manual modes of rod control.

During a partial load rejection, control rods insert automatically, at a speed of 72 steps per minute, due to the mismatch between the rate of change of turbine load and reactor power. Once this rate of change dissipates, the rods continue to insert due to T_{AVG} being greater than its programmed value (T_{REF}). The steam dumps open, during the transient, to simulate the turbine load being lost. As T_{AVG} approaches T_{REF}, the steam dumps start throttling closed. For this analysis, no steam dump actuation is assumed. But, since T_{AVG} returns to T_{REF}, due to rod motion, no steam dump actuation is actually required…

In fact, this small load rejection resembles a normal ramp when looking at only the end points of the transient. *During the transient, however, T_{AVG} will rise several degrees above normal, prior to stabilizing at its final value. Pressurizer level will initially rise,* from a beginning value of 42%, and then decrease to a final value of 35%. An accompanying change in pressurizer pressure will also occur. On the secondary side, T_{SAT} will increase about 10°F, and steam generator pressure will increase about 50 psid

ID: 24632

The following plant conditions exist:
- The plant is operating at 80% power.
- B steam generator feed pump (SGFP) trips.
What is the expected response of the RCS?

Ans Turbine setback causes power mismatch that causes control rods to insert; RCS temperature rise causes an initial PZR level swell before returning to program level.
D1 Turbine setback causes power mismatch that cools RCS thus PZR level initially shrinks before returning to program level.
D2 Turbine setback causes power mismatch that causes control rods to insert; RCS temperature rise causes an initial PZR level shrink before returning to program level.
D3 Turbine setback causes power mismatch that causes control rods to insert. No pressurizer level change observable due to action of steam dumps and rod insertion.
AbbrevLocName
Seabrook 1
ExamDate
5/30/2003
Vendor
WEC
Type
PWR
Distract1Comment A) Does not cool RCS
C) Causes swell initially
D) incorrect
Distract2Comment A) Does not cool RCS
C) Causes swell initially
D) incorrect
Distract3Comment A) Does not cool RCS
C) Causes swell initially
D) incorrect
ExamType
ILO
QuestionComment Explanation of answer: Power mismatch will cause RCS temperature increase, and rods to insert, initial swell causes PZR level to increase.
Cog Level
2
ExamLevel
R
RefMaterial
ParentId
KaNumber
..059.K3.04
KaSegment1
KaSegment2
KaSegment3
059
KaSegment4
K3
KaSegment5
04
KaRevision

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	061 K5.05	
	Importance	2.7	3.2

Proposed Question:

Subsequent to a reactor trip from 100% power, a severe water hammer causes the aux feedwater line to Feed Lead 2-1 to break.

The break is just upstream of the check valve that isolates main feedwater from aux feedwater.

How do the following AFW LCVs respond to this malfunction?

- LCV-110 AFW PP 2-2 discharge to S/G 2-1
- LCV-111 AFW PP 2-2 discharge to S/G 2-2
- LCV-106 AFW PP 2-1 discharge to S/G 2-1

A. High AFW flow to S/G 2-1 will result in LCV-110 throttling. LCV-111 will open in an attempt to feed S/G 2-2. LCV-106 will remain open until closed by operator action.

B. Low pressure on the AFW line to S/G 2-1 will send a signal to close LCV-110 and LCV-106. LCV-111 will be available to feed S/G 2-2.

C. Low AFW PP 2-2 discharge pressure will result in throttling LCV-110 and LCV-111. LCV-106 will remain open until closed by operator action.

D. Low pressure on the AFW line will send a close signal to LCV-110. LCV-111 will open in an attempt to feed S/G 2-2. LCV-106 will remain open until closed by operator action.

Proposed Answer:

C. Low AFW PP 2-2 discharge pressure will result in throttling LCV-110 and LCV-111. LCV-106 will remain open until closed by operator action.

Explanation:

A incorrect, valves throttle on low pressure.
B incorrect, valves will not receive a close signal.
C correct, as pressure decreases, both 110 and 111 will throttle to prevent runout. 106 is operated manually.

D incorrect, valve will not receive a close signal.

Technical Reference(s): D1 – Auxiliary Feedwater

Proposed references to be provided to applicants during examination: none

Learning Objective: 8401 – Explain automatic actions associated with AFW system valves.

Question Source: Bank # DCPP A-0688
Modified Bank # ________
New ______

Question History: Last NRC Exam ____________
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: K/A: 061 K5.05 - Knowledge of the operational implications of the following concepts as the apply to the AFW: Feed line voiding and water hammer

1 A-0688 Points: 1.00 Multiple Choice

<QQ 34655(1410)><<Subsequent to a reactor trip from 100% power, a severe water hammer causes the aux feedwater line to Feed Lead 2-1 to break.

The break is just upstream of the check valve that isolates main feedwater from aux feedwater.

How do the following AFW LCVs respond to this malfunction?

- LCV-110 AFW PP 2-2 discharge to S/G 2-1
- LCV-111 AFW PP 2-2 discharge to S/G 2-2
- LCV-106 AFW PP 2-1 discharge to S/G 2-1>>

A. Low AFW PP 2-2 discharge pressure will result in throttling LCV-110 and LCV-111. LCV-106 will remain open until closed by operator action.

B. High AFW flow to S/G 2-1 will result in LCV-110 throttling. LCV-111 will open in an attempt to feed S/G 2-2. LCV-106 will remain open until closed by operator action.

C. Low pressure on the AFW line will send a close signal to LCV-110. LCV-111 will open in an attempt to feed S/G 2-2. LCV-106 will remain open until closed by operator action.

D. Low pressure on the AFW line to S/G 2-1 will send a signal to close LCV-110 and LCV-106. LCV-111 will be available to feed S/G 2-2.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8401	Explain automatic actions associated with AFW system valves

Reference Id: A-0688
Must appear: No
Status: Active
User Text: 8401.050614
User Number 1: 3.30
User Number 2: 3.60

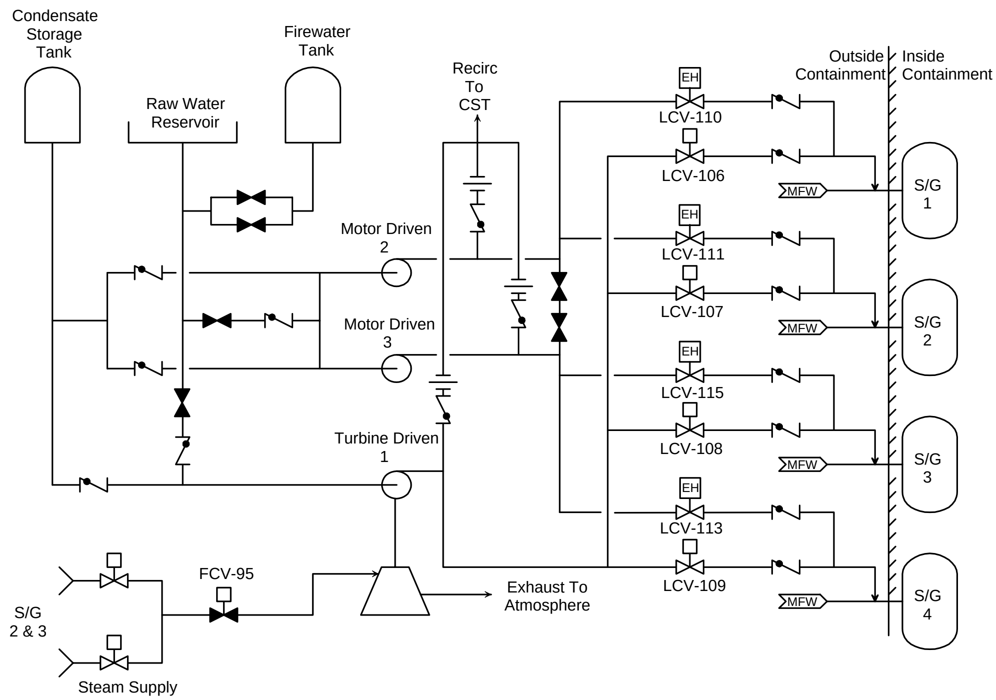
Difficulty:	3.00
Time to complete:	5
Topic:	AFW Pump response to low discharge pressure
Cross Reference:	LD-1, 61
Comment:	copied over from S-2836, que. needs validation 1/15/97 EAD1

Entered KA pedigree, increased difficulty level to 3.0, took to REVISE status 1/19/97.
Validated IAW TQ2.ID3. Taken to Review. 1/23/97 jpsj
Taken active following review, 1/25/97 JMH1
Checked as part of question review for biennial exam 12/22/98 MTC6

Basic Description, Continued

Basic flowpath The basic block and flow diagram of the AFW system is shown here.

Obj 2



D-1-1

TDAFW Pump LCVs (LCV-106, 107, 108, and 109)

Purpose
Obj 13

The purpose of the TDAFW pump LCVs is to control AFW flow to all four S/Gs and to provide isolation capability of a faulted S/G.

Location
Obj 8

LCV-106 and 107 are located in 115′ pipe rack area.
LCV-108 and 109 are located in the 115′ penetration area.

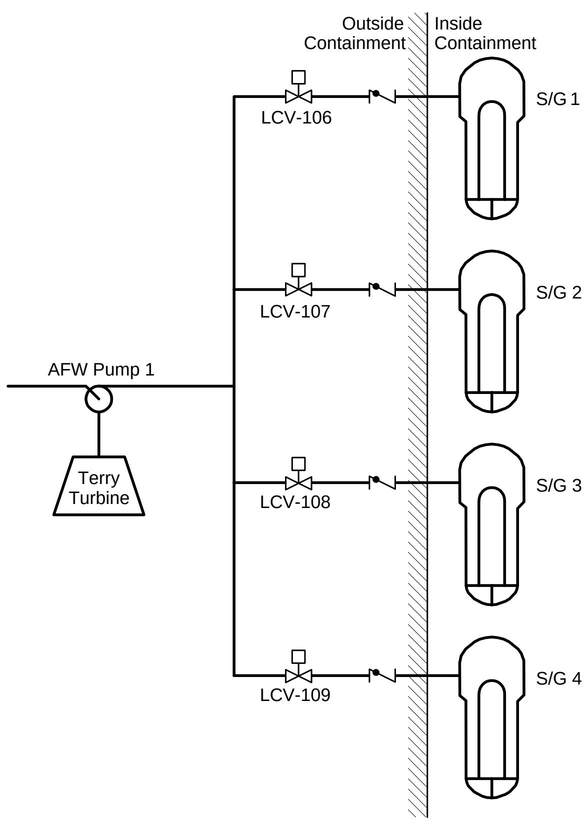
Power supplies
Obj 8

The power supply to all four LCVs is 480 V bus G.

- The bus is located on the 100' elevation of area H below Control Room.

Diagram

The following is a diagram of the flowpath from the TDAFW pump to the S/Gs:



AFW-12

Continued on next page

TDAFW Pump LCVs (LCV-106, 107, 108, and 109), Continued

Description

LCV-106 through -109 are motor operated limitorque valves. The valves are manually controlled and are normally left full open.

Controls
Obj 10, 18

- LCV-106 through -109 have control capability from the:
  - Control Room on VB-3
  - HSD Pnl
  - local handwheel
- The LCV transfer relays (described in HSD Panel STG A-8) have control capability from 480V Bus G.

In the Control Room:

Control	Operation
CLOSE/STOP/OPEN	3 position, spring return to STOP

At the HSD Pnl:

Control for valve:	Operation
CLOSE/STOP/OPEN	3 position, spring return to STOP
CONT RM/LOCAL	2 position, maintained

At 480 V Bus G:

Control for transfer relay	Operation
CUTIN/CUTOUT	2 position, maintained
TRIP/RESET	

Indications

Obj 9

The following indications are available for LCV-106 through 109 in the Control Room and the HSD Pnl:

LCV-106 through 109 control switch indications		
Indicating Lights	**Meaning**	**Normal Status**
Red	Valve is full OPEN.	ON
Green	Valve is full CLOSED.	OFF
NOTE: If BOTH Indicating lights are LIT simultaneously, then the valve is in an intermediate position.		

MDAFW Pump LCVs (LCV-110, 111, 113, and 115), Continued

Logic

The logic associated with the MDAFW pump LCVs Auto/Manual operation is described in the following tables:

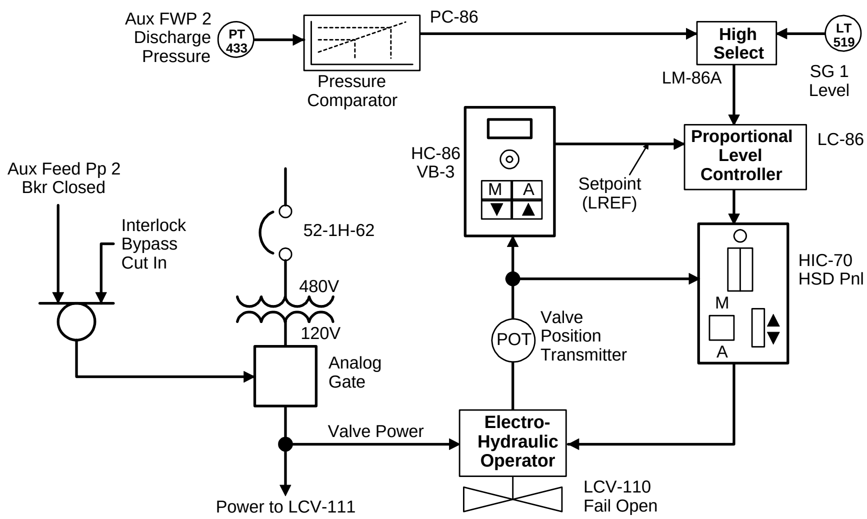
Obj 20, 27, 28

Logic input	Function
MDAFW pump discharge pressure PT-433, 434	• Provides actual pump discharge pressure to a pressure comparator. • Pressure comparator converts the pump discharge pressure signal to an equivalent inverse level signal (see drawing AFW-15 on page 2-44), and sends this level signal to the high select circuit described below. • This input is used as runout protection for the pump, and will cause the valve to close on low pump discharge pressure.
S/G Level LT-519, 529, 539, 549	• Provides actual S/G level input to a high select circuit. • High select allows the highest S/G level input to control valve. • S/G level signal input is from either: ♦ actual S/G level from LT-519 (529, 539, 549) or ♦ AFW pump discharge pressure comparator (converted to an inverse level signal) • Output of high select circuit is sent to a proportional level controller.
Valve Position Transmitter	• Provides feedback of actual valve position to both controllers and meters.

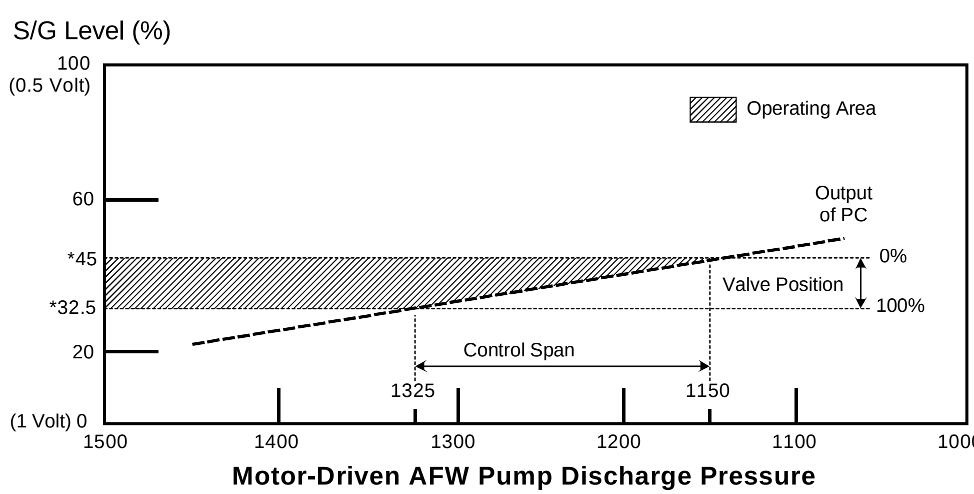
Continued on next page

MDAFW Pump LCVs (LCV-110, 111, 113, and 115), Continued

Logic Diagram The following diagrams depict the logic and interlocks associated with MDAFW pump LCVs:



AFW-14



AFW-15

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RO QUESTION 47

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	______
Group #	1	______
K/A #	062 K2.01	
Importance	3.3	3.4

<u>Proposed Question</u>:

With Unit 1 at 100% power, how does the 12 KV Bus D/E protection circuitry interface with the RCPs?

A. If 2 out of 3 relays on bus E sense < 54 Hz, a trip signal is sent to RCP 1-1 and 1-3 breakers.

B. If 2 out of 3 relays on either bus D or E sense < 54 Hz, a trip signal is sent to all 4 RCP breakers.

C. If 1 out of 2 relays on buses D and E sense < 70% voltage, a trip signal is sent to all 4 RCP breakers.

D. If 1 out 2 relays on bus D sense < 70% voltage, a trip signal is sent to RCP 1-2 and 1-4 breakers.

Proposed Answer:

A. If 2 out of 3 relays on bus E sense < 54 Hz, a trip signal is sent to RCP 1-1 and 1-3 breakers.

Explanation:

A correct, Bus E supplies RCPs 11 and 13.
B incorrect, only the RCPs on the bus get trip signal.
C incorrect, low voltage on both buses results in a reactor trip signal.
D incorrect, does not trip RCPs.

Technical Reference(s): J5 – 12 KV Electrical System

Proposed references to be provided to applicants during examination: none

Learning Objective: 6249 Analyze the logic associated with 12 kV Protection Relay indication lights

Question Source: Bank # DCPP S-44862
Modified Bank # ________
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
55.43

Comments:

K/A: 062 K2.01 - Knowledge of bus power supplies to the following: Major system loads

1 S-44862 Points: 1.00 Multiple Choice

How does the 12 KV Bus D/E protection circuitry interface with the RCPs? (Assume that the reactor is at 100% power)>>

A. If 2 out of 3 relays on bus E sense < 54 Hz, a trip signal is sent to RCP 1-1 and 1-3 breakers.

B. If 2 out of 3 relays on either bus D or E sense < 54 Hz, a trip signal is sent to all 4 RCP breakers.

C. If 1 out of 2 relays on buses D and E sense < 70% voltage, a trip signal is sent to all 4 RCP breakers.

D. If 1 out 2 relays on bus D sense < 70% voltage, a trip signal is sent to RCP 1-2 and 1-4 breakers.
>>

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

6249	Analyze the logic associated with 12 kV Protection Relay indication lights

Reference Id: S-44862
Must appear: No
Status: Active
User Text: 6249.070621
User Number 1: 2.60
User Number 2: 3.10
Difficulty: 3.00
Time to complete: 2
Topic: Analyze the logic associated with 12 kV Protection Relay ind
Cross Reference: LJ-5
Comment:

Power supply

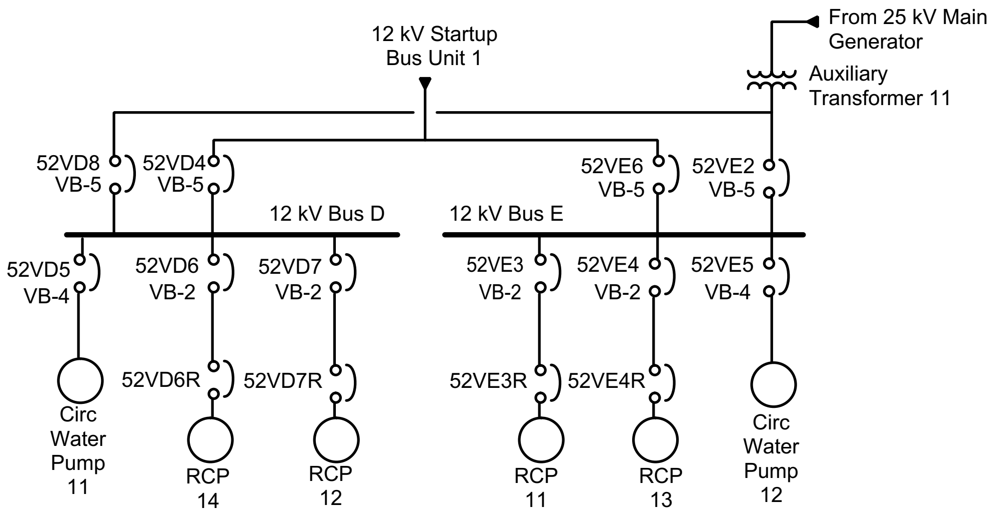
The 12 kV Bus D and E are supplied through the feeder breakers listed below.

12 kV bus D and E supplies:

Load	Startup bus feeder breakers	Auxiliary transformer 1-1 (2-1)
Bus D	52VD4 (52VD6)	52VD8 (52VD2)
Bus E	52VE6 (52VE4)	52VE2 (52VE8)

Unit 1 12 kV bus D and E diagram

The 12 kV bus D and E diagram is shown below for Unit 1.



12KV-08

Continued on next page

12 kV Buses D and E Reactor Protection

Relays that activate Bus D and E Reactor Protection are described below.

Relay and Parameter	Protection

Under voltage condition on 12 kV Buses D and E
 	
27VDR1
27VDR2
27VER1
27VER2
	Each 12 kV bus that supplies the RCPs is equipped with two undervoltage (UV) sensing relays.
	• Should a UV relay associated with one bus sense that pump supply voltage has dropped to [70%] of normal (8050 V), a UV signal is produced for that bus.
	  • A UV signal from each bus is required to produce a reactor trip.
	  • A time delay pickup relay is used to prevent spurious trips from short term voltage perturbations.
	  • This trip is blocked below 10% reactor power.
	• If one of two undervoltage trip relays on each bus senses an undervoltage condition (undervoltage on both Buses D and E), the turbine driven auxiliary feedwater pump starts automatically.

Under frequency (81 relay) condition on 12 kV Buses D and E	Each 12 kV bus that supplies the RCPs is equipped with three underfrequency (UF) sensing relays. • Should two of three UF relays associated with a bus sense that pump supply frequency has dropped to [54 Hz], a UF signal is produced for that bus. • A UF signal from either bus will produce a reactor trip. • A UF signal from a bus will trip the RCP breakers associated with the bus. • A time delay pickup relay is used to prevent spurious trips from short term perturbations. • This trip is blocked below 10% reactor power.

Note:
- If relays associated with the 12 kV Bus D are being tested, the “12 kV BUS D RCP UV & UF RLY ON TEST” annunciator on PK19-08 is actuated.
- For the 12 kV Bus E relays in test, the “12 kV BUS E RCP UV & UF RLY ON TEST” annunciator PK19-13 is actuated.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	1	______
	K/A #	063 K3.01	
	Importance	3.7*	4.1

Proposed Question:

The Unit 1 CO is unable to shutdown Diesel Generator 1-1 due to a total loss of DC power.

You are instructed to shutdown the Diesel locally.

This is accomplished by performing which of the following actions?

A. Operating the local trip lever on the overspeed trip device.

B. Taking local control of the Diesel and going to STOP on the local control switch.

C. Pushing the Emergency shutdown pushbutton located just outside of the D/G room.

D. Placing the local mode control switch in TEST and depressing the local Voltage Shutdown pushbutton.

Proposed Answer:

A. Operating the local trip lever on the overspeed trip device.

Explanation:

A correct, Per OP J-6B:IV Diesel Generator 1-1, Manual Operations
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 north west corner of the engine, forward of the fuel oil filters. (this is the overspeed trip device)
A incorrect, air motors will not cause the diesel to stop.
B, C and D incorrect, are ineffective without control power.

Technical Reference(s): OP J-6B:IV Diesel Generator 1-1, Manual Operations
Drawings 437579 and 437580 (separate PDF file)

Proposed references to be provided to applicants during examination: none

Learning Objective: 69182 - Describe Diesel Generators instrumentation and controls, including symptoms of failure modes.

Question Source: Bank # DCPP C-61753
Modified Bank #
New

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 41.10
55.43

Comments:

K/A: 063 K3.01 – Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: ED/G

5. PRECAUTIONS AND LIMITATIONS

5.1 Paralleling the diesel to Startup power defeats both 2nd level UV relays. Refer to Tech Spec 3.3.5.

5.2 If the D/G is paralleled to the auxiliary transformer and the main unit trips, the aux bus feeder breaker will automatically open and the D/G will carry the bus load. Verify that the bus feeder breaker opened and place the D/G MODE SEL switch into AUTO.

5.3 If the D/G is paralleled to startup power and a loss of offsite power occurs, the startup feeder breaker will not automatically open. The D/G will attempt to supply power to loads connected to the grid. Open the startup feeder breaker or verify the D/G breaker tripped. Place the D/G MODE SEL switch into AUTO if the feeder breaker was opened.

5.4 When paralleling a D/G to any off site power source or the D/G MODE SEL switch is in MANUAL and the D/G is not running, declare the D/G inoperable. In MODES 1, 2, 3, and 4, perform the actions required for Tech Spec 3.8.1. In MODES 5, 6, and irradiated fuel movement, perform the actions required for Tech Spec 3.8.2.

5.5 The D/G should not be operated for an extended period of time below 0.65 MW.
- 5.5.1 If a D/G is operated < 0.65 MW for < one hour, no action is necessary.
- 5.5.2 If a D/G is operated < 0.65 MW for ³ one hour but < 10 hours, then the D/G should be operated ³ 1.30 MW for ³ one hour.
- 5.5.3 If the D/G is to be operated for longer than 10 hours £ 0.65 MW, the D/G should be loaded to ³ 1.3 MW for > one hour at the end of each 10 hour period.

NOTE: Operation of D/G < 0.65 MW for an extended period of time can expose the engine to undesirable conditions which may be detrimental to engine performance and component life.

5.5 Do not operate the D/G below rated speed with the field energized. Excessive field currents and rotor temperatures may occur.

5.6 When paralleling a D/G, pick up load (0.50 MW) as soon as possible (< 15 seconds) after the breaker is closed. This will prevent the D/G breaker from tripping on directional power (DIR PWR).

5.7 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.

5.9 The fuel oil pressure should increase to above 40 PSIG within 60 seconds of engine start. Gauge response is about 15 seconds.

5.10 Do not violate the following limits during normal steady -state operation:

5.10.1 Maximum continuous generator current is 451 amperes.
5.10.2 Maximum stator temperature is 240°F.
5.10.3 Minimum lube oil pressure is 60 PSIG.
5.10.4 Maximum lube oil temperature is 195°F.
5.10.5 Power factor: 1.0 to 0.8 lag.
5.10.6 Load: 2.60 MW at 0.8.

5.11 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 north west corner of the engine, forward of the fuel oil filters.

5.12 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).

1 C-61753 Points: 1.00 Multiple Choice

The Unit 1 CO is unable to shutdown Diesel Generator 1-2 due to a total loss of D.C. power. He instructs you to shutdown the Diesel locally. How can that be accomplished?

A. By operating the local trip lever on the overspeed trip device.

B. By isolating all starting air motors.

C. By pushing the Emergency shutdown pushbutton located just outside of the D/G room.

D. By taking local control of the Diesel and going to STOP on the local control switch.

>>

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

69182	Describe Diesel Generators instrumentation and controls, including symptoms of failure modes.

Reference Id: C-61753
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 2.00
Time to complete: 3
Topic: D/G shutdown with no D.C control power
Cross Reference:
Comment: created for R011C12, jpl1, 5/31/01.

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	______
Group #	1	______
K/A #	064 A3.10	
Importance	2.8	2.8*

Proposed Question:

Diesel DG 13 is carrying 4160V bus F following a loss of normal power to the bus and SI actuation.

The Operator goes to RAISE on the diesel's speed control for three (3) seconds.

Which of the following represents the effect this action has on indicated bus voltage, frequency, Mwe and Kvar?

	VOLTS	FREQ	Mwe	Kvar
A.	UP	SAME	SAME	SAME
B.	SAME	SAME	UP	UP
C.	UP	UP	UP	UP
D.	SAME	SAME	SAME	SAME

Proposed Answer:

D. SAME SAME SAME SAME

Explanation:

A incorrect, this is true if volts is raised (previous answer on question)
B incorrect, load is set by sequencer
C incorrect, this is if in DROOP mode and volts are raised as well.
D correct, Volts set by voltage control switch, freq set at 60 hz, load determined by bus loading.

Technical Reference(s): J6B – Diesel Generator System

Proposed references to be provided to applicants during examination: none

Learning Objective: 4158 - Explain the Isochronous/Droop modes of operation.

Question Source: Bank #
Modified Bank # DCPP P-1228
New ______

Question History: Last NRC Exam DCPP SRO 2/94

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

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments:

K/A: 064 A3.10 - Ability to monitor automatic operation of the ED/G system, including:
Function of ED/G megawatt load controller

1 P-1228 Points: 1.00 Multiple Choice

Given the following:

- Diesel DG 13 is carrying 4160V Bus F following a loss of normal power to the bus. (Normal inputs open and NO other loads added).
- A Safety Injection actuation has loaded the bus.
- The operator goes to RAISE on the diesel's speed AND voltage controls for three (3) seconds.

Which of the following best represents the results of this action?

	VOLTS	FREQ	Mwe	Kvar
A.	UP	SAME	SAME	SAME
B.	SAME	SAME	UP	UP
C.	UP	UP	UP	UP
D.	UP	UP	SAME	SAME

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

6214	Demonstrate the ability to diagnose AC Distribution system malfunctions

Reference Id: P-1228
Must appear: No
Status: Active
User Text: 6214.070525
User Number 1: 2.90
User Number 2: 3.70
Difficulty: 2.00
Time to complete: 3
Topic: Predict DG op parameters for single source supply
Cross Reference:
Comment: DCPP NRC SRO Exam 2/16/94

Normal Operations, Continued

Normal Operations, Continued

Diesel load control
Obj 33

The tables below list how diesel parameters are controlled when the diesel generator is operating isolated, and when it is operating paralleled:

If the D/G is carrying the 4 kV bus by itself (i.e., if D/G is not paralleled):

Parameter	Control
Control Mode	ISOC
Frequency	Controlled to maintain 60 hertz
Voltage	Raised or lowered by the D/G Voltage Control Switch
MW	Determined by bus loading
MVAR	Determined by bus loading

If the D/G is paralleled to the bus (i.e., sharing load):

Parameter	Control
Control Mode	DROOP
Frequency	Determined by Aux. or Startup power (i.e., 60 Hz).
Voltage	Determined by Aux. or Startup power.
MW	Determined by torque applied to the generator by the diesel engine; raised or lowered by the D/G Speed Control switch.
MVAR	Determined by diesel generator excitation; raised or lowered by the D/G Voltage Control switch.

Shifting control

For normal steady-state diesel generator operation, the diesel will always be in:
- Droop control when paralleled, or
- ISOC control when operating isolated.

However, for certain lineup changes, diesel generator control may *momentarily* be in the opposite configuration. Examples:
- When shifting a diesel from paralleled to isolated operation in OP J-6B:IV:
  - ♦ the diesel is initially in Droop control,
  - ♦ the Aux. feeder breaker or Startup feeder breaker is opened, placing the entire bus load on the diesel,
  - ♦ the diesel is placed in AUTO (i.e., shifted from Droop to ISOC control).
- When shifting diesel from isolated operation to paralleled operation in STP M-15, the diesel is shifted to MANUAL (i.e., Droop) prior to paralleling.
- When diesel is isolated in Droop control, frequency drop could occur: [KD21]
  - ♦ If bus frequency lowers, the speed of all the pumps on the bus lowers
  - ♦ If pump speed lowers, pump flow rate lowers (for example, if bus frequency lowers, RHR flow rate would lower)
  - ♦ Operator should <u>immediately check frequency</u> when shifting control modes.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	064 A3.03	
	Importance	3.4	3.3

Proposed Question:

Unit 1 is at 100% power.

The crew is preparing to parallel 11 EDG to its vital bus.

The operator is preparing to close the generator breaker.

To properly close the breaker, what should be the synchroscope indication when the breaker is closed?

A. Rotating slowly in the SLOW direction, lights OFF.

B. Rotating slowly in the SLOW direction, lights FULL BRIGHT.

C. Rotating slowly in the FAST direction, lights OFF.

D. Rotating slowly in the FAST direction, lights FULL BRIGHT.

Proposed Answer:

C. Rotating slowly in the FAST direction, lights OFF.

Explanation:

C correct: OP J-6B:IV, Diesel Generator 1-1, Manual Operations states:
6.2.4 Verify synchroscope working

a. **Lights OFF** at the 12 o'clock position.

b. Lights FULL BRIGHT at the 6 o'clock position.

6.2.5 Adjust engine speed up and down to verify the manual governor is in control. Set engine speed so the **synchroscope is turning slowly in the clockwise (FAST)** direction. This will allow the diesel to pick up load when paralleled to the bus.

6.2.7 When the synchroscope pointer is slightly before the 12 o'clock position, turn generator breaker 52-HH-7 control switch to the CLOSE position.

Technical Reference(s):
OP J-6B:IV, Diesel Generator 1-1, Manual Operations

Proposed references to be provided to applicants during examination: none

Learning Objective:
4158 - Explain the Isochronous/Droop modes of operation.

Question Source: Bank #
Modified Bank # ________
New X

Question History: Last NRC Exam ______________
Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

K/A: 064 A3.03 - Ability to monitor automatic operation of the ED/G system, including: Indicating lights, meters, and recorders

6. INSTRUCTIONS

6.1 Manually Start/Stop Diesel From Control Room.

6.1.1 Verify that cardox fire protection to Diesel Generator 1-1 is not aborted (D/G Room 1-1 Abort Valve Close alarm not in) using the Fire Protection Computer. (Refer to OP K-2C)

6.1.2 Check main annunciator windows associated with the Diesel Generator not in alarm. At minimum, consider the effects of each alarm on diesel operation before proceeding.

6.1.3 Dispatch an operator to engine room with at least these instructions:

a. Make cardox available, if aborted.

b. Establish communication with Control Room, as needed.

6.1.4 At Diesel 1-1 Control Panel (VB-4)

a. Place the D/G 1-1 MODE SEL switch in the MANUAL position.

b. CUT-IN or verify cut-in the Generator Protective Relays.

c. START the Diesel by momentarily turning the start/stop switch to the start position.

d. Verify diesel cranks up to speed of about 900 RPM in 10 seconds or less and 4160V (120V on indicator) in 13 seconds or less.

e. Adjust engine speed to 900 RPM with governor speed control switch if needed.

f. Adjust generator output voltage to 120 volts indicated using the voltage control switch, if needed.

6.1.5 To Shutdown diesel without paralleling to a Bus perform the following:

a. Adjust the DG speed to ~60 Hz.

b. Adjust the DG voltage to ~119VAC.

c. Shutdown the DG.

d. Verify Generator Protective Relays are Cut Out.

e. When DG speed reaches zero, place the DG 11 Mode Sel switch to AUTO.

f. Remove the poly bottles at the DFO leak off lines per STP M-9A, if installed.

6.1.6 Inform SFM of the Diesel Generator status.

6.2 Paralleling/Separating Diesel To/From Bus H

6.2.1 Verify steps 6.1.4a and 6.1.4b have been performed.

6.2.2 Check diesel generator output voltage on each phase. Otherwise, when synchroscope is turned ON (next step) the voltmeter will lock on to phase C.

6.2.3 Cut in the D/G 1-1 FEEDER SYNC Switch.

6.2.4 Verify synchroscope working

a. Lights OFF at the 12 o'clock position.

b. Lights FULL BRIGHT at the 6 o'clock position.

6.2.5 Adjust engine speed up and down to verify the manual governor is in control. Set engine speed so the synchroscope is turning slowly in the clockwise (FAST) direction. This will allow the diesel to pick up load when paralleled to the bus.

NOTE: VARs should be slightly out, ~0.3 MVARS OUT, while the DG is paralleled in the Droop mode.

6.2.6 Adjust generator voltage to within ±2 volts of bus voltage.

6.2.7 When the synchroscope pointer is slightly before the 12 o'clock position, turn generator breaker 52-HH-7 control switch to the CLOSE position.

RO QUESTION 51

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	_____
	Group #	2	_____
	K/A #	073 K1.01	
	Importance	3.6	3.9

Proposed Question:

Unit 2 is at full power. A small steam generator tube leak is causing steam line radiation monitor RM-73 to read 1000 cpm.

If the monitor is functioning properly, what should happen to the indication if power is reduced to 50%?

A. Indication should decrease due to the decrease in N-16 production.

B. Indication should decrease due to the decrease in iodine production.

C. Indication should remain the same due to the continued tube leakage.

D. Indication should increase because there is less steam flow but the same amount of radiation.

Proposed Answer:

A. Indication should decrease due to the decrease in N-16 production.

Explanation:

The steam line radiation monitors detect N-16 from the tube leakage. Once the unit is shutdown, N-16 production ceases and the indication will decrease.

Technical Reference(s): G4A – Radiation Monitoring, SOE-93-001

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

Learning Objective: 8485 Explain the conditions that effect Radiation Monitoring system radiation monitor indications

Question Source: Bank #
Modified Bank # S-1207
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 41.11
55.43 _____

Comments: K/A: 073 K1.01 - Knowledge of the physical connections and/or cause effect relationships between the PRM system and the following systems:Those systems served by PRMs

1 S-1207 Points: 1.00 Short Answer

Steam line Rad monitor RM-73 was reading 1000 cpm with reactor power at 100%. During the rampdown, to take the unit off-line, the reactor tripped. RM-73 is now reading normal background (equal to other steam line Rad Monitors). Why did the reading decrease after the reactor trip?

Answer:

Prior to trip the rad monitor was detecting N-16 gamma from reactor coolant in-leakage. After reactor trip there is effectively no production of N-16 (to decay or be dectected by steam line Rad Monitors.

ASSOCIATED INFORMATION:

Associated objective(s):

8485	Explain the conditions that effect Radiation Monitoring system radiation monitor indications
69298	Explain the basic principles of operation for the Radiation Monitoring System.
69298	Explain the basic principles of operation for the Radiation Monitoring System.
5747	State RADIATION MONITORING system use for diagnosing a SGTR

Reference Id: S-1207
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 0.00
Time to complete: 3
Topic: Steam Line Rad Monitor response after a trip
Cross Reference: LG-4
Comment:

Industry Events

SOE-93-001

Palo Verde 2 (March 14, 1993)

Event

A S/G tube ruptured causing a leak of approximately 240 gpm. Plant operators used the emergency operating procedures to diagnose and mitigate the event but twice failed to diagnose a tube rupture. The radiation monitors that would have led to that diagnosis were not in an alarm status when the applicable step in the procedure was reached. As a consequence, the operators did not isolate the affected S/G until almost 3 hours after the rupture occurred.

Discussion

The Main Steam Line Monitors are sensitive to N^{16} gamma radiation which decays off quickly after shutdown. By the time the procedure called for monitoring these instruments, the readings were down to near normal, inferring no S/G tube rupture.

Corrective Action

The equivalent monitors RM-71 through 74 are recorded on the PPC recorders on VB-2 to identify trends or spikes that might not have been seen during the initial phases of the event.

SGTR Rapid Identification Blowdown Sample Guide: Chemistry analysis procedure CAP AP-1, “Prompt Steam Generator Leak Identification Procedure” provides a method of promptly identifying any leaking/ruptured S/Gs within 30 min. or if the leak rate and/or RCS activity is sufficiently high, the affected S/G(s) may be identified more quickly by using a frisker to determine the highest count rate(s).

Main Steam Line Radiation Monitor Response to a DBA LOCA: Radiation shine from containment will alarm all four (4) main steam line radiation monitors during a DBA LOCA without a SGTR in progress. A knowledge item has been added to six (6) EOP background documents to exclude entering EOP E-3 if all four (4) main steam line radiation monitors exhibit the same order of magnitude response. The six emergency procedures affected are EOP E-0, E-1, E-1.2, E-2, ECA-2.1, and FR-H.3.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	073 A4.02	
	Importance	3.7	3.7

Proposed Question:

Which of the following is the minimum action(s) necessary to defeat CVI actuation due to a high radiation signal from Containment Exhaust Radiation Monitor RM-44A?

A. Place the CVI selector switch in MODE 6.

B. Place the ENABLE/BYPASS for RM-44A in BYPASS.

C. Place the ENABLE/BYPASS for RM-44A in BYPASS and CVI selector switch in MODE 6.

D. Place the CVI selector switch in MODE 6 and open breaker PJRM-11 on 1G.

Proposed Answer:

B. Placing the ENABLE/BYPASS for RM-44A in BYPASS.

Explanation:

A incorrect, CVI selector defeats SSPS input.
B correct, in BYPASS, CVI due to high radiation is defeated.
C incorrect, CVI selector defeats SSPS input.
D incorrect, not necessary to de energize the monitor.

Technical Reference(s): G4B – Digital Radiation Monitoring System

Proposed references to be provided to applicants during examination: none

Learning Objective: 3279 - Explain the operation of Digital Radiation Monitoring System controls in control room.

Question Source:
New X

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 41.11
55.43 _____

Comments:

K/A: 073 A4.02 - Ability to manually operate and/or monitor in the control room:
Radiation monitoring system control panel

Control

The controls associated with the Containment Exhaust Radiation Monitors are shown in the tables below.

- For controls that are common to all LRP and RDU refer to the section LRP and RDU in this STG.

RNRMS1 and 2

Control	Operation
CVI RE-44A	
ENABLE/BYPASS	2 position maintained keyswitch
CVI RE-44B	
ENABLE/BYPASS	2 position maintained keyswitch

Logic

The logic associated with Containment Exhaust Radiation Monitors are shown in the table below.

If the CVI RM-44A switch is in ...	Then ...
ENABLE	the automatic actuation of CVI is enabled from a high alarm on RM-44A.
BYPASS	the automatic actuation of CVI is blocked from a high alarm on RM-44A.
If the CVI RM-44B switch is in ...	**Then ...**
ENABLE	the automatic actuation of CVI is enabled from a high alarm on RM-44B.
BYPASS	the automatic actuation of CVI is blocked from a high alarm on RM-44B.

Continued on next page

Containment Ventilation Isolation Actuation, Continued

Logic (continued)

<table>
<tr><th>IF CVI Mode 6 selector switch is in NORMAL and...</th><th>Then</th></tr>
<tr><td>CVI Train A (B) is RESET</td><td>the Train A (B) CVI signal is reset as follows.
• the manual Phase A or B actuation signals go through a retentive memory unit and will seal in until reset, the CVI cannot be reset until the initiating signal has been reset.
• the SI and RM-44A/B actuation signals go through retentive memory with manual reset units and will also seal in until reset.
♦ The signal can be reset even though the initiating signal is still present.
♦ NOTE: If the unit is reset with one rad monitor in alarm, the signal from the other rad monitor is disabled until both clear.
• PK02-06, CVI will clear.
• Red light for Train A (B) above Monitor Light Box B goes out.</td></tr>
</table>

<table>
<tr><th>IF CVI Mode 6 selector switch is in MODE 6 and...</th><th>Then</th></tr>
<tr><td>RM-44A or B alarm on high radiation</td><td>CVI will be actuated.
• PK02-06, CVI will alarm.
• Red light for Train A (B) above Monitor Light Box B is illuminated.</td></tr>
<tr><td>Phase A CI is manually actuated</td><td rowspan="3">CVI will not be actuated. (TS violation if in Modes 1 through 4)</td></tr>
<tr><td>Phase B CI is manually actuated</td></tr>
<tr><td>any SI signal is generated</td></tr>
</table>

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	076 K3.05	
	Importance	3.0*	3.2*

Proposed Question:

PLANT CONDITIONS:

- RHR cooldown initiated using RHR pump 1-1 and 1-1 heat exchanger
- RHR flow is 3000 gpm
- A cooldown rate of 30 ºF/hr has been established

Which of the following failures will cause the cooldown rate to decrease?

A. Loss of Aux Saltwater.

B. Loss of power to FCV-641A (RHR Pump Recirc valve).

C. Loss of control air to HCV-133 (RHR HX Outlet to CVCS control valve).

D. Maximum control air signal is applied to HCV-670 (RHR HX Bypass FCV).

Proposed Answer:

A. Loss of Aux Saltwater.

Explanation:
A, correct, results in less cooling to the CCW heat exchanger which will lead to less cooling for RHR heat exchanger.
B incorrect, loss of power to the recirc valve (MOV) will result in the valve maintaining its current position.
C incorrect, valve fails closed on loss of air but will have no effect on cooldown rate.
D incorrect, valve will close, all flow thru the heat exchanger, cooldown will increase.

Technical Reference(s): B-2, RHR, pages 1-5, 2-30, 2-38

Proposed references to be provided to applicants during examination: None

Learning Objective: 7009 – Analyze the control logic for RHR system valves.
8089 - Analyze control logic for CCW components.

Question Source:

Modified Bank # INPO 20643

Question History: Last NRC Exam Kewaunee 09/2002

Question Cognitive Level:

Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.4

55.43 _____

Comments:

076 K3.05 - Knowledge of the effect that a loss or malfunction of the SWS will have on the following: RHR components, controls, sensors, indicators, and alarms, including rad monitors

ID: **20643** Given the following plant conditions:

- Reactor Coolant System temperature is 320F.
- Reactor Coolant System pressure is 370 psig.
- RHR cooldown operations has been established with both RHR pumps and heat exchangers in service.
- A cooldown rate of 80F/hr has been established.

Which one of the following failures will cause the cooldown rate to increase?

Ans Loss of control air to RH-626 (RHR HX Bypass FCV).

D1 Loss of power to CC-738A (HX-11A RHR HX-Shell Side Inlet Valve).

D2 Maximum control air signal to RH-624 (HX-11A RHR HX Outlet FCV).

D3 The bellows in FT-626 (RHR System Return Line Flow) fails by rupturing.

AbbrevLocName	ExamDate	Vendor	Type	ExamType	Cog Level	ExamLevel	RefMaterial	ParentId
Point Beach 1	2/2/2002	WEC	PWR	ILO		R		

QuestionComment	
Distract1Comment	
Distract2Comment	
Distract3Comment	

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
..005.a1.01			005	a1	01	

Basic Description

Purpose of the RHR System
Obj 1

The purpose of the Residual Heat Removal (RHR) system is to transfer decay (residual) heat and sensible heat under normal and emergency conditions from the Reactor Coolant System (RCS) (and the core) to the Component Cooling Water (CCW) system, where it is transferred to the Auxiliary Salt Water (ASW) system and then to the ultimate heat sink, the Pacific Ocean. Other purposes are as follows:

- Inject into the RCS during a Loss of Coolant Accident (LOCA) (discussed in the ECCS chapter)
- Prevent boron stratification in the RCS
- Transfer water between the Refueling Water Storage Tank (RWST) and the refueling cavity
- Transfer water to the Chemical and Volume Control System (CVCS) for cleanup and pressure control
- Transfer water from the Containment Recirculation Sump to the Containment Spray Headers for continuing post-accident spray after the Containment Spray System has emptied the RWST.

Basic Description
Obj 2

The RHR system consists of:

Two parallel, redundant flow paths, each consisting of:
- One RHR pump
- One RHR to CCW Heat Exchanger
- One flow control valve to control heat exchanger flow rate
- Each path physically and electrically separated

Each flowpath can take suction from:
- The RWST
- Loop 4 Hot Leg
- Its own suction on the Containment Recirculation Sump

Each flowpath can discharge to:
- Two cold legs
- Hot legs 1 and 2
- A Containment Spray Header
- Safety Injection (SI) Pumps or Centrifugal Charging Pumps (CCP)
- The RWST
- The CVCS

Continued on next page

RHR Flow Control Valves, HCV-637, 638, 670

Purpose
Obj 7

The purpose of HCV-637, 638, and 670 is to control flow through or around the heat exchangers for RCS temperature control.

Location
Obj 8, 11

The RHR Flow Control Valves are located as follows:

- HCV-637 and 638 are on the 85′ elevation in the penetration area.
- HCV-670 is located on the 100′ elevation in the penetration area.

Physical description

HCV-637, 638, and bypass valve 670 are air operated, fail open ball valves.

- HCV-637 and 638 are used to control flow through the heat exchanger.
- HCV-670 is used to adjust bypass flow to maintain total flow constant.

HCV-637 and 638 have mechanical full open stops that ensure:

- the RHR system is capable of delivering the required minimum flow rate, and
- to prevent RHR pump runout.

The 100% open stop for each valve is set via STP V-4A, Functional Test of RHR Check Valves, to get a flow rate of ≥ 3976 gpm and < 4319 gpm.

Control
Obj 20

HCV-637, 638, and 670 have control capability from the:

- Control Room on VB1, and
- local jacking mechanism.

Control for the valve	Operation
Hand controller	0-100%

Indication

The following indications are available for HCV-637, 638, and 670.

HCV-637, 638 and 670 controller		
Indication	**Meaning**	**Normal Status**
HCV-637	Indicates the controller demand to the valve (0% closed, 100% open)	100%
HCV-638	Indicates the controller demand to the valve (0% closed, 100% open)	100%
HCV-670	Indicates the controller demand to the valve (0% closed, 100% open)	0%

HCV-637, 638 Monitor Light Box A on VB1		
Indicating Light	**Meaning**	**Normal Status**
White	Valve not fully OPEN	OFF

RHR HX Outlet to CVCS Control Valve, HCV-133

Purpose

The purpose of HCV-133 is to control flow to the CVCS letdown line for RCS chemistry control during RHR cooldown operations, and to aid in pressure control during solid plant operation.

Physical description

The HCV-133 is an air operated, fail closed valve used to letdown flow from the RHR system to the CVCS system (upstream of the letdown heat exchanger) when RCS pressure is too low to provide sufficient flow through the normal letdown system.

Control

HCV-133 has control capability from the Control Room on VB2.

Control for the valve	Operation
Hand controller	0-100%

Indication

The following indication is available for HCV-133.

HCV-133 controller		
Indication	**Meaning**	**Normal Status**
HCV-133	Indicates the controller demand to the valve (0% closed, 100% open)	0%

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	078 K4.01	
	Importance	2.7	2.9

Proposed Question:

Unit 1 was at 100% power when a total loss of instrument air occurred.

What must be done so that the 10% steam dumps can be controlled, and what is the source of control pressure to operate the valves?

A. Cut in toggle switch on VB-3, control is via backup nitrogen.

B. Leave Hagan controller in auto, control is via backup nitrogen.

C. Leave Hagan controller in auto, control is via backup air bottles.

D. Place Hagan controller in manual, control is via backup air bottles.

Proposed Answer:

B. Leave Hagan controller in auto, control is via backup nitrogen.

Explanation:
A incorrect, cutout switch associated with backup air.
B correct, nitrogen will operate the valves as pressure decreases below 85 psig.
C incorrect, backup air bottles operate if pressure decreases below 80 psig (once nitrogen depleted).
D incorrect, backup air is placed in service by operating the cutout switch on VB3. Even if nitrogen pressure went to 0, no steam dumps could be operated without operating the cutout switch.

Technical Reference(s): C2B – Steam Dump System page 2.1-14

Proposed references to be provided to applicants during examination: none

Learning Objective: 8042 - Explain physical connections and/or cause effect relationships between the Steam Dump System and other systems.

Question Source: Bank # INPO 22521

Question History: Last NRC Exam DCPP 10/2002

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.7

55.43 _____

Comments: K/A: 078 K4.01 - Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control

ID: 22521 Unit 1 was at 100% power when a total loss of instrument air occurred.

What must be done so that the 10% steam dumps can be controlled, and what is the source of control pressure to operate the valves?

Ans Leave Hagan controller in auto, control is via backup nitrogen.

D1 Leave Hagan controller in auto, control is via backup air bottles.

D2 Place Hagan controller in manual, control is via backup air bottles.

D3 Cut in toggle switch on VB-3, control is via backup nitrogen.

AbbrevLocName	ExamDate	Vendor	Type	ExamType	Cog Level	ExamLevel	RefMaterial	ParentId
Diablo Canyon 1	10/1/2002	WEC	PWR	ILO	2	S		

QuestionComment	
Distract1Comment	
Distract2Comment	
Distract3Comment	

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
..065.AA2.07			065	AA2	07	

Air control valves (continued)

Direct Control solenoid valves*

Key number	Solenoid status	Position of ports
8	Solenoid de-energized (switch not in open position)	valve is closed
	Solenoid energized (switch selected to open)	valve is open
9	Solenoid de-energized (switch not in close position)	valve is closed
	Solenoid energized (switch selected to close)	valve is open

* When direct control is enabled the operator may control valve position by applying air to the actuator through (8) or venting air from the actuator through (9).

Backup actuator supplies
Obj 15

Valve actuation pressure is normally from instrument air (supplied at 100 psig). backup sources are shown in the figure above and in the table below.

Source	Supply pressure	When in service
Nitrogen	85 psig	If instrument air pressure drops to less than 85 psig the pressure from the nitrogen system will seat the instrument air check valve and begin to supply the actuator.
Backup air bottles*	80 psig	The backup air bottles are not available unless the supply is cut in on VB-3. Note that when the backup air supply is cut in it is the only source of control air to the valve and is controlled by the operator manually on VB-3.

* When backup air bottles are cutin the air bottle supply can operate the Steam Dump valve through ten complete cycles for up to 6 hours at the minimum pressure, (260psig) normally pressure is 1000 to 2250 psig.

T.S.LCO
Obj 28

The back-up air bottle for each 10% Steam Dump valve must be greater than 260 psig for the Steam Dump valve to be Operable.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	103 A2.03	
	Importance	3.5*	3.8*

Proposed Question:

You are performing the actions of Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status.

At step 2, Verify Containment Isolation Phase A, you note that one Red light is lit, the other is out and approximately half the White lights are lit.

Which of the following describes the probable failure and action necessary to address the failure?

A. The train of Phase A with the Red light lit failed to actuate. Manually actuate Containment Isolation Phase A or manually reposition the components corresponding to the lights that are out.

B. The train of Phase A with the Red light lit failed to actuate. Manually actuate Containment Isolation Phase A or manually reposition the components corresponding to the lights that are lit.

C. The train of Phase A with the Red light out failed to actuate. Manually actuate Containment Isolation Phase A or manually reposition the components corresponding to the lights that are out.

D. The train of Phase A with the Red light out failed to actuate. Manually actuate Containment Isolation Phase A or manually reposition the components corresponding to the lights that are lit.

Proposed Answer:

D. The train of Phase A with the Red light out failed to actuate. Manually actuate Containment Isolation Phase A or manually reposition the components corresponding to the lights that are lit.

Explanation:

A and B incorrect, the Red light indicates that train has actuated, no action required for that train.
C incorrect, the white lights that are out correspond to the train that actuated.
D correct, the light out indicates that train has failed to actuate. The action is to actuate Containment Phase A or manually close valves with white lights lit.

Technical Reference(s):
E-0, Appendix E
B6A, ESFAS pages 2.2-17

Proposed references to be provided to applicants during examination: none

Learning Objective: 4006 – Explain the conditions that affect Monitor Light Box lights.

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 41.7
55.43

Comments:

K/A: 103 A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation

APPENDIX E
ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. CHECK Main Generator Tripped:	
a. PK14-01, Unit Trip - ON	a. Manually initiate a Main Unit Trip.
b. Verify 500KV Brkrs - Both Open o Green lights - ON OR o Turbine speed - decreasing	b. Notify 500KV Switchyard to locally open the failed 500KV Brkrs.
c. Check Exciter Field Breaker - OPEN	c. WHEN Both 500KV Brkrs are open, THEN Open the Exciter Field Brkr manually or locally.
2. VERIFY Containment Isolation Phase A:	Do one of the following:
a. Phase A portion of Monitor Light Box B: o Red Activated Light-ON o White Status Lights-OFF	o Manually actuate CONTMT ISOL PHASE A OR o Manually Close the Phase A Isol vlvs with White Status Lights-ON.
3. VERIFY Containment Vent Isol:	Do one of the following:
# a. Containment Vent Isol portion of # P rqlwru#Dljkw#Er{#E=# o Red Activated Light - ON o White Status Lights - OFF	o Actuate CVI by Manual CONTMT # LVRO#SKDVH#D# OR o Manually Close the CVI vlvs with White Status Lights-ON.

Phase A Containment Isolation Actuation

Purpose
Obj 25

The purpose of the Phase A Containment Isolation (CI) Actuation signal is to isolate all non-essential process lines into or out of the Containment, to provide Containment integrity during accident conditions.

Description
Obj 13

The Phase A CI is either automatically actuated by a Safety Injection signal or can be manually actuated with control switches in the Control Room.

Controls
Obj 32, 33

The following controls are available for the Phase A CI Actuation signal.

In the Control Room:

Control	Operation
ACTUATE Trains A&B on VB-1	2 position, spring return to neutral
ACTUATE Trains A&B on CC-2	
RESET Train A	pushbutton
RESET Train B	

Logic
Obj 27, 29

The following logic applies to the Phase A CI Actuation signal.

IF	Then
Safety Injection is actuated	Phase A CI will be actuated. • PK02-01, CI Phase A/B will alarm. • Red light for Train A above Monitor Light Box B is illuminated. • Red light for Train B above Monitor Light Box B is illuminated.
manual Phase A CI is actuated from either switch	Phase A CI will be actuated. • Containment Ventilation Isolation is actuated. • PK02-01, CI Phase A/B will alarm. • Red light for Train A above Monitor Light Box B is illuminated. • Red light for Train B above Monitor Light Box B is illuminated.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	2	______
	K/A #	001 K6.13	
	Importance Rating	3.6*	3.7*

Proposed Question:

The DRPI Accuracy Mode Switch is in the “A + B” position. Indicated Control Bank D rod position on VB-2 is 12 steps.

Which of the following states what indicated rod position on VB-2 will be if the Accuracy Mode switch is taken to “Data A only” or “Data B only” positions?

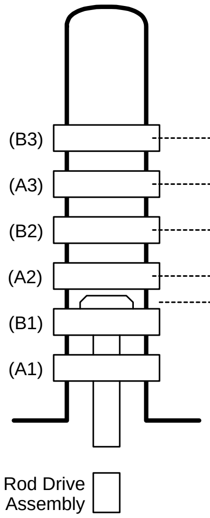
	Switch in “Data A only” Indication on VB-2 is:	**Switch in “Data B only” Indication on VB-2 is:**
A.	6 steps	6 steps
B.	6 steps	18 steps
C.	12 steps	12 steps
D.	18 steps	6 steps

Proposed Answer:

C. 12 steps 12 steps

Explanation: The resolution of the detector is six steps (3.75 inches).

The tables below describe how detector accuracy is obtained for various detector configurations.



	Coil Data (When Penetrated)		VB-2 Indicated Rod Position			True Rod Position
	A Data	B Data	A & B Data	A Only	B Only	(Steps)
	↑ Etc	↑ Etc	↑ Etc	↑ Etc	↑ Etc	↑ Etc
(B3)		(2)	24	24	24	21
(A3)	(2)		18	24	12	15
(B2)		(1)	12	12	12	9
(A2)	(1)		6	12	0	3
						0
(B1)			0	0	0	
(A1)						

Technical Reference(s): System text Digital Rod Position Indication System.
Vision questions S-0879 and S-0880

Proposed references to be provided to applicants during examination: None

Learning Objective: 4913 – Explain the operation of DRPI

Question Source:

Modified Bank # S-0879 and S-0880 (attached)
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 41.6
55.43 _____

Comments:

K/A: 001 K6.13 – Knowledge of the effect of a loss or malfunction on the following CRDS components: Location and operation of RPIS

1 S-0879 Points: 1.00 Multiple Choice

Given that Bank D is at 6 steps, What will indicated rod position be if Data A fails?

A. 0 steps

B. 4 steps

C. 6 steps

D. 12 steps

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

4913	Explain the operation of DRPI

Reference Id: S-0879
Must appear: No
Status: Active
User Text: 4913.010013
User Number 1: 3.00
User Number 2: 3.90
Difficulty: 3.00
Time to complete: 2
Topic: rod position indication if data A fails
Cross Reference: STG A3B
Comment: Checked OK, 2/25/96, GES
VALIDATION DATE: 4/3/95 DLH4 (no changes)

2 S-0880 Points: 1.00 Multiple Choice

Given that Control Bank D is at 6 steps, What will indicated position be if Data B fails?

A. 0 steps

B. 4 steps

C. 6 steps

D. 12 steps

Answer: D

ASSOCIATED INFORMATION:

Associated objective(s):

4913	Explain the operation of DRPI

Reference Id: S-0880
Must appear: No
Status: Active
User Text: 4913.010013
User Number 1: 3.00
User Number 2: 3.90
Difficulty: 3.00
Time to complete: 2
Topic: rod position indication if Data B fails
Cross Reference: STG A3B
Comment: Checked OK, 2/25/96, GES
LA-3B
VALIDATION DATE: 4/3/95 DLH4 (no changes)

Section 2.0

Major Components

Rod Position Detectors

Purpose
Obj 4

The purpose of the Rod Position Detectors is to locate the position of the control rod drive mechanism shaft within the drive housing so the position of the control rod within the core can be known.

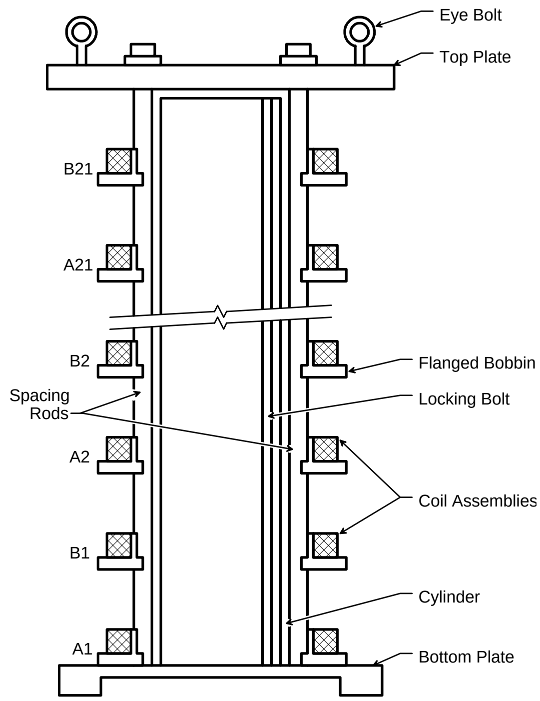
Location
Obj 5

The Rod Position Detectors are located around the CRDM housing of each rod.

Description

Each detector unit is composed of 42 coils of wire divided into two groups (21 group A and 21 group B).

Each coil is located at a 3.75″ interval along a stainless steel tube.



RPI-02

Continued on next page

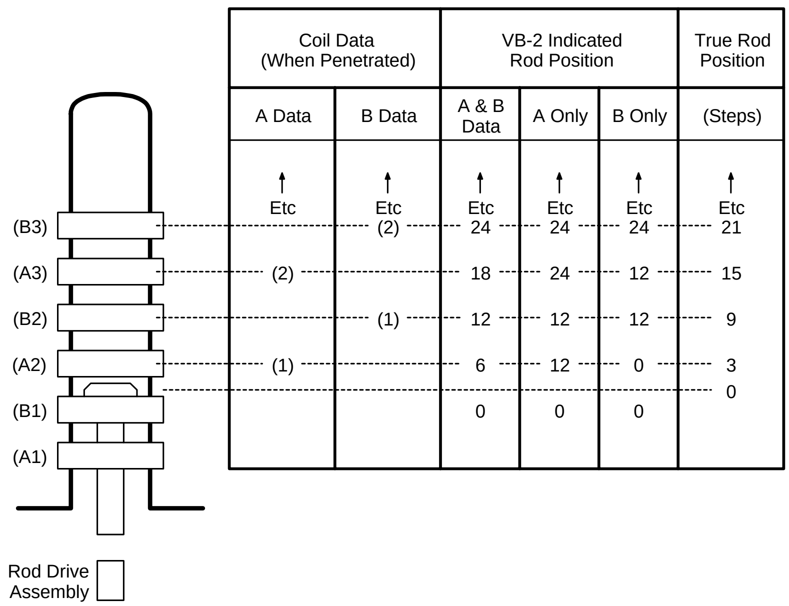
Rod Position Detectors, Continued

Detector accuracy
Obj 11

The resolution of the detector is six steps (3.75″).

- When true rod position is 0 steps the top of the CRDM shaft will be mid-way between coils B1 and A2.
  - ♦ This provides a nominal accuracy of ± 3 steps.
- Due to potential positioning errors and temperature effects another 1 step is subtracted from the accuracy making assumed accuracy ± 4 steps.

The tables below describe how detector accuracy is obtained for various detector configurations.



RPI-03

Full accuracy rod position (group A and group B coils operable)

Stage	Description
1	Determine highest coil penetrated in each group (e.g. A4 and B3)
2	Add together numeric values assigned to each coil (one less than the identifying number e.g. 3 and 2).
3	Add the values together and multiply by 6. (2+3)×6=30 steps.
4	The rod may actually be located at 30 steps ± 4 (26 steps to 34 steps).

Continued on next page

Rod Position Detectors, Continued

Detector accuracy (continued)

Half accuracy rod position (Data B coils only).

Stage	Description
1	Operable coils are only located every 12 steps.
2	Determine the highest B group coil penetrated (e.g. B2).
3	Double the value for B2 (1) and multiply by 6. 2×1×6=12 steps.
4	The rod may actually be located between the actual position of B2 (9 steps) -1 additional step for accuracy and just below the actual position of B3 (21 steps) +1 additional step for accuracy. So the rod is between 8 steps and 22 steps, giving you an assumed accuracy of +10 to -4 steps from indicated position (12 steps).

Half accuracy rod position (Data A coils only).

Stage	Description
1	Operable coils are only located every 12 steps.
2	Determine the highest A group coil penetrated (e.g. A2).
3	Double the value for A2 (1) and multiply by 6. 2×1×6=12 steps.
4	The rod may actually be located between the actual position of A2 (3 steps) -1 additional step for accuracy and just below the actual position of A3 (15 steps) +1 additional step for accuracy. So the rod is between 2 steps and 16 steps, giving you an assumed accraracy of -10 to +4 steps from indicated position (12 steps).

Control Board Display Unit, Continued

Control (continued)

Control	Operation
Accuracy Mode • A+B • A only • B only	3 position, maintain contact selector switch. • Selects full and half accuracy for the entire system.
Manual Disconnect • Off/I/II/III	4 position, maintain contact selector switch. • Disconnects an individual central control card.

Logic
Obj 9

The logic associated with the Accuracy Mode switch is shown below.

Switch position	Result
A+B	Both group A and Group B detectors are used to determine rod position. Gives full accuracy indication (±4 steps).*
A only	Only group A detectors are used to determine rod position. Gives half accuracy indication (+4 steps -10 steps).*
B only	Only group B detectors are used to determine rod position. Gives half accuracy indication (+10 steps -4 steps).*

*See the detector section for a discussion of detector accuracy.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	2	______
	K/A #	014 A4.04	
	Importance Rating	2.7	2.7

Proposed Question:

PLANT CONDITIONS:
- A Unit 1 startup is in progress following a reactor trip.
- All equipment is operable.
- The crew is placing Rod Control in service.

The operator takes the Rod Control Startup Reset switch to RESET to zero the Group Step Demand counters.

What additional change(s) should occur solely as a result of going to RESET?

A. Indicated Rod Speed goes to approximately 48 spm.

B. PK03-17, Rod Cont Urgent Failure annunciator clears.

C. PK03-18, Rod Cont Non Urgent Failure annunciator clears.

D. PK03-21, DRPI Failure/Rod Bottom annunciator clears.

Proposed Answer:
B. PK03-17, Rod Cont Urgent Failure annunciator clears.

Explanation:
A incorrect, rod speed goes to 48 spm when rods are placed in Manual.
B correct, the Urgent Failure alarms clears due to the reset.
C incorrect, the Non-Urgent failure should not be in.
D incorrect, rods are still on the bottom. The alarms clears during control bank withdrawal.

Technical Reference(s): A3A, Rod Control System, page 3-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 9917 - Explain the operation of ROD CONTROL system

Question Source:

	Modified Bank #	
	New	X
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	41.6
	55.43	_____

Comments: Resampled KA. Original KA selected (A4.03) is not applicable. Randomly selected from KA's 014 A4.01, A4.02 and A4.04.

K/A: 014 A4.04 Ability to manually operate and/or monitor in the control room: Re-zeroing of rod position prior to startup

Normal Operations, Continued

Effects of operation, switches, Rod Control System

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.

<table>
<tr><th>Effect of operating ...</th><th colspan="2">Consequence

The operator should expect to see ...</th></tr>
<tr><td>Urgent Failure Alarm Reset switch</td><td colspan="2">• Urgent Failure Alarm annunciator will clear
• Rod Motion if rods are in AUTO and a temperature deviation exists</td></tr>
<tr><td>Manual Rod Control In/Out Select switch</td><td colspan="2">• Rod height to change
• RCS temperature to change</td></tr>
<tr><td>Control Rod Lift Coil Disconnect switches</td><td colspan="2">• Rods will NOT move</td></tr>
<tr><td>Rod Control Startup switch</td><td colspan="2">• Group step counters reset to 0
• Master cycler resets to 0
• Slave cyclers reset to 0
• Bank Overlap resets to 0
• Urgent Failure Alarm resets
• P/A converters reset to 0</td></tr>
<tr><td rowspan="6">Rod Bank/Mode Select switch</td><td>Switch position</td><td>Consequence

If Manual Rod Control IN/OUT is operated then...</td></tr>
<tr><td>SBA, SBB SBC or SBD</td><td>The selected shutdown bank will move.</td></tr>
<tr><td>MAN</td><td>The control rod banks will move in their normal sequence and with correct bank overlap.</td></tr>
<tr><td>CBA, CBB, CBC or CBD</td><td>The selected control rod bank will move with no programmed sequence or bank overlap</td></tr>
<tr><td colspan="2"></td></tr>
<tr><td>AUTO</td><td>Control rods will move as determined by the Reactor Control Unit.</td></tr>
</table>

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	1	_____
	K/A #	016 K3.01	
	Importance	3.4*	3.6*

Proposed Question:

Which of the following events would cause Tave to decrease approximately 4ºF?

A. Power is reduced from 100% to 90% maintaining Tave on program.

B. 10% atmospheric steam dump valve fails open at 100% power, EOL, all rods out.

C. At 1% power, in a normal startup lineup, Main Steam pressure transmitter, PT-507 fails high. (No operator action)

D. At 100% power, with control systems in AUTO, Turbine Impulse pressure transmitter, PT-505 fails low. (No operator action)

Proposed Answer:

C. At 1% power, in a normal startup lineup, Main Steam pressure transmitter, PT-507 fails high. (No operator action)

Explanation:
A incorrect, at 100% power, Tave is 572ºF. Over all, Tave changes 25ºF, or .25ºF/%. Therefore a 10% change will cause Tave to decrease 2.5ºF.
B incorrect, power will increase approximately 2.5%. Tave will decrease less than 4ºF
C correct, PT-507 failing high will cause steam dumps (in steam pressure mode) to fail open. They will close at P-12. Tave will decrease from 547 to 543ºF, or 4ºF.
D incorrect, PT-505 failing low will cause rods to drive in. Tave will drop but without operator action, will decrease much more than 4ºF.

Technical Reference(s): A1 – Reactor Coolant System
C2B – Steam Dump System

Proposed references to be provided to applicants during examination: none

Learning Objective: 10546, 10549, 10551 - Calculate the following parameters associated with the RCS for various power levels - TH, - TC, - DT, - TAVG, - Pzr Level
8004 - Analyze Steam Dump System control logic.

Question Source: New X

Question History: Last NRC Exam N/A
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

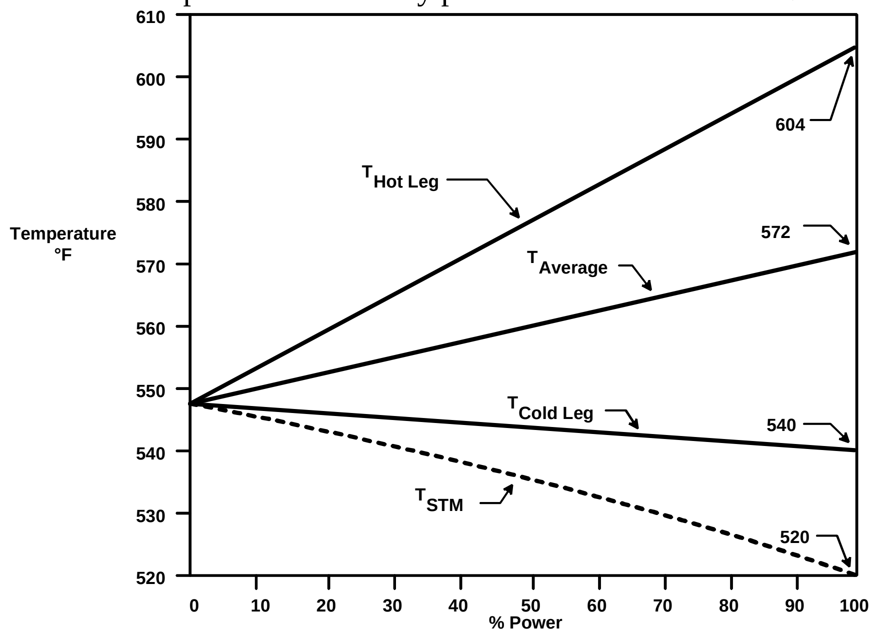
10 CFR Part 55 Content: 55.41 41.5
55.43 _____

Comments:

K/A: 016 K3.01 - Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: RCS

Variations with power

The variation with power level for key parameters is shown below.



RCS-15

Temperature calculations
Obj 45

Any temperature can be calculated as follows:

$T = P(T_{FP} - T_{ZP}) + T_{ZP}$

Where: P = present power level fraction
T_{FP} = full power value of the temperature, °F
T_{ZP} = zero power value of the temperature, °F

or,

$\Delta T = T_H - T_C$
$T_{AVG} = (T_H + T_C)/2$

Continued on next page

Unusual control response

The table below lists some unusual Steam Dump control system responses due to system failures.

Refer to OIM figure C-2-8

If	When	Then	Alternate alignments
PT-507 fails low	Steam Press mode (controller in automatic)	Steam pressure will indicate below setpoint and group 1&2 valves will close (if open).	Steam Pressure controller in manual allows direct control of valves.
PT 507 fails high	Steam Press mode (controller in automatic)	Steam pressure will indicate above setpoint and group 1&2 valves will open. • At P-12 setpoint, T_{avg} = 543°F, the group 1 & 2 valves will close. • Group 1 valves will not close if interlock is bypassed.	Steam Pressure controller in manual allows direct control of valves. Selecting Off Reset will close all valves
PT-505 fails low	T_{avg} Mode (if valves armed by a subsequent load rejection).	T_{ref} will drop to its minimum value (547°F) and armed valves will open. Load rejection controller will attempt to maintain T_{avg} = 547°F + the offset value for the unit (4°F for Unit 1 and 3°F for Unit 2).	Steam Press mode set at pressure for current power.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	2	_____
	K/A #	028 K2.01	
	Importance Rating	2.5*	2.8*

Proposed Question:

Which of the following conditions will cause the WHITE “Power In Available” light to be lit on Hydrogen Recombiner 1-2?

A. Either the normal or redundant breaker closed at 480V Vital Bus G.

B. Either the normal or redundant breaker closed at 480V Vital Bus H.

C. Both the normal and redundant breakers closed at 480V Vital Bus G.

D. Both the normal and redundant breakers closed at 480V Vital Bus H.

Proposed Answer:

D. Both the normal and redundant breakers closed at 480V Vital Bus H.

Explanation:
A incorrect, Bus G does not power recombiner 12.
B incorrect, correct power supply, but both must be closed.
C incorrect, Bus G does not supply recombiner 12
D correct, both supply breakers must be closed.

Technical Reference(s): OP H9, Inside Containment H2 Recombination System

Proposed references to be provided to applicants during examination: None

Learning Objective: 3650 - Identify the power supply for the Containment Hydrogen Recombiners

Question Source: New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: K/A: 028 K2.01 - Knowledge of bus power supplies to the Hydrogen Recombiners

INSTRUCTIONS

Record the completion of the steps required on Attachment 9.3.

6.1 Recombiner Operation

6.1.1 Select an OPERABLE recombiner unit and close its associated breaker, verify redundant breaker closed.

	NORMAL	REDUNDANT BKR*
Recombiner unit 1-1	52-1G-67	52-1G-67R
Recombiner unit 1-2	52-1H-35	52-1H-35R

* Redundant breakers installed for overcurrent protection of containment electrical penetrations. These breakers should remain closed at all times and should not be used as clearance points.

6.1.2 On the recombiner unit control panel, verify that the white lamp labeled "POWER IN AVAILABLE" is lit. This lamp indicates that the main power breaker to the unit is closed and that power is available to the control and power supply panel.

6.1.3 Verify that the potentiometer labeled "POWER ADJUST" is set at zero.

6.1.4 Turn the switch labeled "POWER OUT SWITCH" to the ON position. Verify that the red lamp on the switchplate is lit.

6.1.5 Measure and record (on the data sheet) the present containment pressure reading.

6.1.6 Determine and record the PRE-LOCA containment temperature.

6.1.7 Using Figure 1, Attachment 9.1, determine and record the power correction factor, C_p.

6.1.8 Record the Vol. 9 Reference Power on the data sheet. Multiply the Reference Power for the unit being used by the C_p to determine the recombiner power setting. Record this value on the data sheet.

6.1.9 Turn the potentiometer clockwise until 5KW is obtained on the meter labeled "POWER OUT." Hold for 10 minutes, then advance to 10KW and hold for 10 more minutes, then advance to 20KW for 5 minutes, then advance to the power setting obtained in step 6.1.8.

6.1.10 Adjust the potentiometer as required to maintain power setting in step 6.1.9 above.

6.1.11 Read and record the digital temperature indicator (all three thermocouples) and power output every 15 minutes on the data sheet until the average temperature stabilizes above the recombination temperature of 1225°F. Record the temperatures every four hours thereafter.

6.1.12 If there is any indication that the recombiner is not operating properly, turn it off and place the standby unit in operation.

6.2

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	______
Group #	2	______
K/A #	029 A1.03	
Importance	3.0*	3.3*

Proposed Question:

An alternate containment purge is being performed on unit 2 in accordance with OP H-4:II, Alternate Method For Purging Containment to reduce radionuclide concentration prior to personnel entry.

The purge should be discontinued when containment pressure reaches which of the following?

A. 0.9 psig

B. 0.0 psig

C. –0.5 psig

D. –0.9 psig

Proposed Answer:

B. 0.0 psig

Explanation:

A incorrect, this is just below the TS limit pressure.
B correct, purge should be stopped at approximately 0.0 psig.
C incorrect, operator is directed not to allow pressure to decrease below –0.3 psig.
D incorrect, this is close to the TS value.

Technical Reference(s):
OP H-4:II
TS 3.6.4

Proposed references to be provided to applicants during examination: none

Learning Objective: 5129 – State the limits for Containment Purge System

Question Source:

New	X	

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 41.10
55.43 _____

Comments:

K/A: 029 A1.03 - Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the Containment Purge System controls including: Containment pressure, temperature, and humidity

TITLE: Alternate Method for Purging Containment

06/22/04
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

   1.1 This procedure provides an alternate means to improve containment air quality or reduce radionuclide concentration prior to personnel entry.

   1.2 It is normally performed prior to a unit refueling outage, or upon request of radiation protection.

2. DISCUSSION

   2.1 This procedure describes an alternate method for purging containment utilizing the service air system.

   2.2 Performance and Verification of valve seal removal and installation for AIR-S-1-200 is documented in this procedure in accordance with OP1.DC20.

3. RESPONSIBILITIES

   3.1 Shift foreman (SFM) for operation of plant equipment as required by this procedure.

   3.2 Radiation protection engineer for analysis of purge permit results and to provide the air change requirements to the SFM prior to starting this procedure.

   3.3 Radiation protection personnel for coordinating with chemistry personnel in taking the required samples.

   3.4 Chemistry personnel for taking and analyzing all purge permit samples.

4. PREREQUISITES

   4.1 The Service Air System is inservice and the Instrument Air System is isolated from the Service Air System (AIR-0-83 closed) during the performance of this procedure. This requirement will preclude any affect on the plant Instrument Air System during this evolution.

   4.2 The Service Air System is inservice and aligned to support the large air flow rate required to perform this procedure (up to 2550 CFM).

   4.3 Obtain purge permit from Chemistry in accordance with CAP A-6.

   4.4 Obtain the most current Containment Air Sample results just prior to the start of the test.

   4.5 A RWP/SWP is established and in effect for this test.

   4.6 Check that Containment pressure is less than 0.9 psig.

   4.7 Contact the Radiation Protection Engineer to determine the volume of air or air changes required to adequately purge containment and return the air quality to acceptable levels.

5. PRECAUTIONS AND LIMITATIONS

5.1 Do not exceed 0.9 psig containment pressure.

5.2 Continue purge until containment pressure drops to approximately 0.0 psig. Do not allow containment pressure to drop below -0.3 psig.

5.3 Operator must standby continuously at Penetration 56 and maintain communications with the Control Room while valve AIR-S-1-200 is open, per Tech Spec 3.6.3 (ITS 3.6.3).

5.4 The requirements and precautions established in the RWP/SWP must be adhered to at all times. Due to personnel contamination risk, do not open the Service Air header drain valve (AIR-S-1-116) located under the fuel transfer canal, unless explicitly permitted by the RWP/SWP.

5.5 Monitor Service Air header pressure as required, at PI-1701 U2 140' near RCA Access. Do not allow Service Air header pressure to drop below 90 psig. PK13-17 will alarm when service air header pressure reaches 93 psig.

5.6 Service Air manual isolation AIR-S-0-83 shall be closed whenever Service Air System is in service, except as required by OP AP-9.

5.7 During use of this procedure, if Instrument Air requires backfeed from Service Air, per OP AP-9, immediately isolate service air to Containment by CLOSING AIR S-1-200.

6. INSTRUCTIONS

6.1 Review all Prerequisites, Precautions and Limitations.

6.2 Open up to 6 service air taps at the 91' El. level of containment and circle below which valves were opened. (Pipe wrench may be needed for caps.)

Unit 1	AZIMUTH
AIR-S-1-157	292°
AIR-S-1-160 (header drain)	300°
AIR-S-1-161	340°
AIR-S-1-164	24°
AIR-S-1-171	110°
AIR-S-1-182	153°
AIR-S-1-151	195°
AIR-S-1-154	227°

6.3 Station an operator at valve AIR-S-1-200 (Penetration 56) and establish communications with Control Room.

****
CAUTION: Closely monitor service air header pressure PI-1701, 140' U2 @ Col. G/22. Throttle the air flow into containment as required to maintain Service Air header pressure above 90 psig. PK13-17 will alarm when Service Air header pressure reaches 93 psig.
****

6.4 Break the seal and crack open valve AIR-S-1-200 at Penetration 56. Log seal change on Attachment 9.1.

6.5 Operate additional CFCUs as desired to enhance mixing of containment atmosphere.

6.6 Monitor containment pressure on VB1 or PPC. Upon reaching approximately 0.9 psig, initiate purge/vent in accordance with OP H-4:I.

NOTE: The following step may be performed at the discretion of the SFM to allow securing the watch at AIR-S-1-200 while venting the containment.

6.7 If it is desired to isolate air while venting the containment, close and seal AIR-S-1-200 and secure the watch. Log seal change on Attachment 9.1.

6.8 Discontinue purge/vent when containment pressure drops to approximately 0.0 psig.

6.9 Request Radiation Protection to take containment air sample (as appropriate).

6.10 Repeat steps 6.3 thru 6.9 as required until the containment air quality is determined acceptable by the radiation protection engineer.

6.11 Close and seal valve AIR-S-1-200 at Penetration 56 and complete seal change data on Attachment 9.1. (This step will be N/A if valve was closed in step 6.7.)

6.12 Close (and replace cap, if so equipped) the service air taps in containment that were opened in step 6.2.

6.13 Align the Service Air System as required by the shift foreman.

7. REFERENCES

7.1 OVID Dwg. 106725, Sheet 58.

7.2 OP1.DC20, "Sealed Components."

7.3 OP H-4:I, "Containment Ventilation Make Available and Place In Service."

8. RECORDS

8.1 File completed Attachment 9.1 in the completed Sealed Valve Change Form binder.

9. ATTACHMENTS

9.1 "AIR-S-1-200 Seal Removal and Installation Log," 04/21/98

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be $\geq$ -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. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	______
Group #	2	______
K/A #	034 G2.4.12	
Importance	3.4	3.9

Proposed Question:

PLANT CONDITIONS

- Core off load is in progress
- The FHB Bridge Crane is positioned over the intended fuel rack and the crew is about to commence lowering the fuel assembly

RE-58, Spent Fuel Pool Area Rad Mon goes into high alarm in the FHB

Which of the following actions should the FHB crane operator immediately take?

A. Place the fuel assembly in the upender and lower the upender then exit the FHB.

B. Lower the assembly to the bottom of the spent fuel rack, then exit the FHB.

C. Verify Fuel Handling Ventilation transferred to Iodine Removal Mode and exit the FHB.

D. Exit the FHB and notify the control room for instructions.

Proposed Answer:

B. Lower the assembly to the bottom of the spent fuel rack, then exit the FHB.

Explanation:
A incorrect, should be placed in the closest safe location
B correct per AR PK11-10,
Fuel Handling Building Worker Actions
5.2.1 Place all fuel bundles in a safe storage location.
a. Spent Fuel Storage rack
b. New Fuel elevator (new fuel only)
c. Fuel Transfer conveyer upender
d. New Fuel Storage Rack (new fuel only)
5.2.2 Perform an orderly evacuation of all persons to Access Control.
C incorrect, this is a control room action
D incorrect, the fuel bundle must be placed in a safe location.

Technical Reference(s): AR PK 11-10, FHB High Radiation RE-58 AND 59

Proposed references to be provided to applicants during examination: None

Learning Objective:
6540 - Explain the actions to take on a fuel handling building evacuation alarm

Question Source: Bank # DCPP B-0049

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 41.13
55.43 _____

Comments:

K/A: 034 G2.4.12 – Fuel Handling Equipment - Knowledge of general operating crew responsibilities during emergency operations.

1 B-0049 Points: 1.00 Multiple Choice

The following plant conditions exist:

- Core off load is in progress
- The FHB Bridge Crane is positioned over the intended fuel rack and the crew is about to commence lowering the fuel assembly
- Fuel Handling Building Rad Monitors RE-58 and 59 alarm
- Gas bubbles being released from the element is noted

Which of the following actions should the FHB crane operator immediately take?

A. Lower the assembly to the bottom of the fuel rack, then exit the FHB.

B. Place the fuel assembly in the upender and lower the upender then exit the FHB.

C. Open the supply breaker to the bridge crane and exit the FHB.

D. Exit the FHB and notify the control room for instructions.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

6540	Explain the actions to take on a fuel handling building evacuation alarm

Reference Id: B-0049
Must appear: No
Status: Active
User Text: 6540.110343
User Number 1: 3.60
User Number 2: 4.10
Difficulty: 3.00
Time to complete: 3
Topic: LPA-21 Actions to take on a fuel handling building
Cross Reference: AP-21
Comment: created for r993c15, jpl1.
Converted to B bank from R-54078 during 00 exam review mtc6
Reviewed for 00 biennial exam 1/17/01 mtc6
Reviewed for 02 biennial exam 1/7/03 mtc6
Reviewed for 04 biennial exam 12/23/04 mtc6

TITLE: **FHB High Radiation RE-58 AND 59**

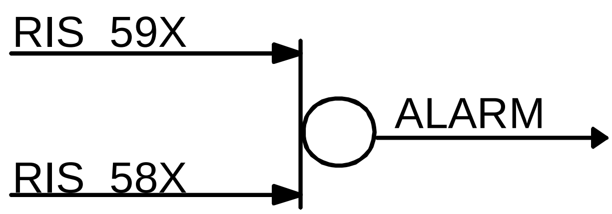
1

04/19/02
EFFECTIVE DATE

SUR FHGXUH#FODVVLILFDWLRQ=##TXDOLW\#UHODWHG#

#

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
RIS 59X	1169	New Fuel Storage Area Rad Mon RE-59, Hi Rad	Note 1
RIS 58X	1167	Spent Fuel Pool Area Rad Mon RE-58, Hi Rad	Note 1

NOTE 1: Refer to I&C Rad Monitors Data Book in the Control Room.

3. PROBABLE CAUSE

3.1 True high radiation level at the detector.

3.2 Surveillance test in progress on monitor.

3.3 Moving operation selector switch to source check.

4. AUTOMATIC ACTIONS

4.1 Fuel Handling Building evacuation alarm sounds.

4.2 Fuel Handling Building ventilation changes to "Iodine Removal" mode of operation.

5. OPERATOR ACTIONS

5.1 Control Room Operator Actions, RM-58 and RM-59 in service.

5.1.1 Verify FHB alarm is initiated (use intercom).

5.1.2 Check Radiation Level on RM-58 and RM-59 at PAM 2 panel.

5.1.3 Verify that the Fuel Handling Ventilation transferred to Iodine Removal Mode.

#

#

5.1.4 Make a PA announcement to evacuate the Fuel Handling Building.

5.1.5 Refer to the following procedures as necessary:

a. EP G-1, "Emergency Classification and Emergency Plan Activation"

b. OP AP-21, "Irradiated Fuel Damage"

c. OP AP-22, "SFP Low Level/Hi Temp/Hi Rad"

5.2 Fuel Handling Building Worker Actions

5.2.1 Place all fuel bundles in a safe storage location.

a. Spent Fuel Storage rack

b. New Fuel elevator (new fuel only)

c. Fuel Transfer conveyer upender

d. New Fuel Storage Rack (new fuel only)

5.2.2 Perform an orderly evacuation of all persons to Access Control.

5.3 Control Room Operator Actions, RM-58 or 59 out of service (Tech Spec 3.3.8).

5.3.1 If the Fuel Handling Building Radiation alarm initiates by the in-service monitor, follow Section 5.2 of this response.

5.3.2 If the portable continuous monitor installed to satisfy action A.1.2.1 of Tech Spec 3.3.8 alarms:

a. Fuel Handling Personnel should notify the Control Room immediately of the alarm.

b. Control Room Personnel shall initiate the Fuel Handling Building Radiation Alarm by performing a source check on the in-service Fuel Handling Building Radiation Monitor. If both monitors are out of service or the alarm fails to sound, a P.A. announcement shall be made informing personnel of the condition.

c. Control Room Operators shall follow Step 5.1.3 through Step 5.1.5.

d. Fuel Handling Building workers shall follow all steps in Section 5.2.

5.4 If RE-58 or RE-59 has failed or is out of service for a surveillance test, notify the Shift Chem Tech per OP O-3.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	2	_____
	K/A #	041 A2.02	
	Importance	3.6	3.9

Proposed Question:

PLANT CONDITIONS

- Unit 1 trips from 100% power due to an error while performing a surveillance.
- The crew has entered E-0.1, Reactor Trip Response.

The CO notes that RCS temperature is 545ºF and decreasing and two Group 2 steam dump valves indicate partially open. All others indicate fully closed.

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

A. Shut the MSIVs and bypass valves.

B. Implement AP-6, Emergency Boration.

C. Place the Steam Dump Bypass switches, 43/SDA and 43/SDB in OFF.

D. Take Steam Dump controller, HC-507 to Manual and press the DECREASE pushbutton.

Proposed Answer:

A. Shut the MSIVs and bypass valves.

Explanation:
A correct, all valves should be closed at this time. Per E-0.1, the MSIVs are closed.

B incorrect, this would be the action if the cooldown continued after shutting the MSIVs.

C incorrect, placing the Bypass switches in OFF removes the arming signal, but would have no effect if the valves are not being controlled by HC-507.

D incorrect, all valves should be closed at this time. E-0.1 states to verify all Steam Dump valves closed, which this would do if there was a demand for them to be open.

Technical Reference(s):

ro tier 2 group 2_62 rev2.doc

C-2B, Steam Dump System
E-0.1, Reactor Trip Response

Proposed references to be provided to applicants during examination: none

Learning Objective: 9993 - Explain the operation of the Steam Dump System.

Question Source:
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 41.5
55.43

Comments: K/A: 041 A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: Steam valve stuck open

Control System Overview, Continued

Physical description (continued)

Part	Function
Mode selector switch	These switches establish the operating mode of the Steam Dump system.
Bypass switches	
Load Rejection Controller	This controller modulates the Steam Dump valves on a load rejection. It attempts to maintain T_{avg} at a T_{ref} value (derived from PT-505 as above) plus a defined offset (4°F Unit 1 and 3°F Unit 2).
Reactor Trip Controller	This controller modulates the Group 1 and 2 Steam Dump valves on a reactor trip. It attempt to maintain T_{avg} at $T_{noload\ (nl)}$ (normally set at 547°F).
Steam Pressure Controller	This controller modulates the Group 1 and 2 Steam Dump valves when steam pressure mode is selected on the Mode Selector switch. It attempts to maintain steam header pressure at the value set at the controller. The steam pressure controller is normally set at 1005 psig. This value is varied when performing heatups or cooldowns.
UI-500	This meter indicates the output from all Steam Dump controllers (excluding the individual pressure controllers).

Continued on next page

Bypass Switches (43/SDA, 43/SDB)

Purpose
Obj. 11, 16

The purpose of 43/SDA and 43/SDB is to allow the P-12 interlock to be bypassed so the group 1 Steam Dump valves may be used to cooldown below P-12 (T_{avg} = 543°F).

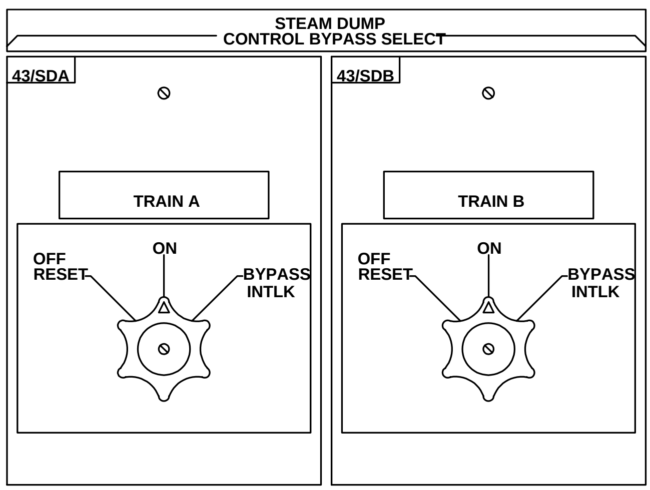
The switches may also be used to select whether or not the Steam Dump valves can be armed.

Location

43/SDA and 43/SDB are located on the CC-2 benchboard.

Control
Obj. 12

The following control positions are available for the 43/SDA and 43/SDB switches.



SDS-09

Switch position	Purpose
OFF RESET (maintained contact)	Defeats all Train A / Train B Steam Dump valve arming signals, resets bypass.
ON (maintained contact)	Steam Dump arming is aligned for normal operation.
BYPASS INTLK (spring return to ON)	P-12 will normally prevent arming of all Steam Dump valves. This switch position will bypass P-12 for group 1 Steam Dump valves only (after bypass interlock is selected, switch must be positioned to OFF RESET to remove the bypass).

Continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **CHECK RCS Temperature - STABLE OR TRENDING TO 547°F** o Average Temperature if any RCP is running OR o Cold Leg Temperature if NO RCP is running	Perform the following: a. IF RCS Temperature is LESS THAN 547°F and decreasing, THEN 1) Verify ALL Steam Dump Valves closed. 2) Verify S/G Blowdown Isol Vlvs OC - CLOSED 3) Press Reset on MSR Cont Panel. 4) Control Total Feed Flow. Maintain total feed flow GREATER THAN 435 GPM until NR level is GREATER THAN 6% [16%] in at least one S/G. 5) IF Cooldown continues, THEN (a) Close MSIVs and MSIV Bypass Vlvs. (b) Adjust 10% steam dump controllers as needed to maintain S/G pressure LESS THAN OR EQUAL TO 1005 PSIG (8.38 turns). 6) IF Cooldown continues AND is UNCONTROLLED, THEN IMPLEMENT OP AP-6 EMERGENCY BORATION. 7) GO TO Step 2 (Page 4).

THIS STEP CONTINUED ON NEXT PAGE

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	_____
	Group #	2	_____
	K/A #	068 K4.01	
	Importance	3.4	4.1

Proposed Question:

How would the liquid radwaste system respond if RE-18, Liquid Radwaste Rad Monitor, alarmed during a discharge of a Floor Drain Receiver?

The following component names apply to this question:

- FCV-647, Filter 0-4 to ASW Overboard or EDRs
- RCV-18, Liquid Waste to Overboard
- FCV-477, Filters 04 and 05 outlet to EDRs
- FCV-720, FDR recirc valve

A. RCV-18 closes. The running Floor Drain Receiver pump will receive a trip signal.

B. FCV-720 opens and RCV-18 closes. The tank that is on discharge will swap to recirculation.

C. FCV-720 opens and FCV-647 closes. The tank that is on discharge will swap to recirculation.

D. RCV-18 closes and FCV-477 opens. Flow is directed to the Equipment Drain Receiver that is on fill.

Proposed Answer:

D. RCV-18 closes and FCV-477 opens. Flow is directed to the Equipment Drain Receiver that is on fill.

Explanation:
A incorrect, RCV-18 closes, but the pump does not trip.
B incorrect, FCV-720 is not affected by RE-18.
C incorrect, neither action occurs.
D correct, the discharge terminates (RCV-18 closes) and it is directed to the EDR inlet header (FCV-477)

Technical Reference(s): G1 – Liquid Radwaste System

Proposed references to be provided to applicants during examination: none

Learning Objective: 8442 Explain automatic actions associated with Liquid Rad Waste system valve

Question Source: Bank # DCPP A-0542

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 41.13

55.43 _____

Comments:

K/A: 068 K4.01 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic, and radioactive liquids

1 A-0542 Points: 1.00 Multiple Choice

How would the liquid radwaste system respond if RE-18, Liquid Radwaste Rad Monitor, were to come in alarm during a discharge of a Floor Drain Receiver?

A. RCV-18 closes and FCV-477 opens. Flow is directed to the Equipment Drain Receiver that is on fill.

B. RCV-18 closes and FCV-477 opens. Flow is directed to the Floor Drain Receiver that is on fill.

C. RCV-18 opens and FCV-477 closes. The tank that is on discharge will swap to recirculation.

D. RCV-18 closes. The running Floor Drain Receiver pump will receive a trip signal.
>>

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8442	Explain automatic actions associated with Liquid Rad Waste system valves
69251	H{sodlq#wkh#dxwrpdwlf#dfwlrqv#dvvrfldwhg#zlwk#wkh#Oltxlg#Udgzdvwh# v|vwhp1

Reference Id: A-0542
Must appear: No
Status: Active
User Text: 8442.130494
User Number 1: 2.90
User Number 2: 3.40
Difficulty: 3.00
Time to complete: 3
Topic: RMS - Hi rad effect on a liquid radwaste discharge.
Cross Reference: 49
Comment: Copied question S-1178. Modified stem, the answer, and distractors. Should validate. 11/12/96 MAP
Validated IAW TQ2.ID3. Taken to Review. 01/23/97 jpsj
Taken to Active after review. 01/24/97 jmh1
Checked as part of question review for biennial exam 12/22/98 MTC6
Reviewed for 00 biennial exam 1/17/01 mtc6
Reviewed for 02 biennial exam 1/7/03 mtc6
Reviewed for 04 biennial exam 12/21/04 mtc6
Reviewed for L031 NRC Practice Exam 12/22/04 CNH4
Reference: STG G-1 Page 2.5-21

Liquid Radwaste Monitor, RE-18

Purpose
Obj 7

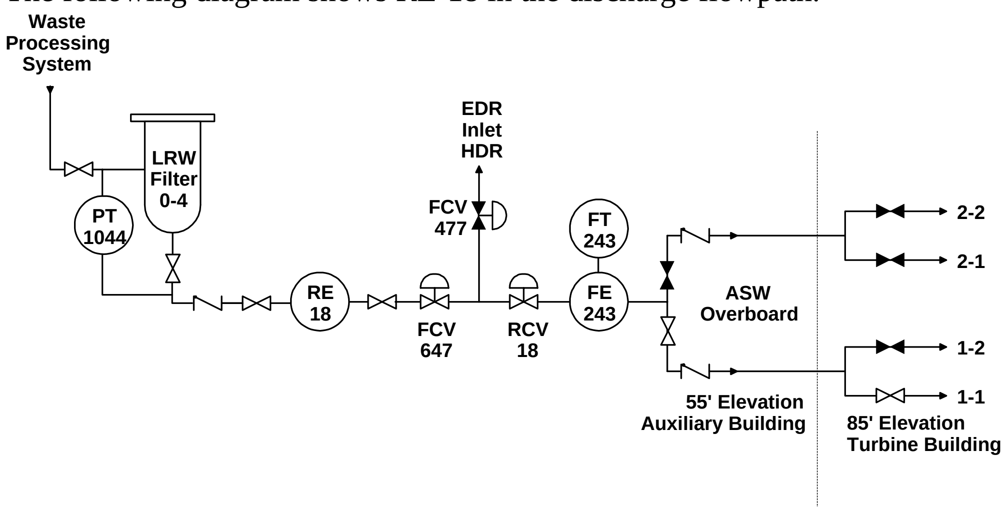
The purpose of Liquid Radwaste Monitor, RE-18, is to monitor liquid radwaste discharges for radioactivity.

Location
Obj 6

RE-18 is located on the 54′ elevation in the Auxiliary Building.

Diagram
Obj 24

The following diagram shows RE-18 in the discharge flowpath.



LRS-18

Automatic actions

Two automatic actions occur when RE-18 has detected high radiation.

- Liquid Waste to Aux Saltwater System Overboard isolation valve (RCV-18) will close shutting off flow to the Auxiliary Salt Water System.
- Filters 0-4 Outlet to the Equipment Drain Receivers valve (FCV-477) will open to recycle flow back to the EDRs.

Continued on next page

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	______
	Group #	2	______
	K/A #	072 K5.02	
	Importance	2.5	3.2

Proposed Question:

Health Physics is about to transfer a small spherical radioactive source through the Auxiliary Building. The source measures 1000 mrem/hr gamma at 6 inches. The transport route of the source will take it 5 feet away from an Area Radiation Monitoring System (ARMS) detector.

Which of the following describes the maximum radiation (due to the source) shown on the ARMS indicator?

A. 1.0 mrem/hr.

B. 10.0 mrem/hr.

C. 100.0 mrem/hr.

D. 200.0 mrem/hr.

Proposed Answer:

B. 10.0 mrem/hr.

Explanation:
B correct, Dose = DR x $(d_1/d_2)^2$
Dose = 1000 mr/hr $(.5/5)^2$
Dose = 10.0 mrem
A incorrect, math error, using either 100 mrem or .05
C incorrect, if the result of (0.5/5) is not squared.
D incorrect, if 1000/5.

Technical Reference(s): NM-10, RP for Non-Licensed Operators

Proposed references to be provided to applicants during examination: none

Learning Objective: 72414 - Given a dose rate from a point source at a given distance, estimate the dose rate at multiples of fractions of that distance time (within one order of magnitude)

Question Source: Modified Bank # INPO 22492

Question History: Last NRC Exam DCPP 10/2002

Question Cognitive Level:

Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 _____

Comments: K/A: 072 K5.02 - Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation intensity changes with source distance

ID: 22492 Health Physics is about to transfer a small spherical radioactive source through the Auxiliary Building. The source measures 100 mrem/hr gamma at 1 foot distance. The transport route of the source will take it 5 feet away from an Area Radiation Monitoring System (ARMS) detector.

Which one of the following describes the correct maximum radiation (due to the source) shown on the ARMS indicator?

Ans 4 mrem/hr.

D1 1 mrem/hr.

D2 10 mrem/hr.

D3 20 mrem/hr.

AbbrevLocName	ExamDate	Vendor	Type	ExamType	Cog Level	ExamLevel	RefMaterial	ParentId
Diablo Canyon 1	10/1/2002	WEC	PWR	ILO	2	R		

QuestionComment	
Distract1Comment	
Distract2Comment	
Distract3Comment	

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
..072.K5.02			072	K5	02	

Distance

Objective 7

Given a dose rate from a point source at a given distance, estimate the dose rate at multiples or fractions of that distance time (within one order of magnitude).

Introduction

Distance is oft times the easiest and best of the ALARA tools. When an RP tech insists that you move away from a hot spot or use long handled tools, they are not doing it to harass you.

Main Idea

$I_{moved} = I_{known} \text{ X } (\text{distance}_{known}/\text{distance}_{moved})^2$

Example

A hot particle reads 20 mR/h at 2 feet, what is its dose rate at 6 inches? What is it at 4 ft?
I_{moved} = 20 mR/h X (2 ft$_{known}$/0.5 ft$_{moved}$)^2 = 320 mR/h at 6 inches
I_{moved} = 20 mR/h X (2 ft$_{known}$/4 ft$_{moved}$)^2 = 5 mR/h at 4 ft.

Practice / Feedback

A one inch valve reads 14 mR/h at 4 ft. what will it probably read at 1 ft? Ans.: 224 mR/h

RO QUESTION 65

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	______
Group #	2	______
K/A #	075 K1.08	
Importance	3.2*	3.2*

Proposed Question:

When would it be appropriate to cross-tie the ASW and Circ Water bays?

A. If one unit loses its ASW pumps and the other unit's ASW pumps are not available.

B. If the Circ Water screens are severely clogged and the ASW screens are not.

C. If Chlorine injection into the ASW system is necessary.

D. If the ASW pumps are losing suction.

Proposed Answer:

D. If the ASW pumps are losing suction.

Explanation:

A incorrect, the correct action would be to trip the reactor.
B incorrect, .this is the opposite of when the systems would be tied.
C incorrect, Chlorination is to both the Circ Water and ASW bays.
D correct, if the bay level is low and the pumps are losing suction (or cavitating), cross-tying maybe appropriate if the Circ Water screens are less affected.

Technical Reference(s):
AR PK01-03, Aux Saltwater Pumps (rev 13)
AP-10 pages 2, 3 (rev 8)

Proposed references to be provided to applicants during examination: none

Learning Objective: 3547 - Describe the actions to be taken in the event the ASW screens cannot be unclogged

Question Source:

ro tier 2 group 2_65.doc

New X

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 41.4

55.43 _____

Comments:

K/A: 075 K1.08 - Knowledge of the physical connections and/or causeeffect relationships between the circulating water system and the following systems: Emergency/essential SWS

#

PACIFIC GAS AND ELECTRIC COMPANY	**NUMBER** **OP AP-10**
NUCLEAR POWER GENERATION	**REVISION** **8**
DIABLO CANYON POWER PLANT	**PAGE** **1 OF 6**
ABNORMAL OPERATING PROCEDURE	**UNITS**

TITLE: Loss of Auxiliary Salt Water

1 AND 2

08/30/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. <u>SCOPE</u>

1.1 This procedure provides guidance in restoring auxiliary saltwater (ASW) flow to a CCW heat exchanger in the event flow is lost due to:

1.1.1 Failure of both ASW pumps on the same unit.

1.1.2 Loss of suction to the pump in service.

1.1.3 A rupture in the system piping or fouling of a CCW heat exchanger.

1.1.4 Extensive equipment damage at the intake due to tsunami or other causes.

1.2 This procedure should be used in Modes 1-4. If in Modes 5 and 6, OP AP SD-3, "Loss of Auxiliary Saltwater," should be used.

2. <u>SYMPTOMS</u>

2.1 Possible Main Annunciator Alarms:

2.1.1 AUXILIARY SALT WATER SYSTEM (PK01-01)

a. CCW HX ____ Aux Salt Wtr Diff Press Hi-Lo

b. Aux Salt Wtr Hdr ____ Press Lo

2.1.2 AUX SALT WATER PUMPS (PK01-03)

a. Aux Salt Wtr Pps OC Trip

b. Aux Salt Wtr Pp ____ Bay Lvl Lo

c. Aux Salt Wtr Pp ____ Auto Start

2.1.3 CCW VITAL HDR A/B (PK01-06)

CCW HX ____ Outlet Temp

2.1.4 BAR RACKS SCREENS (PK13-01)

Screen Diff Hi

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY an ASW Pump Running** • ASW Pp 1 • ASW Pp 2	a. If ASW Pumps are available on the opposite unit, then on the opposite unit: 1) Start standby pump, if available. a) IF Standby is Pump No. 1, THEN Close FCV-495 AND open FCV-496 -OR- IF Standby is Pump No. 2, THEN Close FCV-496 AND open FCV-495. b) Open Unit 1 and 2 ASW cross-tie valve, FCV-601. 2) IF Only one pump is available, THEN Open Unit 1 and 2 ASW cross-tie valve, FCV-601 to supply both units with one pump. b. Comply with action statement of Tech Spec 3.7.8 for both units OR Tech Spec 3.0.3 as applicable. c. Stop any radwaste discharge in progress. d. If ASW Pumps are not available on the opposite unit and it is determined that pumps are not capable of being placed in service, then GO TO OP AP-11, Section A. e. GO TO Step 4.
2. **VERIFY ASW Pump Has Not Lost Suction:** Running pump amps - STEADY	a. Start standby pump. b. Secure cavitating pump. c. Determine cause of cavitation. 1) Verify both ASW Inlet Gates 8 and 9 OPEN.

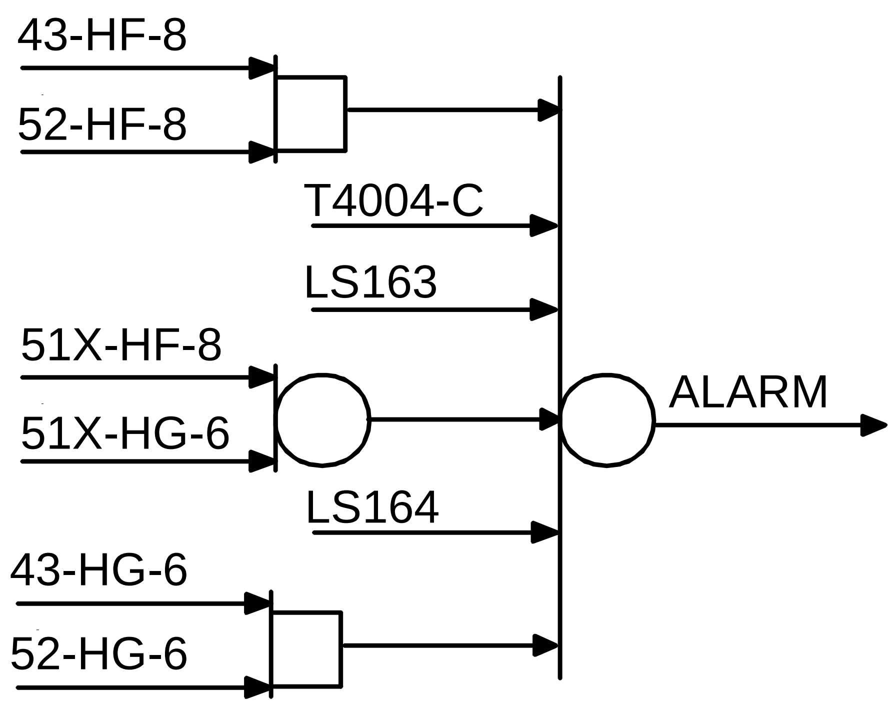
ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
2. **VERIFY ASW Pump Has Not Lost Suction: (Continued)**	
	NOTE: The ASW pumps have been successfully tested to run, without cavitation, at sea level of -20 feet. Alarm setpoint is at -6 feet.
	2) CHECK PK01-03 NOT in alarm due to ASW Bay Level Low. If in alarm: • VERIFY no Screen Wash Pump Running in affected bay • Screen differential pressure not high (PK13-01 NOT lit) - If high, ensure screens AND screen wash pumps in operation. • Crosstie ASW and Circ Water Bays by: 1) OPEN FCV-432 (ASW Pp 1 Bay) OR FCV-433 (ASP Pp2) Bay. 2) Locally OPEN FCV-604 OR 605
	IF Both ASW pumps cavitating AND the cause of the cavitation cannot be corrected. THEN 1) Secure both ASW pumps 2) RETURN TO Step 1
	--- -

TITLE: AUX SALTWATER PUMPS

03/04/04
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	PPC ADDRESS
T4004C	426	Aux Salt Wtr Pps Temp PPC	
		The inputs to device T4004C are:	
		1. ASW Pp 1-1 motor top bearing	T2435A
		2. ASW Pp 1-1 motor bottom bearing	T2436A
		3. ASW Pp 1-1 stator temp	T2437A
		4. ASW Pp 1-2 motor top bearing	T2439A
		5. ASW Pp 1-2 motor bottom bearing	T2440A
		6. ASW Pp 1-2 stator temp	T2441A

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
51X-HF-8	427	Aux Salt Wtr Pps OC Trip	Min. 75 amps
			Inst 645 amps
51X-HG-6	427	Aux Salt Wtr Pps OC Trip	Min. 75 amps
			Inst 645 amps
LS 163	1013	Aux Salt Wtr Pp 1-1 Bay Lvl Lo	-6 feet
LS 164	1014	Aux Salt Wtr Pp 1-2 Bay Lvl Lo	-6 feet
43-HF-8	61	Aux Salt Wtr Pp 1-1 Auto Start	Cont. Switch in AUTO
52-HF-8			Bkr Closed
43-HG-6	74	Aux Salt Wtr Pp 1-2 Auto Start	Cont. Switch in AUTO
52-HG-6			Bkr Closed

3. PROBABLE CAUSE

3.1 Motor problems

3.1.1 Loss of ventilation

3.1.2 Motor overloaded

3.1.3 Excessive flow

3.1.4 Failed bearing

3.1.5 Excessive vibration

3.2 Suction pit low level

3.2.1 Pump running with suction pit inlet gate closed.

3.2.2 Traveling screen clogged.

3.3 Auto start

3.3.1 Low discharge header pressure on running pump.

3.3.2 U.V. on 4KV Vital Bus of running pump.

3.3.3 Safety Injection

3.3.4 Transfer to DSL with no S.I.

3.3.5 Auto Transfer to Startup Power.

4. AUTOMATIC ACTIONS

4.1 Possible trip of running pump.

4.2 Possible auto start of standby pump.

5. <u>OPERATOR ACTIONS</u>

5.1 Running ASW pump tripped or secured. **Alarm Input 427**

5.1.1 Verify standby ASW pump running with normal current. If all ASW flow is lost, GO TO OP AP-10/AP SD-3, and implement step 5.3 of this procedure.

5.1.2 Verify normal parameters on CCW-ASW HX, ASW flow and ASW discharge pressure.

5.1.3 Verify continuous chlorination secured to idle ASW pump bay. Refer to OP E-3:VI.

5.1.4 Refer to TS 3.7.8 and ECG 17.2.

5.2 Motor problems. **Alarm Input 426**

5.2.1 If a pump tripped on overcurrent, perform Step 5.1, AND:

a. Dispatch an Operator to inspect the switchgear. Refer to Operations Policy B-2.

b. Notify Maintenance Services of O.C. trip.

5.2.2 If an ASW pump motor stator or bearing temperature is in alarm then:

a. Check the room ventilation fan running.

b. Check the motor oil levels.

c. Swap to an alternate pump if available.

d. If the pump must be left running then follow Step 5.2.3.

5.2.3 An ASW pump can be run up to the following temperatures without affecting the lifetime of the component. If any of these limits are reached or exceeded then record how long and by how far the limits are exceeded and initiate an Action Request.

a. ASW pump motor stator - 248°F

b. ASW pump motor top bearing - 210°F

c. ASW pump motor bottom bearing - 210°F

5.3 Suction pit low level. **Alarm Inputs 1013, 1014**

5.3.1 Transfer pumps.

NOTE: The ASW pumps have been successfully tested to run, without cavitation, at sea level of -20 feet. Alarm setpoint is at -6 feet sea level. Therefore, if it must be done, the pump can handle it down to -20 feet.

5.3.2 Shutdown the pumps with the low level.

a. Implement step 5.1 of this procedure.

5.3.3 Check the inlet gate open, if inlet gate is closed open it after bay level is reestablished.

5.3.4 If the traveling screen is plugged due to excessive debris, refer to OP AP-10/OP SD-3 and OP AP-11/OP SD-4.

5.3.5 Consider opening any combination of demusseling valves (FCVs 432, 433, 604 and 605) if CWP screens are less impacted than the ASW screen.

5.3.6 Consider having Maintenance manually rolling the ASW screens to expose clear screens into the flow path by turning the motor coupling either by hand or with a drill motor drive.

5.3.7 Refer to TS 3.7.8.

5.4 Auto start. **Alarm Input 74**

5.4.1 Check normal current on the running pump.

5.4.2 Verify normal parameters on CCW-ASW HX, ASW flow and ASW discharge pressure.

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	______
Group #	1	______
K/A #	G2.1.10	
Importance	2.7	3.9

Proposed Question:

Which of the following conditions, by itself, would be a violation of the Facility Operating License for Unit 1?

A. 101% indicated NI power.

B. Calculated power of 3415 MWth.

C. Changing load in excess of 5 megawatts per minute.

D. Performing a power ascension at greater than 5%/hour.

Proposed Answer:

B. Calculated power of 3415 MWth.

Explanation:
A incorrect, 101% NI does not necessarily correlate to power in excess of RTP.
B correct, as a condition of the operating license, PG&E can operate the unit not in excess of 3411MWth.
C incorrect, this exceeds the limit set to minimize the effects of xenon oscillations on axial offset.
D incorrect, this would violate fuel conditioning guidelines but not the Facility License.

Technical Reference(s):
Facility Operating License, Units 1 and 2.

Proposed references to be provided to applicants during examination: none

Learning Objective: 9666 - Identify the operating license contents.

Question Source:
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 41.10	
	55.43	

Comments:

K/A: G2.1.10 – Knowledge of conditions and limitations in the facility license.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 178, are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Initial Test Program

The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods, or acceptance criteria for any test identified in section 14 of PG&E's Final Safety Analysis Report, as amended, as being essential;

c. Performance of any test at a power level different from that described in the program; and

d. Failure to complete any test included in the described program (planned or scheduled for power levels up to the authorized power level).

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	______
Group #	1	______
K/A #	G2.1.21	
Importance	3.1	3.2

Proposed Question:

You have been given a procedure to perform work.

You notice that the procedure was "issued for use" 4 days ago. The work has not been started.

Which one of the following actions, if any, is necessary prior to beginning the work?

A. Revalidate the procedure using the Procedure Navigator.

B. The current procedure cannot be used, get the procedure reissued.

C. Check the copy you have has a signature on the front page and the number of pages and attachments match with another copy of the procedure.

D. Once the procedure has been “issued for use”, no action is necessary prior to beginning the work.

Proposed Answer:

A. Revalidate the procedure using the Procedure Navigator.

Explanation:

A correct, Immediately prior to initially starting a job (i.e., within the shift), issue the procedure for use (see below instructions) <u>or revalidate it if already issued for use. To revalidate procedures, use the Procedure Navigator procedure properties or list of recently revised procedures.</u>

B incorrect, reissuing is not necessary.

C incorrect, This is part of issuing the procedure for use.

D incorrect, validation is required each shift.

Technical Reference(s):

AD1.ID1 R13, Procedure Use and Adherence

Proposed references to be provided to applicants during examination: none

Learning Objective: 9798 – State requirements for procedure deviation.

Question Source: Bank #
Modified Bank # DCPP N-72598
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 41.10
55.43 _____

Comments:

K/A: G2.1.21 – Ability to obtain and verify controlled procedure copy.

4.3 ISSUING PROCEDURES FOR USE

1) Procedures copied from controlled manuals or printed from the Procedure Navigator shall be "issued for use" in accordance with the below instructions prior to performing physical work. The intent of issuing a procedure for use is to ensure the performer has the latest revision when initially starting a job.
   a) Immediately prior to initially starting a job (i.e., within the shift), issue the procedure for use (see below instructions) or revalidate it if already issued for use. To revalidate procedures, use the Procedure Navigator procedure properties or list of recently revised procedures.
   b) Once issued for use, chemistry and radiation protection procedures (other than surveillance tests) that are implemented for recurring, repetitive tasks do not require revalidation prior to the start of each task for the duration of the issued for use period.

2) To issue a procedure for use:
   a) Obtain a copy of the procedure. Include any OTSCs outstanding against the procedure. Do not use copies of procedures that have an "uncontrolled procedure" banner.
   b) Verify that the procedure is the current version (revision level and OTSCs) by comparison against the Procedure Navigator procedure properties. Priority 1 controlled manuals may also be used for this purpose. If the Procedure Navigator is unavailable, the shift manager may authorize that both control copy #5 (plant library) AND the EC/OTSC drop box (site procedure group area) be checked to determine the current version.
   c) If not already available, enter the following information (or similar) on the first page or cover sheet:

      ISSUED FOR USE By: _________ Date: _________ Expires: _________

   d) Sign, date, and enter an expiration date in the Issued for Use block. The issued for use period may be up to 31 days.
   e) Ensure the level of use classification is indicated on the procedure or cover sheet.
   f) Only the pages needed for work need be issued for use.
      - Identify all pages issued on the first page of the procedure or cover sheet.
      - Ensure all precautions and prerequisites applicable to the task are included.

3) Procedures in use longer than the issued for use period shall be re-validated and re-issued for use.

4) Procedure Forms and Attachments
   a) Continuous, Periodic, or Multiple Use Procedures - These forms and attachments are part of a procedure that is issued for use.
   b) Reference Use Procedures - These forms and attachments do not need to be issued for use. However, the user shall ensure that current forms and attachments are used.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	_____
	Group #	1	_____
	K/A #	G2.1.29	
	Importance	3.4	3.3

Proposed Question:

Which of the following is the proper method for independently verifying the position of a normally SEALED CLOSED manual valve?

A. Visually check the position of the valve stem to verify the valve position.

B. Remove the seal and attempt to move the valve in the open direction without using excessive force, then close and reseal.

C. Check the sealed component checklist binder to determine if the valve has been opened.

D. Remove the seal, attempt to move the valve in the closed direction without using excessive force then reseal.

Proposed Answer:

D. Remove the seal, attempt to move the valve in the closed direction without using excessive force then reseal.

Explanation:

A incorrect, visual verification may be made on a throttle valve if other indications are unavailable
B incorrect, valve is not moved in the closed direction.
C incorrect, seal must be removed and position verified.
D correct, to verify the valve open:

1. Compare the component tag number with the implementing document.
2. Physically verify the valve position by checking the valve open.
   - Move the valve in closed direction without using excessive force.

   Sealed or locked valves shall be verified with the sealing or locking devices removed.

Technical Reference(s): OP1.DC2 page 6 of 9

Proposed references to be provided to applicants during examination: none

Learning Objective:	7912 – Explain actions to take when performing an alignment on valves based on their position
Question Source:	Bank # DCPP P-1544

Question History: Last NRC Exam STP 11/94

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	____

10 CFR Part 55 Content: 55.41 41.10
55.43 _____

Comments:

K/A: G2.1.29 - Knowledge of how to conduct and verify valve lineups.

4.3.2 How to Perform Peer Checking

1) Before another individual performs a task,
   a) The peer establishes visual and verbal communication with the performer.
   b) The performer informs the peer of his intended actions, the purpose of those actions, and establishes hand contact with the component to be manipulated.
   c) The peer will concur with the component selected and the intended manipulation.
      (1) The peer should have a clear view of the performer's actions. This includes being on the same side of a control console as a performer when the action is performed on the control console.
      (2) The peer should check integrator and potentiometer settings made by the performer.
   d) The performer then manipulates the component.

4.4 DOCUMENTATION REQUIREMENTS

All plant procedures that have been evaluated to require independent verification shall include provisions for documenting independent verification.

1) The procedure or data sheet shall provide spaces for both the performer and the independent verifier to initial each step where components are manipulated.

2) The procedure, data sheet, work package, or combination thereof shall provide spaces for the initials and names of both performer and the independent verifier.

4.5 EQUIPMENT VERIFICATION TECHNIQUES

The following table the verification techniques that should be used when verifying the position of various components.

Table 1: Verification Techniques

Device & Position	Verification Technique
Manual Valves CLOSED	1. Compare the component tag number with the implementing document. 2. Physically verify the valve position by checking the valve closed. • Attempt to move the valve in the CLOSED direction without using excessive force. Sealed or locked valves shall be verified with the sealing or locking devices removed.
Manual Valves OPEN	1. Compare the component tag number with the implementing document. 2. Physically verify the valve position by checking the valve open. • Move the valve in closed direction enough to verify stem movement. • Return the valve to the full open position.

Table continues on next page.

1 P-1544 Points: 1.00 Multiple Choice

WHICH ONE (1) of the following is the proper method for independently verifying the position of a normally SEALED OPEN valve?>>

A. Remove the seal, move the valve in the closed direction enough to verify stem movement, return valve to original position and seal.

B. Visually check the position of the valve stem to verify the valve position.

C. Remove the seal and attempt to move the operator in the open direction, and reseal.

D. Check the sealed component checklist binder to determine if the valve has been closed.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3598	Explain the characteristics of the sealed valve program

Reference Id: P-1544
Must appear: No
Status: Active
User Text: 3598.13ALLN
User Number 1: 3.90
User Number 2: 4.10
Difficulty: 1.00
Time to complete: 3
Topic: Verification of sealed valve position
Cross Reference: OP1.DC2, LADM-9, NADM1
Comment: SOUTH TEXAS RO EXAM 11/01/94
OP1.DC2

RO QUESTION 69

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	_____
	Group #	2	_____
	K/A #	G2.2.28	
	Importance	2.6	3.5

Proposed Question:

OP B-8H, Non-Refueling Fuel Handling Instructions, states that the new fuel elevator is normally used only for lowering new fuel assemblies into the SFP.

An exception is to raise a new fuel assembly out of the spent fuel pool.

Whose approval is necessary to perform this exception?

A. Shift Manager

B. Fuel Handling Supervisor

C. Station Director or his delegate

D. Vice President and General Manager or his delegate

Proposed Answer:

C. Station Director or his delegate

Explanation:
A incorrect, SFM responsible operation of the plant and plant equipment needed to support the fuel movements and the proper logging of the movements in the Operations Shift Log.
B incorrect, FHS responsible for supervision of fuel or load handling operations, response to high radiation or other emergency conditions and communications with the Control Room.
C correct, per caution in procedure, The new fuel elevator is normally used only for lowering new fuel assemblies into the SFP. The exceptions to this are for movement of the dummy fuel assembly, raising a new fuel assembly out of the SFP or for fuel repair. <u>These exceptions require prior permission of the Station Director or his delegate.</u>
D incorrect, VP level is not necessary.

Technical Reference(s): OP B-8H page 3 of 8

Proposed references to be provided to applicants during examination: none

Learning Objective: 4501 - Explain the operation of NEW FUEL ELEVATOR

Question Source:

New X

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 41.13

55.43

Comments: K/A: G2.2.28 – Knowledge of new and spent fuel movement procedures.

4.8 If irradiated fuel assemblies are to be raised in the New Fuel Elevator, the elevator shall be configured for such use in accordance with PEP M-42.01.

4.9 The Spent Fuel Bridge Crane has been visually inspected in accordance with MP M-50.3 within one day of use and re-performed prior to use on subsequent days.

4.10 The Spent Fuel Bridge Crane and associated interlocks have been functionally tested per STP M-27A within seven days prior to the start of fuel handling operations.

4.11 The Fuel Handling Building overhead crane prior to use has been:

4.11.1 Visually inspected in accordance with M-50.3, and

4.11.2 The interlocks verified per STP M-43.

4.12 The SFP Bridge Crane Auxiliary Hoist load cells shall be in calibration to assist in determining drag limits described in step 5.5. Calibration is needed only during fuel movement.

4.13 The FHB Overhead Crane operator shall be qualified per maintenance services intranet site, qual. number MG0851Q.

4.14 The NFSV and SFP Radiation Monitors (RE-58 and 59) and a Gaseous Activity FHB Ventilation Change Radiation Monitor is OPERABLE per TS 3.3.8.

4.15 Housekeeping Zones have been established per AD4.

4.16 Perform a tailboard with involved personnel regarding the overall plan and the applicable portions of sections 5 and 6 of this procedure. For fuel or component movements an ICA map showing the approved moves will normally accompany the Movement Authorizations. In addition, a daily tailboard is required for personnel addressing responses to high radiation alarms and evacuations per step 6.1.

4.17 Prior to movement of the SFP bridge crane ensure the skimmer suction T-handles are lowered below the crane rail.

5. PRECAUTIONS AND LIMITATIONS

****

CAUTION: The new fuel elevator is normally used only for lowering new fuel assemblies into the SFP. The exceptions to this are for movement of the dummy fuel assembly, raising a new fuel assembly out of the SFP or for fuel repair. These exceptions require prior permission of the Station Director or his delegate.

****

NOTE: The breaker for the new fuel elevator has a lock controlled by the shift foreman to preclude inadvertent use.

5.1 Because of criticality considerations, placement of fuel in the SFP shall be per TS 3.7.17. Observe "cells not to be used for fuel" on spent fuel rack maps.

5.2 Personnel should observe all crane safety rules in addition to the special fuel handling requirements. This includes no mounting or dismounting of a moving crane.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	______
	Group #	2	______
	K/A #	G2.2.33	
	Importance	2.5	2.9

Proposed Question:

Unit 2 is performing a plant shutdown and driving in the control banks.

As the rods are inserted, at what point will the Control Bank A rods begin to insert?
Note: Overlap thumbwheels are set for 100 steps.

A. When the 4 Control Bank B rods reach 100 steps.

B. When the 4 Control Bank B rods reach 128 steps.

C. When the 8 Control Bank B rods reach 100 steps.

D. When the 8 Control Bank B rods reach 128 steps.

Proposed Answer:

C. When the 8 Control Bank B rods reach 100 steps.

Explanation:

A incorrect, there are 8 CBB rods for Unit 2, 4 for Unit 1.
B incorrect, there is 100 steps of overlap for CBB and CBA, therefore, CBA will begin moving at CBB at 100.
C correct, there are 8 CBB rods for Unit 2, 4 for Unit 1 and there is 100 steps of overlap for CBB and CBA, therefore, CBA will begin moving at CBB at 100.
D incorrect, there is 100 steps of overlap for CBB and CBA, therefore, CBA will begin moving at CBB at 100.

Technical Reference(s):

STP R-1A, attachment 9.1 and 9.2
OIM A-3-4

Proposed references to be provided to applicants during examination: none

Learning Objective: 5038 - Explain the unit differences for the Rod Control system.
5048 - Explain the operation of the Bank Overlap Unit.

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 41.6
55.43

Comments: Unit difference.

K/A: G2.2.33 – Knowledge of control rod programming.

DIABLO CANYON POWER PLANT
STP R-1A
ATTACHMENT 9.1

1

TITLE: Table 1 - Exercising Unit 1 Rod Cluster Control Assemblies

DATE ________________ TIME ____________

ROD BANK SELECTOR SWITCH POSITION _____________ POWER LEVEL ________________ %

ROD BANK	ROD GROUP	ROD	INITIAL POSITION STEP COUNTER	INITIAL POSITION DRPI*	EXERCISED POSITION STEP COUNTER	EXERCISED POSITION DRPI*	FINAL POSITION STEP COUNTER	FINAL POSITION DRPI*
S_A	1	D4						
		D12						
		M12						
		M4						
	2	G5						
		E9						
		J11						
		L7						
S_B	1	G3						
		C9						
		J13						
		N7						
	2	C7						
		G13						
		N9						
		J3						
S_C		H2						
		B8						
		H14						
		P8						
S_D		F6						
		F10						
		K10						
		K6						
C_A	1	E3						
		C11						
		L13						
		N5						
	2	C5						
		E13						
		N11						
		L3						

* Arrows may be used when data are the same.

STP R-1A (UNIT 1)
ATTACHMENT 9.1

TITLE: Table 1 - Exercising Unit 1 Rod Cluster Control Assemblies

ROD BANK	ROD GROUP	ROD	INITIAL POSITION STEP COUNTER	INITIAL POSITION DRPI*	EXERCISED POSITION STEP COUNTER	EXERCISED POSITION DRPI*	FINAL POSITION STEP COUNTER	FINAL POSITION DRPI*
C_B	1	D8						
		M8						
	2	H4						
		H12						
C_C	1	C3						
		C13						
		N13						
		N3						
	2	H6						
		F8						
		H10						
		K8						
C_D	1	F2						
		B10						
		K14						
		P6						
	2	B6						
		F14						
		P10						
		K2						
		H8						

* Arrows may be used when data are the same.

REMARKS: ______________________________

____

____

____

____

Performed by: ______________________________ Date ______________

DIABLO CANYON POWER PLANT
STP R-1A
ATTACHMENT 9.2

2

TITLE: Table 2 - Exercising Unit 2 Rod Cluster Control Assemblies

DATE ________________ TIME ____________

ROD BANK SELECTOR SWITCH POSITION _____________ POWER LEVEL ________________ %

ROD BANK	ROD GROUP	ROD	INITIAL POSITION		EXERCISED POSITION		FINAL POSITION	
			STEP COUNTER	DRPI*	STEP COUNTER	DRPI*	STEP COUNTER	DRPI*
S_A	1	D2						
		B12						
		M14						
		P4						
	2	B4						
		D14						
		P12						
		M2						
S_B	1	G3						
		C9						
		J13						
		N7						
	2	C7						
		G13						
		N9						
		J3						
S_C		E3						
		C11						
		L13						
		N5						
S_D		C5						
		E13						
		N11						
		L3						
C_A	1	H6						
		H10						
	2	F8						
		K8						

* Arrows may be used when data are the same.

STP R-1A (UNIT 2)
ATTACHMENT 9.2

TITLE: Table 2 - Exercising Unit 2 Rod Cluster Control Assemblies

ROD BANK	ROD GROUP	ROD	INITIAL POSITION STEP COUNTER	INITIAL POSITION DRPI*	EXERCISED POSITION STEP COUNTER	EXERCISED POSITION DRPI*	FINAL POSITION STEP COUNTER	FINAL POSITION DRPI*
C_B	1	F2						
		B10						
		K14						
		P6						
	2	B6						
		F14						
		P10						
		K2						
C_C	1	H2						
		B8						
		H14						
		P8						
	2	F6						
		F10						
		K10-						
		K6						
C_D	1	D4						
		D12						
		M12						
		M4						
	2	H4						
		D8						
		H12						
		M8						
		H8						

* Arrows may be used when data are the same.

REMARKS: ______________________________

____

____

____

____

Performed by: ______________________________ Date ______________

RO QUESTION 71

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	______
Group #	3	______
K/A #	G2.3.1	
Importance	2.6	3.0

<u>Proposed Question</u>:

Which of the following identifies the person who must approve the issuing of a Very High Radiation Area (VHRA) key?

A. RP Foreman

B. RP Manager or his designee

C. Shift Manager

D. Operations Manager or his designee

Proposed Answer:

B. RP Manager

Explanation:

A incorrect, the foreman approves transfer of issued keys to others
B correct, per RCP D-222, Radiation Protection Lock and Key Control, the RP Manger (or his designee) must approve issuing key to VHRA
C incorrect, SM approval not required for key issue.
D incorrect, Ops Manager approval not required.

Technical Reference(s): RCP D-222, Radiation Protection Lock and Key Control

Proposed references to be provided to applicants during examination: none

Learning Objective: 71145 - STATE the 10CFR20 radiation dose limits for TEDE, skin, extremeties, lense of the eye, individual organs, the embroyo/fetus and a member of the public

Question Source: Bank # INPO 23099

Question History: Last NRC Exam Salem 2002

Question Cognitive Level:

	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	____

10 CFR Part 55 Content: 55.41 41.12
55.43 43.4

Comments:

K/A: G2.3.1 - Knowledge of 10 CFR: 20 and related facility radiation control requirements.

7.2 Key Controls**{**tc "Key Controls" \f C \l 2**}**

7.2.1 LHRA and VHRA keys may be issued to qualified individuals, as specified in 5.1 and 5.3 of this procedure, provided access to the area (s) of interest has been authorized in accordance with RCP D-220.

****

CAUTION: Prior to issuing a VHRA key, approval from the radiation protection manager or designee is required.

****

7.2.2 LHRA and VHRA keys issued from the key box shall be logged in the key issue log, or electronic equivalent, regardless of the length of time that the key is out.

7.2.3 Issued keys may be transferred in the field to other qualified individuals provided that:

a. A foreman grants prior approval.

b. The key issue log or electronic equivalent is updated to show that the key has been checked in then checked out to the new recipient.

7.2.4 Each key should be marked with a unique identification number.

7.2.5 Each key stored in key boxes described in 7.1 should be attached to a numbered tag (typically brass), corresponding to the storage location in the box.

7.2.6 A color-coded second tag shall be attached to each LHRA, and VHRA key. The following colors shall be used:

a. LHRA= yellow tag

b. VHRA=red tag

7.2.7 A key issue log (see attachment 10.1) or electronic equivalent shall be kept for all keys controlled by this procedure.

7.2.8 A key inventory shall be performed at least once per shift and should contain the following information:

a. Key tag number

b. Status (e.g., logged out, downgraded, or missing).

c. Key control (non-rad, LHRA, VHRA).

d. Date and time inventory completed.

e. Name of person performing inventory.

ID: 23099 Which one of the choices correctly identifies the position title of the person or persons who must be notified prior to issuance of Very High Radiation Area key?

Ans Operations Superintendent and Radiation Protection Manager

D1 Operations Manager and Radiation Protection Manager

D2 Radiation Protection Manager Only

D3 Operations Superintendent Only

AbbrevLocName	ExamDate	Vendor	Type	ExamType	Cog Level	ExamLevel	RefMaterial	ParentId
Salem Unit 1	11/4/2002	WEC	PWR	ILO	1	R		

QuestionComment	
Distract1Comment	
Distract2Comment	
Distract3Comment	

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
..G2.3.1			G2	3	1	

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	______
Group #	3	______
K/A #	G2.3.2	
Importance	2.5	2.9

Proposed Question:

The Total Effective Dose Equivalent (TEDE) is defined as:

A. The sum of the Total Organ Dose Equivalent (TODE) and the Shallow Dose Equivalent (SDE)

B. The sum of the Deep Dose Equivalent (DDE) and the Committed Dose Equivalent (CDE)

C. The sum of the Deep Dose Equivalent (DDE) and the Committed Effective Dose Equivalent (CEDE)

D. The sum of the Total Organ Dose Equivalent (TODE) and the Committed Effective Dose Equivalent (CEDE)

Proposed Answer:

C. The sum of the Deep Dose Equivalent (DDE) and the Committed Effective Dose Equivalent (CEDE)

Explanation:
A incorrect, does not use TODE or SDE

B incorrect, TODE = The summed total of the DDE (external whole body exposure) and the CDE (internal dose to the organs or tissues from internal exposure from non-stochastic ALI), when CDE is not zero.

C correct, TEDE = The sum of the DDE (for external exposures) and the CEDE (for internal exposures).

D incorrect, does not use TODE

Technical Reference(s): LEP3, EP RB PROCEDURES

Proposed references to be provided to applicants during examination: none

Learning Objective:
71144 - DEFINE the following:
a. CDE
b. CEDE
c. DDE
d. LDE
e. SDE
f. TEDE
g. Total Organ Dose Equivalent
h. Declared Pregnant Female
i. Restricted Area

Question Source: Bank # P-69970

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 41.12
55.43 43.4

Comments: K/A: G2.3.2 – Knowledge of facility ALARA program.

1 P-69970 Points: 1.00 Multiple Choice

The Total Effective Dose Equivalent (TEDE) is defined as:

A. The sum of the Deep Dose Equivalent (DDE) and the Committed Effective Dose Equivalent (CEDE)

B. The sum of the Deep Dose Equivalent (DDE) and the Shallow Dose Equivalent (SDE)

C. The sum of the Deep Dose Equivalent (DDE) and the Committed Dose Equivalent (CDE)

D. The sum of the Total Organ Dose Equivalent (TODE) and the Committed Effective Dose Equivalent (CEDE)

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

71144	DEFINE the following: a. CDE b. CEDE c. DDE d. LDE e. SDE f. TEDE g. Total Organ Dose Equivalent h. Declared Pregnant Female i. Restricted Area

Reference Id: P-69970
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 1.00
Time to complete: 2
Topic: LEP3 - Define TEDE
Cross Reference: LEP3 Obj 1
Comment: 71144

PERSONNEL EXPOSURE TERMS AND DEFINITIONS

Obj 1 *118]* The following terms and definitions are used to track personnel exposure to prevent violation of Legal Exposure Limits.

Acronym	Term	Definition
CDE	Committed Dose Equivalent	The dose equivalent to an organ that will be received from an intake by an individual during the 50 year period following the intake.
CEDE	Committed Effective Dose Equivalent	The product of the Committed Dose Equivalent and a weighting factor to equate dose to the organ to the equivalent dose (risk) from uniform irradiation of the whole body.
DDE	Deep Dose Equivalent	Dose associated with external exposure of the whole body (depth 1 cm).
LDE	Lens Dose Equivalent	External exposure to the lens of the eye (depth of 0.3 cm).
SDE	Shallow Dose Equivalent	The dose that applies to the external exposure of the skin or any extremity (depth of 0.007 cm).
TEDE	Total Effective Dose Equivalent	The sum of the DDE (for external exposures) and the CEDE (for internal exposures).
TODE	Total Organ Dose Equivalent	The summed total of the DDE (external whole body exposure) and the CDE (internal dose to the organs or tissues from internal exposure from non-stochastic ALI), when CDE is not zero.
DPW	Declared Pregnant Woman	A woman who has voluntarily informed DCPP, in writing, of her pregnancy and the estimated date of conception.
RESTRICTED AREA		The Protected Area of the plant associated within the power block. The Restricted Area does not include Intake Area.

RO QUESTION 73

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	_____
Group #	3	_____
K/A #	G2.3.4	
Importance	2.5	3.1

Proposed Question:

An operator has a Total Effective Dose Equivalent (TEDE) of 4 REM for the current year.

He has been approved to exceed the Administrative Guideline.

Which of the following is the longest the operator can stay in a 100 mR/hr radiation area without exceeding the DCPP Administrative Exposure Limit for the year?

A. None, the Admistrative Guideline limit and Administrative Exposure Limit for a year are the same.

B. 1 hour

C. 5 hours

D. 10 hours

Proposed Answer:

C. 5 hours

Explanation:
DCPP limit is 10% below federal limit – 4.5 rem. Max stay time is 5 hours.
A incorrect, maximum permissible is 4.5 rem.
B incorrect, this would still be under to the limit.
C correct, exposure would be 4.5 rem – the DCPP exposure limit.
D incorrect, this would be 5.0 rem.

Technical Reference(s):
Student handout for Radiation Worker Training – page 45 (rev 1/02)

Proposed references to be provided to applicants during examination: none

Learning Objective: State the DCPP exposure limits and guidelines (including declared pregnant female).

Question Source:
Modified Bank # INPO 22396

Question History: Last NRC Exam DCPP 10/02

Question Cognitive Level:

Memory or Fundamental Knowledge	____
Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41 41.12
55.43

Comments:

K/A: G2.3.4 - Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

OBJECTIVE #C6

State the DCPP exposure limits and guidelines (including declared pregnant female) (Site Specific).

INTRODUCTION

The station exposure limits at DCPP are different from federal limits in that station limits can be raised after review and approval by management.

NEED TO KNOW

DCPP station exposure limits are 90% of the federal limits.

Diablo Canyon Station limits:

TEDE administrative **guideline**	2.0 rem/year
TEDE limit	4.5 rem/year*
Lens of the eye	13.5 rem/year*
Skin	45 rem/year*
Extremity limit	45 rem/year*
Organ limit	45 rem/year*
Declared pregnant worker	ALARA#

*calendar year (January 1st through December 31st)

#As Low As Reasonably Achievable (Definitely less than the federal limit of 0.5 rem/term)

HELP

Because most exposure is from gamma radiation (whole body exposure), the TEDE limit is the most likely to be exceeded. It is also the lowest limit.

ID: 22396 An operator has a Total Effective Dose Equivalent (TEDE) of 4 REM for the current year. He has been approved to exceed the Administrative Guideline of 2 REM for the year.

How long can the operator stay in a 50 mR/hr radiation area without exceeding the DCPP Administrative Exposure Limit for the year?

Ans 10 hours

D1 0 hours

D2 5 hours

D3 20 hours

AbbrevLocName	ExamDate	Vendor	Type	ExamType	Cog Level	ExamLevel	RefMaterial	ParentId
Diablo Canyon 1	10/1/2002	WEC	PWR	ILO	2	R		

QuestionComment	
Distract1Comment	
Distract2Comment	
Distract3Comment	

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
..2.3.4			2	3	4	

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	_____
	Group #	4	_____
	K/A #	G2.4.12	
	Importance	3.4	3.9

Proposed Question:

A LOCA has occurred on Unit 2.

The crew is reviewing the foldout page as part of a procedure transition tailboard.

As a minimum, what amount of repeat back is required by an operator who is assigned a foldout page item to monitor?

A. The high level action.

B. Simple acknowledgement of the assignment.

C. A brief summary of the action and the parameters to monitor.

D. The high level action and the specific parameters and values to monitor.

Proposed Answer:

A. The high level action.

Explanation:

A correct, per OP1.DC11 page 10 (rev 24)…. foldout page should be reviewed with the crew as a part of the procedure transition tailboard. Specific assignments should be made to appropriate control room operators by assigning the foldout page number and the operator repeating back the high level action. Specific parameters and values are not required to be repeated back. A copy of the foldout page should be given to any operator with an assignment.
B incorrect, repeat high level action.
C incorrect, repeat back the high level action.
D incorrect, specific parameters etc not required.

Technical Reference(s): OP1.DC11 - Conduct of Operations - Abnormal Plant Conditions

Proposed references to be provided to applicants during examination: none

Learning Objective: 7922 - Discuss the characteristics of a tailboard

Question Source:
New X_

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 41.10
55.43 _____

Comments:

K/A: G2.4.12 - Knowledge of general operating crew responsibilities during emergency operations.

5.5.11 Procedure steps flagged as "continuous action" items should be assigned to the appropriate control room individual to ensure that the requirements are met if conditions change later in the procedure.

5.5.12 EOP foldout page information should be handled in the following manner:

a. For procedures which have no procedure transition tailboard, the shift foreman and the WCSFM should monitor the foldout page items and notify the control room crew of any actions necessary from the foldout page.

b. For other EOPs, the foldout page should be reviewed with the crew as a part of the procedure transition tailboard. Specific assignments should be made to appropriate control room operators by assigning the foldout page number and the operator repeating back the high level action. Specific parameters and values are not required to be repeated back. A copy of the foldout page should be given to any operator with an assignment.

5.5.13 The critical safety function (CSFs) status trees shall be monitored at the periodicity required by the EOP rules of usage (Attachment 7.1).

a. When the SPDS is in service, PERIODIC (once every 10 to 20 minutes) monitoring of the CSF status trees is performed to verify that no red or magenta paths exist. This should be accomplished by selecting one of the CSF screens and using the displayed CSF numbers to check colors.

b. When the SPDS is in service, CONTINUOUS monitoring of the CSF status trees is performed when a red or magenta path exists (exceptions to this are discussed in Attachment 7.1). This should be accomplished by selecting one of the CSF status trees. The person responsible for CSF monitoring remains in the direct area of the SPDS panels allowing immediate detection of a CSF status change.

c. When the SPDS is out of service, the CSF monitoring function is satisfied by using the sheets provided in EOP F-0. In this case, the monitor should review all the sheets in sequence, and should repeat the review periodically or continuously based upon the CSF status per EOP F-0.

d. When a red or magenta path exists, the redundant instrumentation should be verified prior to transitioning to an FR procedure. The intent of this check is to ensure an FR is not implemented based on inaccurate SPDS indication.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	______
	Group #	4	______
	K/A #	G2.4.45	
	Importance	3.3	3.6

Proposed Question:

The crew has entered E-0, Reactor Trip or Safety Injection. SI has actuated. Immediate actions are complete.

Which of the following alarms should be brought to the attention of the SFM as soon as it is received and reviewed by the operator?

A. AR PK09-11, Feedwater Isolation

B. AR PK09-12, Main Feedwater Pump Trip (RED)

C. AR PK09-18, Turbine Driven Aux FW Pp

D. AR PK12-13, AMSAC Tripped

Proposed Answer:

C. AR PK09-18, Turbine Driven Aux FW Pp

Explanation:

OP1.DC11 states: All alarms received in the control room shall be reviewed for significance. Significant alarms received during the use of AOPs and EOPs shall be brought to the attention of the SFM by the person acknowledging the alarm.
A incorrect, expected alarm.
B incorrect, expected alarm.
C correct, alarm signals trouble with a running AFW pump.
D incorrect, expected alarm.

Proposed references to be provided to applicants during examination: none

Learning Objective: 7951 - Explain the duties on a reactor trip

Technical Reference(s):
OP1.DC11 – Conduct of Operations – Abnormal Plant Conditions
AR PK12-13
AR PK09-18
AR PK09-12
AR PK09-11

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 41.10
55.43

Comments:

K/A: G2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.

5.5.14 In many cases, the AOPs, EOPs, or ARPs refer to normal operating procedures to accomplish various tasks necessary to mitigate a plant event. In these cases, certain precautions, limitations, or procedure steps in the normal operating procedures may not be applicable or desirable during the emergency use of that procedure.

a. If, in the judgment of the shift foreman, time is critical or the operator is knowledgeable about the task being referred to, following consultation with the SM and/or WCSFM the SFM may direct the operator to take actions consistent with these normal operating procedures but using different methodologies or without having the procedure in hand (reference: AD2.ID1).

5.5.15 Use of Annunciator Response Procedures (ARPs)

a. It is vital that the crew use diagnostics, teamwork, and communications to get pertinent information to the SFM.

b. The SFM shall use his/her command and control of the situation and the crew's team skills to prioritize the actions that need to be taken. Depending on plant conditions, these actions may include completion of all or part of the ARP.

c. All alarms received in the control room shall be reviewed for significance. Significant alarms received during the use of AOPs and EOPs shall be brought to the attention of the SFM by the person acknowledging the alarm.

d. The SFM should determine the priority of annunciators and the appropriate response, based on plant conditions or events in progress.

e. The SFM should assign a crewmember the responsibility of monitoring and addressing alarms.

1. The appropriate ARP should be reviewed for each alarm deemed significant. The SFM will determine when this action will occur, based on manpower available and the events in progress.

2. The SFM should be kept updated on any actions performed that could impact the EOP or AOP procedure flowpath.

#

PACIFIC GAS AND ELECTRIC COMPANY	**NUMBER** AR PK12-13
NUCLEAR POWER GENERATION	**REVISION** 10
DIABLO CANYON POWER PLANT	**PAGE** 1 OF 1
ANNUNCIATOR RESPONSE	**UNIT**

TITLE: AMSAC TRIPPED

1

02/20/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
AMSAC K101A K101B	793	AMSAC TRIPPED	

3. PROBABLE CAUSE

   3.1 Turbine Power >32% within the last 4 minutes with 3/4 S/G's <11% narrow range level for >25 seconds.

   3.2 Restoration of power to AMSAC system.

4. AUTOMATIC ACTIONS

   4.1 Trip Main Turbine

   4.2 Start all AFW pumps

   4.3 Blowdown isolation outside containment

   4.4 S/G sample valve isolation

5. OPERATOR ACTIONS

   NOTE: After an AMSAC actuation, the AMSAC generated turbine trip must be manually reset on VB-3 after this alarm clears. This action is required prior to relatching the turbine due to time delays in the circuitry. AFW pumps may be stopped and blowdown isolation valves may be reopened as soon as either actuating signal has cleared.

   5.1 If reactor has tripped, go to EOP E-0.

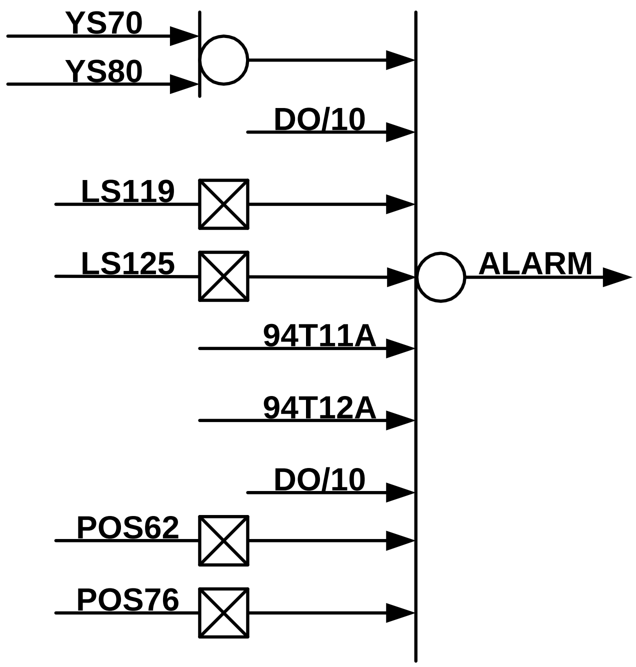
   5.2 If plant conditions require a reactor trip, manually trip the reactor and go to EOP E-0.

TITLE: MAIN FEEDWATER PUMP TRIP (RED)

08/11/04
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM



2.

ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
YS 70	554	FWP Turb 11 & 12 Thrust Brg Wear Trip	20 Mils (2 of 2 probes YT70A and YT70B)
YS 80	554	FWP Turb 11 & 12 Thrust Brg Wear Trip	20 Mils (2 of 2 probes YT80A and YT80B)
LS 119	555	FWP Turb 1-1 L.O. Lvl Lo-Lo Trip	39" Below Flange
LS 125	556	FWP Turb 1-2 L.O. Lvl Lo-Lo Trip	39" Below Flange
POS 62	559	FWP Turb 1-1 Lo Vac Trip	10" to 15" Hg Abs
POS 76	560	FWP Turb 1-2 Lo Vac Trip	10" to 15" Hg Abs
94 T 11A	557	FWP Turb 1-1 Trip	Low Control Oil Pressure: 58.5-61.5 Psig
94 T 12A	558	FWP Turb 1-2 Trip	Low Control Oil Pressure: 58.5-61.5 Psig
DO/10	699	FWP 1-1 Speed Controls Trip	
DO/10	743	FWP 1-2 Speed Controls Trip	

3. PROBABLE CAUSE

3.1 Main F.W.P. Trip due to:

3.1.1 Thrust Brg Wear

3.1.2 Low-Low Lube Oil Tank Level

3.1.3 Low Vacuum

3.1.4 Overspeed

3.1.5 Manual trip

3.1.6 SIS

3.1.7 P-14

3.1.8 Hi Discharge Pressure

3.1.9 HPU Pressure ≤ 100 psig

4. AUTOMATIC ACTIONS

4.1 Possible auto start of AFW pumps.

4.2 Possible Reactor Trip.

4.3 Automatic programmed ramp to 650 MW initiated at 225 MW/Min then a ramp to 550 MW at 25 MW/Min.

5. OPERATOR ACTIONS

5.1 Check annunciator printout and control room instrumentation to verify plant conditions. An overspeed trip can be verified locally, per step 5.4.

5.2 If the reactor trips, go to EOP E-0.

5.3 Go to Abnormal Operating Procedure, OP AP-15.

5.4 An overspeed trip can be verified at the local OPVIEW panel as follows:

5.4.1 At the Woodward control panel on the FWP pedestal, verify the keyswitch is selected to "OPVIEW DISABLED."

5.4.2 Touch the screen to awaken. Verify the message "LOCAL CONTROL Modbus Commands Disabled" appears in the upper left corner of the screen. This indicates the Control Room has pump control.

5.4.3 The cause of the trip should be shown across the bottom of the screen. Optionally, if not selected, touch "Main Menu" to select.

5.4.4 Select "Alarm Log."

5.4.5 The cause of the trip will be shown across the bottom of the screen.

5.5 If a programmed ramp has been initiated:

5.5.1 Verify validity of the ramp.

a. Place ramp on HOLD if not valid.

5.5.2 Implement AP-25.

6. REFERENCES

6.1 501128 Main Annunciator Schematic Diagram

6.2 663300-11 Boiler F.P. Turbine Test Data (POS 62, 76)

6.3 Loop Test 20-70 (YS 70)

6.4 Loop Test 20-80 (YS 80)

6.5 Loop Test 20-22FF (LS-119)

6.6 Loop Test 20-23FF (LS-125)

6.7 Loop Test 20-23K (PS-86; 94T11A)

6.8 Loop Test 20-23K (PS-99; 94T12A)

6.9 437567 FWP Turbine Control Schematic Diagram

TITLE: FEEDWATER ISOLATION

APPROVED: ______________________ **11/07/88** **11/07/88**
DATE **EFFECTIVE DATE**

1. LOGIC DIAGRAM

K 0614 ⟶ **ALARM**

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
K 0614	47	FW Isol From React Trip & Lo Tavg 2/4	LT 554°F

3. PROBABLE CAUSE

3.1 A reactor trip (P-4) with a low TAVG signal, either train A or B.

NOTE: This alarm will not be initiated by a feedwater isolation as a result of stm. gen. Hi-Hi level or an S.I. initiation signal.

4. AUTOMATIC ACTIONS

4.1 Closure of all S/G feedwater control and bypass valves.

5. OPERATOR ACTIONS

5.1 Check annun. printout and control room instrumentation to verify plant conditions.

5.2 With Reactor TRIP, GO TO Emergency Procedure EP E-0.

NOTE: If this alarm occurs, feedwater control valve isolation can only be reset via the two feedwater isolation reset pushbuttons on VB-3.

#

PACIFIC GAS AND ELECTRIC COMPANY | **NUMBER** AR PK09-18
NUCLEAR POWER GENERATION | **REVISION** 13B
DIABLO CANYON POWER PLANT |
ANNUNCIATOR RESPONSE | **UNIT**

TITLE: TURBINE DRIVEN AUX FW PP

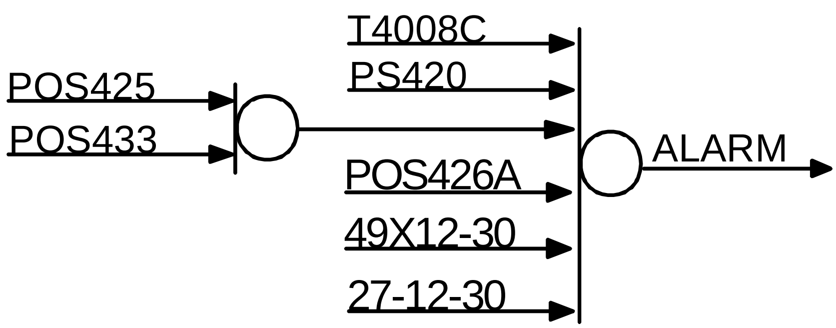
1

09/05/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT	PPC ADDRES S
T4008C	594	Aux FWP 1-1 Temp PPC		
	The inputs to device T4008C are as follows:			
	1. Aux FWP 1-1 Turb Inbd Brg Temp			T2359A
	2. Aux FWP 1-1 Turb Outbd Brg Temp			T2360A
	3. Aux FWP 1-1 Pp Inbd Brg Temp			T2363A
	4. Aux FWP 1-1 Pp Thr Brg Temp			T2365A
PS 420	1061	Aux FWP 1-1 Suct Press Lo	LT 8.5 psig	
POS 425(H)	595	Aux FWP Turb 1-1 FCV-37 or FCV-38 Clsd	Vlv Clsd	
POS 433(F)	595	Aux FWP Turb 1-1 FCV-37 or FCV-38 Clsd	Vlv Clsd	
49X-12-30	1164	Aux FWP Turb 11 FCV-95 OC Trip		

#

#

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT	PPC ADDRES S
27-12-30	1165	Aux FWP Turb 11 FCV-95 Cont UV		
POS 426A	475	Aux Feed Pp Turb 1-1 Stm Inlet FCV-152 Clsd	Vlv Clsd	

3. PROBABLE CAUSE

3.1 Loss of cooling water.

3.2 Low CST level

3.3 Steam supply valve closed.

3.4 Overcurrent trip of FCV-95.

4. AUTOMATIC ACTIONS

None

5. OPERATOR ACTIONS

5.1 If bearing high temperature:

5.1.1 Monitor bearing temperature.

5.1.2 Locally check cooling water supply and adjust as necessary.

5.2 If FCV-152 is closed refer to OP D-1:IV to reset.

5.3 Check the position of FCV-37 or FCV-38.

5.3.1 Closure of FCV-37 or FCV-38 makes the Turbine Driven Auxiliary Feedwater Pump inoperable.

5.4 If low suction pressure:

5.4.1 Check CST level and local valve lineup.

5.4.2 Refer to OP D-1:V for alternate AFW supplies.

5.5 Check power supply for FCV-95.

5.6 Refer to T.S. 3.7.1.2 (ITS 3.7.5).

SRO QUESTION 76

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	______	1
Group #	______	1
K/A #	EPE 011 G2.2.25	
Importance		3.7

Proposed Question:

Technical Specification 3.5.1, Accumulators, applies until the plant is in MODE 3, less than 1000 psig.

Why are the Accumulators allowed to be removed from service below 1000 psig?

A. Nitrogen injection is a larger risk.

B. The probability of a large LOCA with a blowdown phase is sufficiently low.

C. ECCS injection is sufficient to ensure peak clad temperature remains below 2200ºF.

D. Accumulator boron concentration is typically less than required shutdown boron concentration.

Proposed Answer:

C. ECCS injection is sufficient to ensure peak clad temperature remains below 2200ºF.

Explanation:
A incorrect, this is why Accumulators are isolated in many EOPs.
B incorrect, a large LOCA will still have a blowdown phase.
C correct, below 1000 psig, ECCS injection is sufficient to maintain core cooling.
D incorrect, Required boron concentration in an accumulator is higher than any required shutdown boron concentration.

Technical Reference(s):
B 3.5.1 – Accumulators Bases
TS 3.5.1 - Accumulators

Proposed references to be provided to applicants during examination: none

Learning Objective: 9694E - Discuss Technical Specification Bases.

Question Source:

sro tier 1 group 1_76 rev1.doc

New X

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 _____

55.43 43.2

Comments:

K/A: EPE 01 1 G2.2.25 – Large Break LOCA, Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

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.	6 hours
	AND	
	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)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2200 ppm and ≤ 2500 ppm.	31 days AND ---------NOTE------- Only required to be performed for affected accumulators. ------------------------ Once within 6 hours after each solution volume increase of ≥ 5.6% of narrow range indicated level that is not the result of addition from the refueling water storage tank.
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is > 1000 psig.	31 days

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the latter phase of blowdown to the beginning phase of reflood of a loss of coolant accident (LOCA). The ECCS injection mode following a large break LOCA consists of three phases: 1) blowdown, 2) refill, and 3) reflood.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of ECCS water.

The refill phase is complete when the injection of ECCS water has filled the reactor vessel downcomer and the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods.

The reflood phase follows the refill phase and continues until the reactor vessel has been filled to the extent that core temperature rise has been terminated.

The accumulators function in the later stage of blowdown to the beginning of reflood to fill the downcomer and lower plenum. The injection of the ECCS pumps aid during refill. Reflood and the following long term heat removal is accomplished by water pumped into the core by the ECCS pumps.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by an open motor operated isolation valve

(continued)

BACKGROUND (continued)	(8808A, B, C, and D) and by two check valves in series. The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.
APPLICABLE SAFETY ANALYSES	The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1 and 3). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits. In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow with no credit taken for ECCS pump flow until an effective delay has elapsed. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. No operator action is assumed during the blowdown stage of a large break LOCA. The limiting large break LOCA is a double ended guillotine break in the RCS piping. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure. The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The accumulators do not discharge above the pressure of their nitrogen cover gas (579 to 664 psig.) At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) that are applicable for the accumulators will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown and reflood phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46, though their water volume is credited as part of the long term cooling inventory.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume (814 cubic feet to 886 cubic feet) is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty. Depending on the NRC-approved methodology used to analyze large breaks, an increase in water volume may result in either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of ≥814 cubic feet and ≤886 cubic feet. The implementation of these values is performed accounting for instrument uncertainty.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	A reduction below the accumulator LCO minimum boron concentration would produce a subsequent reduction in the available containment recirculation sump boron concentration for post LOCA shutdown and an increase in the sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH. The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure (579 psig), since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit (664 psig) provides margin to assure inadvertent relief valve actuation does not occur. These analysis-assumed pressures are specified in the SRs. Volumes are shown on the control board indicators as % readings on accumulator narrow range level instruments. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained. The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3). The accumulators satisfy Criterion 2 and Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above a nominal pressure of 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at RCS pressures > 1000 psig. At pressures $\leq$ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure $\leq$ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are normally closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. This condition is in agreement with the TS 3.4.12 LCO requirement.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analyses demonstrate that the accumulators will discharge following a large main steam line break. The impact of their discharge is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 hour Completion Time to open the valve, remove

(continued)

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	1
	K/A #	APE 022 G2.1.33	
	Importance		4.0

Proposed Question:

PLANT CONDITIONS:

- Unit 1 is in MODE 6.
- The crew is preparing to fill the refueling cavity - it is estimated it will require 360,000 gallons.
- Both Boric Acid Storage Tanks are inoperable

As a minimum, for the current plant conditions, what must the level in the RWST (in whole %) be to allow the filling of the refueling cavity?

A. 73%

B. 77%

C. 84%

D. 90%

Proposed Answer:

C. 84%

Explanation:
A incorrect, this is what would be required to do the fill, but neglecting the necessary 50,000.
B incorrect, ECG 8.8 bases states the volume is 50,000 gallons but that includes unuable volume and 17,865 gallons is required. 17,864 + 360K = 378 or 77%.
C correct, required volume is 360,0009 gallons (74%) plus 50,000 gallons to have the RWST operable for ECG 8.8. This brings the required volume to 410,000 gallons (84% - 411,381 gallons).
D incorrect, this would be 50K added to the minumum volume at 0%, then added to 360,000K.

Technical Reference(s):
ECG 8.8, Reactivity Control Systems – Borated Water Source – Shutdown
Refueling Water Storage Tank Percent Indicated Vs. Volume, Page IE-9.2b

Proposed references to be provided to applicants during examination:
ECG 8.8, Reactivity Control Systems – Borated Water Source – Shutdown
Refueling Water Storage Tank Percent Indicated Vs. Volume, Page IE-9.2b

Learning Objective: 66041 - Discuss the requirements of System 8 ECGs

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 _____
55.43 43.2/43.3

Comments:

K/A: APE 022 G2.1.33 - Loss of Reactor Coolant Makeup - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

8.0 CHEMICAL AND VOLUME CONTROL SYSTEM

8.8 Reactivity Control Systems - Borated Water Source - Shutdown

ECG 8.8 A minimum of one borated water source shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

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.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No borated water source OPERABLE.	A.1 Suspend all operations involving CORE ALTERATIONS or positive reactivity changes.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---

SR 8.8.1 through SR 8.8.4 shall be current for at least the credited OPERABLE borated water source.

SURVEILLANCE		FREQUENCY
SR 8.8.1	Verify the RWST temperature is within limit when it is the source of borated water and the outside ambient air temperature is less than 35°F.	24 hours
SR 8.8.2	Verify the boron concentration of the water is within limits.	7 days
SR 8.8.3	Verify the contained borated water volume is within limits.	7 days
SR 8.8.4	Verify the boric acid storage tank solution temperature is within limit when it is the source of borated water.	7 days

SURVEILLANCE	FREQUENCY

BASES

BACKGROUND	ECG 8.8 was developed to relocate the requirements from current TS (CTS) 3/4.1.2.5 to the ECGs as approved by Reference 1. The boration subsystem of the chemical and volume control system (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e. to control the chemical neutron absorber (boron) concentration in the reactor coolant system (RCS) and to help control the boron concentration to maintain shutdown margin (SDM). To accomplish this functional requirement, the boration systems require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head. The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from an OPERABLE diesel generator.
APPLICABLE SAFETY ANALYSES	Final Safety Analysis Report (FSAR) Section 9.3.4.2.2.4, “Chemical Shim and Reactor Coolant Makeup,” describes the design bases of the CVCS chemical shim (boric acid) and reactor coolant makeup system. FSAR Section 15.2.4, “Uncontrolled Boron Dilution,” provides an analysis of the uncontrolled boron dilution accident. FSAR Section 9.3.4, “Chemical And Volume Control System (CVCS),” states that the boron injection system, for cold shutdown conditions, is designed to increase the RCS boron concentration to the cold shutdown concentration. It is capable of borating the RCS through either one of two flowpaths and from either one of two boric acid sources. The amount of boric acid stored in the CVCS always exceeds that amount required to borate the RCS to cold shutdown concentration assuming that the control assembly with the highest reactivity worth is stuck in its fully withdrawn position. (continued)

SURVEILLANCE	FREQUENCY

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reference 2 states that the boration subsystem is not assumed to operate to mitigate the consequences of a design basis accident or transient. In the case of a malfunction of the CVCS, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event. Furthermore, Reference 3 notes that the normal capability to control reactivity with boron is not credited in the accident analysis.

Reference 4 states with the RCS temperature below 200°F, one boron injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The boron capability required below 200°F is sufficient to provide an SDM of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,499 gallons of 7,000 ppm borated water from the boric acid storage tanks or 17,865 gallons of 2,300 ppm borated water from the RWST.

LCO

The LCO requires at least one of two sources of borated water to be OPERABLE. The acceptable sources and limits are:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 2,499 gallons,
2) A boron concentration between 7,000 and 7,700 ppm, and
3) A minimum solution temperature of 65°F.

b. The Refueling Water Storage Tank (RWST) with;

1) A minimum contained borated water volume of 50,000 gallons,
2) A minimum boron concentration of 2300 ppm, and
3) A minimum solution temperature of 35°F.

The contained water volume limits for each source include allowance for water not available because of discharge line location and other physical characteristics (Reference 4).

(continued)

BASES

APPLICABILITY

ECG 8.8 is only applicable in MODES 5 and 6 (RCS temperature ≤ 200°F). When RCS temperature is below 200°F, only one boron injection flow path is required, without single failure consideration, as discussed in the safety analyses section above. For operation in MODES 1 - 4, refer to ECG 8.4, which requires a minimum of two boron injection flow paths to be OPERABLE to ensure functional capability in the event an assumed failure renders one of the flow paths inoperable (Reference 4).

ACTIONS

A.1

Condition A applies when none of the borated water sources are OPERABLE. With this condition, Action A.1 requires that all operations involving CORE ALTERATIONS or positive reactivity changes be suspended immediately. This will preclude any positive reactivity changes. With no source for injection of borated water, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

SURVEILLANCE REQUIREMENTS

SR 8.8.1 through SR 8.8.4 are required to be current for at least the credited OPERABLE borated water source.

SR 8.8.1

This surveillance verifies that the RWST solution temperature is not less than 35°F when it is the source of borated water and the outside ambient air temperature is less than 35°F. This assures that the boric acid solution remains soluble. The 24 hour frequency is appropriate based on the large volume of solution that has to undergo the temperature change and on engineering judgment.

SR 8.8.2 through 8.8.4

These surveillances verify that the boron concentration, borated water volume, and temperature of the boric acid storage tank, when it is the source of borated water, are within limits specified by the LCO Bases. The 7 day Frequency has been shown to be acceptable through operating experience.

(continued)

BASES

REFERENCES

1. License Amendments 135 (Unit 1) and 135 (Unit 2) dated May 28, 1999.
2. License Amendment Request 97-09, Attachment 21, page 7 (PG&E Letter DCL-97-106 dated June 2, 1997)
3. J. D. Andrachek, et. Al., Methodically Engineered, Restructured, and Improved Technical Specifications, (MERITS) Program - Phase II Task 5, Criteria Application, WCAP-11618, dated November 1987.
4. CTS Bases 3/4.1.2 (relocated to ECG 8.8 Bases).

6/27/2000
Effective Date

REFUELING WATER STORAGE TANK
PERCENT INDICATED VS. VOLUME

%	Gallons	%	Gallons	%	Gallons	%	Gallons
0	26438	25	141004	50	255571	75	370137
1	31021	26	145587	51	260153	76	374719
2	35603	27	150170	52	264736	77	379302
3	40186	28	154752	53	269318	78	383885
4	44769	29	159335	54	273901	79	388467
5	49351	30	163918	55	278484	80	393050
6	53934	31	168500	56	283066	81	397633
7	58517	32	173083	57	287649	82	402215
8	63099	33	177665	58	292232	83	406798
9	67682	34	182248	59	296814	84	411381
10	72265	35	186831	60	301397	85	415963
11	76847	36	191413	61	305980	86	420546
12	81430	37	195996	62	310562	87	425129
13	86012	38	200579	63	315145	88	429711
14	90595	39	205161	64	319728	89	434294
15	95178	40	209744	65	324310	90	438877
16	99760	41	214327	66	328893	91	443459
17	104343	42	218909	67	333476	92	448042
18	108926	43	223492	68	338058	93	452624
19	113508	44	228075	69	342641	94	457207
20	118091	45	232657	70	347224	95	461790
21	122674	46	237240	71	351806	96	466372
22	127256	47	241823	72	356389	97	470955
23	131839	48	246405	73	360971	98	475538
24	136422	49	250988	74	365554	99	480120
						100	484703

Total Contained Volume: 484703 gallons
Unmeasured Volume: 26438 gallons
Unusable Volume: 19780 gallons

SRO QUESTION 78

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	1
	K/A #	APE 025 AA2.02	
	Importance		3.8

Proposed Question:

PLANT CONDITIONS:

- Unit 1 is in MODE 5.
- RHR pump 1-1 is in service.

PK11-21, High Radiation alarms. A few minutes later, PK02-16, RHR System and PK02-17 RHR Pumps also go into alarm.

Which of the following procedures should the SFM utilize to address the current plant conditions?

A. AP-1, Excessive Reactor Coolant System Leakage.

B. AP-16, Malfunction of the RHR System.

C. AP-24, Shutdown LOCA.

D. AP SD-2, Loss of RCS Inventory.

Proposed Answer:

D. AP SD-2, Loss of RCS Inventory.

Explanation:
A incorrect, not appropriate in MODE 5 (applies in MODEs 1 - 4).
B incorrect, not applicable in MODE 5 or appropriate if there is a loss of RCS inventory.
C incorrect, not applicable in MODE 5.
D correct, this is the procedure to use for loss of RCS inventory in MODE 5.

Technical Reference(s):
PK02-16, PK02-17, PK11-21
OP AP-24, OP AP SD-2, OP AP -1, OP AP-16

Proposed references to be provided to applicants during examination: none

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures

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 ______
55.43 43.5

Comments:

K/A: APE 025 AA2.02 - Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere

TITLE: Malfunction of the RHR System

09/04/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. SCOPE

1.1 This procedure covers the steps to be taken following malfunction of the Residual Heat Removal system due to a loss of flow or leakage while in Mode 4.

1.2 OP AP-24, "Shutdown LOCA," provides the action necessary for maintaining core cooling and protecting the reactor core in the event of an RCS leak while on Residual Heat Removal. This procedure may be implemented from OP AP-24 to provide diagnosis for and corrective actions for leaks occurring in systems other than the RCS that cause a loss of RCS inventory.

2. SYMPTOMS

2.1 Abnormal fluctuations in RHR flow on FI-970A & B and/or FI-971A & B.

2.2 Increasing level indication in the PRT.

2.3 Increasing level indication in CCW Surge Tank.

2.4 Possible Annunciator Alarms

2.4.1 CCW SYS SURGE TK LVL/MK-UP (PK01-07)

1) CCW Surge Tk Level Hi/Lo

2.4.2 RHR SYSTEM (PK02-16)

1) RHR Suction Valve Open

2) RHR Pp _____ Room Sump Lvl Hi

3) RHR Pp _____ Disch Press Hi

4) RHR Pp 2-1 and 2-2 Disch Flow Low

2.4.3 RHR PUMPS (PK02-17)

1) RHR Pps OC Trip

2) RHR Pp _____ Room Sump Pps run

2.4.4 HIGH RADIATION (PK11-21)

1) Contmt Area Mon High Rad RE-2

2) Process Mon Hi-Rad (RE-17A & B)

TITLE: Excessive Reactor Coolant System Leakage

08/08/01
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 (TR-26, PR-933; 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).

Loss of RCS Inventory

11/26/04
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure provides guidance for locating and isolating an RCS leak in Modes 5 and 6.

1.1.1 Steps 1 thru 22 are to be used if RCS level is less than 108' or if level is less than 111' and dropping rapidly such that loss of RHR is imminent.

a. Step 2 checks for cavitation, or possible cavitation, of the RHR pumps and reduces flow or stops the pumps to stop the cavitation.

b. Step 6 checks for a LOCA from an RHR relief valve.

c. Steps 7 thru 10 add water from the RWST and restart the RHR pumps in the injection mode.

d. Steps 11 thru 13 are used to vent the RHR System.

e. Steps 14 thru 22 are plant stabilization actions.

1.1.2 If RCS level is greater than 108' and inventory loss is manageable, the inventory loss reduction, isolation and recovery actions begin at step 24.

a. Steps 24 thru 27 shut down the RCPs, if running, and reduce RCS pressure to reduce the inventory loss.

b. Step 28 restores inventory.

c. Steps 29 thru 33 check for the location of the leak by monitoring tank levels, doing local checks and by isolating RHR or the charging and letdown lines.

d. If the leak has not been found by this time the leak is assumed to be on an RHR train, steps 34 thru 36 are used to isolate a leaking RHR train.

1.2 This procedure also provides guidance to restore RCS inventory.

1.3 This procedure covers the steps required to regain RHR cooling after a loss of inventory and subsequent loss of RHR pumps while at reduced RCS inventory conditions (Reactor Vessel level less than 111').

2.

TITLE: SHUTDOWN LOCA

10/02/97
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

This procedure provides actions for protecting the reactor core in the event of a Loss of Coolant Accident (LOCA) that occurs during either Mode 3 after the accumulators are isolated or Mode 4.

2. SYMPTOMS

2.1 The following symptoms are indicative of a Loss of Coolant Accident (LOCA) during Modes 3 and 4:

2.1.1 Uncontrolled decrease in PZR level

2.1.2 Uncontrolled decrease in RCS subcooling

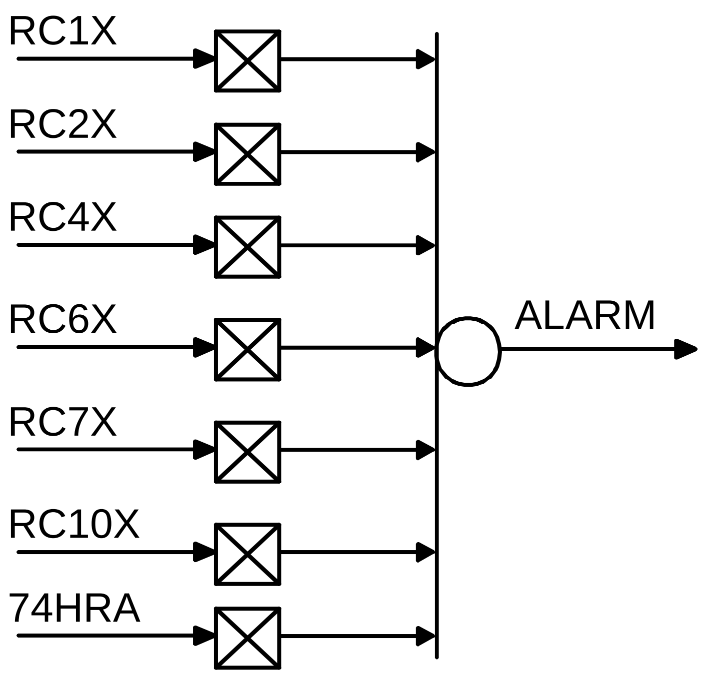
TITLE: HIGH RADIATION

1

10/08/03
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
RC 1X	662	Control Room Area Mon High Rad RE-1	NOTE
RC 2X	663	Contmt Area Mon High Rad RE-2	NOTE
RC 4X	664	Charging PP Area Mon High Rad RE-4	NOTE
RC 10X	665	Aux Bldg Cont Bd Area Mon Hi Rad RE-10	NOTE
RC 6X	668	Pri Sampling Rm Area Mon Hi Rad RE-6	NOTE
RC 7X	669	In-Core Seal Table Area Mon Hi Rad RE-7	50 mR/HR
74 HRA	1066	Process Monitor Hi-Rad (See Step 5.3.1 for listing of monitors)	

NOTE: Refer to Volume 9 in the Control Room for set points.

3. PROBABLE CAUSE

3.1 High radiation sensed by the detector.

3.2 Surveillance test in progress.

3.3 Instrument failed high.

4. AUTOMATIC ACTIONS

4.1 RE-1, 2, 3, 4, 6, 7, 10, 11, 12 and 13 - none.

4.2 RE-44A and 44B - containment ventilation isolation.

NOTE: If CVI bypass switch is selected to BYPASS, CVI actuation is disabled for that RM.

4.3 RE-17A and 17B - isolate CCW surge tank vent.

4.4 RE-18 - isolates discharge to outfall and opens recirc to EDR.

4.5 RE-22 - isolates gas decay tank discharge to plant vent.

5. OPERATOR ACTIONS

5.1 Check annunciator printout.

5.2 For all high radiation alarms, investigate the cause of the alarm.

5.2.1 If alarm is expected or spurious then Reset by performing the following:

a. Notify CO of expected alarm during reset, PK11-23 input 1068.

b. Check RE reading below high alarm setpoint.

c. Place the RE Operation Selector switch to Reset.

d. Place the RE Operational Selector switch to Operate.

e. Check the high alarm is reset on the radiation monitor and PK.

5.3 If a process monitor high rad alarm. **Alarm Input 1066**

5.3.1 Check process monitors RE-3, 11, 12, 13, 17A, 17B, 18, 22, 44A and 44B.

NOTE: If the digital display of a digital radiation monitor is overranged the LED's will display "E.EEE+E."

5.3.2 If RM-13, RM-44A, and/or RM-44B is valid, and is NOT due to Sampling or pre-planned evolution by Radiation Protection/Engineering:

a. At the POV-1 and POV-2 panels place the "S" Signal Test switch to the "S" Signal Test position.

b. At VB-4 place the Aux Bldg Vent Char Fltr Prehtr Control switch to ON.

5.3.3 If the process monitor high rad alarm was already in, check the high rad reflash module to see which new alarm came in or check rad monitor panels for alarms.

5.3.4 If automatic actions are required for the rad monitor, verify appropriate actions have taken place.

5.4 If RE-17A or 17B is in alarm, check CCW surge tank level (LI-139 and LI-140). If leakage into the CCW System is suspected, GO TO OP AP-11, "CCW System Inleakage," section.

5.5 If the RM-44A or B detector is overranged:

5.5.1 The following actions will occur:

a. The digital display will fail to "E.EEE+E."

b. At the Control Room panel the ALERT, HIGH, and FAIL/ACK lights will be lit.

c. Containment Ventilation Isolation will occur.

d. At the local panel the HIGH VOLTAGE DISABLE light will be lit.

e. The anti-jam circuit fuse opens disabling the detector.

5.5.2 Notify MS

5.6 If RM-12 is out of service or a CVI occurs, refer to PK01-17 for placing the CFCU collection system in service.

5.7 To restart RE 11 or 13 pump perform the following:

5.7.1 Verify Sample Selector position: RE 11 normally selected to Main Sample. RE 13 normally selected to Ducts 1 & 2.

5.7.2 Verify High Pressure alarm reset, red alarm light off. To reset high alarm Place High Pressure switch to Reset and then return to Operate.

5.7.3 Verify Paper Drive Selector to Operate.

5.7.4 Place Pump Power Selector to Start and hold until low flow alarm clears. Allow switch to spring return to Local (normal setting).

5.7.5 Check associated alarms and AR PKs clear.

5.8 If RM-7 is in alarm, **Alarm Input 669**, and an incore flux map is not in progress, then perform the following:

5.8.1 Check the MIDS power distribution panel for a seal table leakage alarm. Refer to OP O-21.

5.8.2 Contact the Chemistry Foreman, Radiation Protection Foreman, and the Reactor Engineering Section to inform them of a potential seal table leak.

5.8.3 Upon notification from the Reactor Engineer, increase monitoring of RM-7 for approximately 24 hours for any continued alarm after storage of the neutron detectors.

5.9 If a tank rupture is suspected GO to OP AP-14. **Alarm Inputs 664, 665 & 668**

5.10 Inform Chemistry and Rad Protection Foremen of any unusual condition.

5.11 Refer to R-2 if high airborne is suspected.

5.12 Refer to EP G-1 and/or XI1.ID2 for reportability requirements.

TITLE: RHR PUMPS

1

06/12/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
51X-HG-8	455	RHR PPS OC Trip	Min 67.5 AMPS Inst 645 AMPS
51X-HH-11	455	RHR PPS OC Trip	Inst 645 AMPS
TIS 170	1089	RHR PP 1-1 Mtr Brg Temp Hi	GT 180°F
TIS 171	1089	RHR PP 1-1 Mtr Brg Temp Hi	GT 185°F
TIS 172	1090	RHR PP 1-2 Mtr Brg Temp Hi	GT l85°F
TIS 173	1090	RHR PP 1-2 Mtr Brg Temp Hi	GT l85°F
42-1F-6	457	RHR PP 1-1 Room Sump PPS run	Contactor Clsd
42-1G-6	457	RHR PP 1-1 Room Sump PPS run	Contactor Clsd
42-1F-15	458	RHR PP 1-2 Room Sump PPS run	Contactor Clsd
42-1G-18	458	RHR PP 1-2 Room Sump PPS run	Contactor Clsd
T4500C	454	RHR PPS Temp PPC	

The inputs to device T4500C are:	PPC ADDRESS
1. RHR PP 1-1 STATOR TEMP	TO659A
2. RHR PP 1-2 STATOR TEMP	T0665A

3. PROBABLE CAUSE

3.1 RHR pump OC trip.

3.2 RHR pump Hi motor bearing temp.

3.3 RHR pump Hi stator temp.

3.4 RHR pump room sump pump start.

4. AUTOMATIC ACTIONS

4.1 Possible RHR pump trip.

5. OPERATOR ACTIONS

5.1 Check main annunciator printout.

5.2 Check stability of operating RHR pumps (presence of OC lights, pump pressures, and flows).

5.3 If alarm is due to a pump trip:

5.3.1 Monitor operation of remaining pump closely if running and refer to OP AP-16, AP SD-2 or AP SD-5.

5.3.2 Send an operator to check physical condition of pump and motor if conditions permit.

5.3.3 Except in an emergency, do not attempt a restart of the pump until it has been check electrically.

5.3.4 Refer to Step 5.7.1 for Technical Specification (Tech Spec) limitations.

5.4 If alarm is due to high motor bearing temperature.

5.4.1 Send an operator to check motor bearing temperatures (panel BTK-109 in outer hallway) and bearing oil conditions locally if conditions permit.

5.4.2 Check for proper pump area ventilation.

5.4.3 An RHR pump can be run with bearing temperatures up to 180°F without affecting the lifetime of the component. If this temperature is reached or exceeded then record how long and by how far the limit was exceeded and initiate an Action Request.

5.4.4 If pump operation should not continue, refer to Step 5.7.1 for Tech Spec limitations and refer to OP AP-16, AP SD-2 or AP SD-5.

5.5 If alarm is due to high stator temp.:

5.5.1 Check stator temperature indication on PPC.

5.5.2 Check for excessive motor current.

5.5.3 Check for proper pump area ventilation.

5.5.4 An RHR pump can be run with its stator temperature up to 248°F without affecting the lifetime of the component. If this temperature is reached or exceeded then record how long and by how far the limit was exceeded and initiate an Action Request.

5.5.5 Refer to Step 5.7.1 for Tech Spec limitations.

5.6 If alarm is due to RHR room sump pump operation:

5.6.1 Have an operator check sump pump operation.

5.6.2 Send an operator to check RHR pump room for source of water if conditions permit.

5.6.3 If sump pump operation is due to RHR pump seal failure, determine if pump should be cleared. If so, refer to Step 5.7 below.

5.7 If an RHR pump must be cleared:

5.7.1 Refer to Tech Spec:

a. In Modes 1, 2 and 3, refer to Tech Spec 3.5.2 (ITS 3.5.2).

b. In Mode 4 refer to Tech Spec 3.5.3 (ITS 3.5.3) and 3.4.1.3 (ITS 3.4.6).

c. In Mode 5 refer to Tech Spec 3.4.1.4.1 (ITS 3.4.7) and 3.4.1.4.2 (ITS 3.4.8).

d. In Mode 6 refer to Tech Spec 3.9.8.1 (ITS 3.9.5) and 3.9.8.2 (ITS 3.9.6).

5.7.2 Refer to OP B-2:III to clear an RHR pump.

TITLE: RHR SYSTEM

1

03/07/03
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
PC 405BX POS 674X	57	RHR Suction Valve Open	435 PSIG
PC 403BX POS 669X	57	RHR Suction Valve Open	435 PSIG
LS 426	246	RHR PP 1-1 Room Sump Lvl Hi	GT 21" H_2O
LS 430	l72	RHR PP 1-2 Room Sump Lvl Hi	GT 21" H_2O
PC 647	362	RHR PP 1-2 Disch Press Hi	GT 600 PSIG
PC 635 FIS 641AX FIS 641BX	1412 1078	RHR PP 1-1 Disch Press Hi RHR PP 1-1 and 1-2 Disch Flow Low	GT 600 PSIG LT 475 GPM and GT 20 SEC

PROBABLE CAUSE

3.1 RHR Pump Room Hi sump level.

3.2 RHR Pump Hi discharge pressure.

3.2.1 RHR flow less than 3000 gpm per train.

3.3 RHR Pump discharge flow low.

3.3.1 Failure of flow sensor, or valve closed.

3.3.2 Closure of valves 8701/8702.

3.3.3 Low RCS level during refueling.

3.3.4 Low frequency on vital 4KV Bus G or H.

3.4 8701 or 8702 not fully closed during RCS pressurization.

3.5 RCS check valve backleakage.

4. AUTOMATIC ACTIONS

4.1 A Hi sump level will start one or more RHR Pump Room sump pumps and bring in PK02-17.

4.2 Possible lifting of RHR System relief valves.

5. OPERATOR ACTIONS

****

CAUTION 1: IF RHR discharge pressure is high, pressure should be reduced below 600 psig within 20 minutes to avoid damage to the pump seals.

****

****

CAUTION 2: RHR and CCW System piping and heat exchangers are susceptible to waterhammer. Care should be taken to avoid situations which could lead to waterhammer and a subsequent loss of system integrity. These situations include:

- The formation of steam voids due to insufficient CCW flow through the RHR heat exchanger when the RHR System temperature is high,
- The formation of steam voids due to rapid depressurization of a large volume of water at a high temperature,
- Subsequent collapse of steam voids when cold water is introduced to the system,
- Subsequent collapse of steam voids when the system is rapidly repressurized.

****

NOTE: Refer to OP AP-16 for RHR malfunctions while in MODE 4.

5.1 Check main annunciator printout to determine which condition initiated alarm.

5.2 High RHR discharge pressure with RCS Solid: **Alarm Input 362, 1412**

5.2.1 Verify greater than 3000 gpm flow for each RHR train in service. Increase flow as necessary.

5.2.2 Verify OPEN or fully OPEN HCV-133.

5.2.3 Take Manual Control of PCV-135 and OPEN to reduce RHR/RCS pressure, if necessary.

5.2.4 Verify RCS low pressure protection is "CUT-IN" on Pressurizer PORVs, PCV-455C and PCV-456.

5.2.5 OPEN Pressurizer PORVs as necessary to reduce RHR/RCS pressure to desired operating band.

5.2.6 If RHR System relief valves have lifted, verify they reclosed when system pressure returns to normal.

5.2.7 Go to Step 5.4.

5.3 High RHR discharge pressure with a bubble in the Pressurizer: **Alarm Input 362, 1412**

5.3.1 Shutdown the RHR Pump in service.

5.3.2 Reduce RHR/RCS pressure to desired operating band with PZR auxiliary spray or PZR PORVs, as necessary.

5.3.3 If RHR System pressure is increasing while performing RCS pressurization, then,

a. Stop RCS pressurization.

b. Decrease RCS pressure as necessary to reduce RHR pressure below 600 psig.

c. Contact the Engineering Section to determine if check valve backleakage is occurring.

d. Contact Plant Management for further guidance to seat the check valve.

5.3.4 If RHR System pressure increases after the RCS is at normal operating pressure, contact Plant Management for further guidance to depressurize the RHR System.

5.3.5 If RHR System relief valves have lifted, verify they reclosed when system pressure returns to normal.

5.4 If RHR System pressure exceeded 600 psig, perform a system inspection to verify RHR System integrity. **Alarm Input 363, 1412**

5.4.1 Write an AR for an engineering evaluation of system operability.

5.5 If alarm is due to a Hi sump level, **Alarm Input 172, 246**

5.5.1 Have an operator check sump pump operation.

5.5.2 Check RHR Pump Room for level and source of water.

5.5.3 If an RHR Pump seal has failed, refer to OP B-2 to clear RHR Pump.

5.5.4 If an RHR Pump must be cleared, refer to Technical Specifications (Tech Specs) for limitations on plant operation.

a. In Modes 1, 2 and 3 refer to Tech Spec 3.5.2.

b. In Mode 4 refer to Tech Spec 3.5.3 and 3.4.6.

c. In Mode 5 refer to Tech Spec 3.4.7 or 3.4.8 as appropriate.

d. In Mode 6 refer to Tech Spec 3.9.5 or 3.9.6 as appropriate.

5.6 If the RHR low flow alarm is received; **Alarm Input 1078**

5.6.1 Check for low frequency on vital 4KV Bus G or H.

5.6.2 Verify the running RHR pump's recirc valve (FCV-641A or B) is in AUTO or OPEN.

5.6.3 <u>GO TO</u> OP AP SD-2 or SD-5.

5.7 If the RHR suction valve open alarm is received, **Alarm Input 57**

5.7.1 If the RHR System is in service, stop the RCS pressurization.

5.7.2 If the RHR System is not in service, shut 8701 and 8702.

SRO QUESTION 79

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	______	3
	Group #	______	1
	K/A #	APE 026 AA2.06	
	Importance		3.1

Proposed Question:

Unit 1 is at full power when the running ASW pumps trip and no other ASW pumps can be started. CCW heat exchanger outlet temperature is 95ºF.

Which of the following describes the approximate time limit for establishing backup cooling to one CCP and the concern if it is not established in time?

A. 1 hour – loss of integrity to RCP seals.

B. 1 hour – damage to the seals of the CCPs.

C. 4 hours – loss of integrity to RCP seals.

D. 4 hours – damage to the seals of the CCPs.

Proposed Answer:

A. 1 hour – loss of integrity to RCP seals.

Explanation:
A total loss of ASW will send the operator from AP-10, LOSS OF AUXILIARY SALTWATER, which will send the crew to AP-11 when no ASW pumps are available
A correct, per Appendix C of AP-11, Without either seal injection or thermal barrier cooling the integrity of the RCP seals will be lost in approximately one hour.
B incorrect, the concern is the RCP seal package.
C incorrect, the time is one hour.
D incorrect, the concern is the RCP seal package.

Technical Reference(s): AP-11, appendix C`

Proposed references to be provided to applicants during examination: none

Learning Objective:
3502, 3495, 5664 - Explain the alternate cooling lineup for the CCP using fire water and how it affects the CCP cooling operations.

Question Source:

New X

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 ______

55.43 43.5

Comments:

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

APPENDIX C

BACKUP COOLING TO A CENTRIFUGAL CHARGING PUMP

1. SCOPE

1.1 The purpose of this appendix is to make available a cooling water supply to one centrifugal charging pump (lube oil and gear oil coolers) when CCW cooling has been lost. Without either seal injection or thermal barrier cooling the integrity of the RCP seals will be lost in approximately one hour.

2. DISCUSSION

2.1 The backup cooling system can be connected to either centrifugal charging pump, however, this system can only be used to supply cooling for one charging pump at a time. The cooling water will come from a fire sprinkler drain with a quick disconnect fitting. The hoses to be used have quick disconnect fittings. The backup cooling hoses are stored near the #3 CCW pump. Each hose is labeled with a letter designation corresponding to the specified hose in the following instructions.

3. INSTRUCTIONS

NOTE 1: Unit 2 valve numbers are in parentheses.

NOTE 2: If difficulty is encountered connecting quick disconnect fittings it may be due to pressure buildup behind the quick disconnect. To relieve the pressure, tap the internal center pin of the quick disconnect fitting using the "quick disconnect adjuster" (rod and hammer) stored with the hoses.

3.1 Connect one end of the 2"x34' hose (Hose 'A') at FP-1-347 (FP-2-350) (sprinkler test valve near the #3 CCW pump room entrance) and connect other end to connection on fire door B21 (B39-2) (outside entry to CCP room).

3.2 Connect one end of the 2"x51' hose (Hose 'B') at fire door B21 (B39-2) (inside entry of CCP room), and connect other end at FP-1(2)-1295 (manifold on wall of CCP room).

3.3 Connect the 1"x12' hose from the manifold outlet to the CCP oil coolers:

NOTE: The hose can only be connected to one CCP.

	CCP 1-1	CCP 1-2	CCP 2-1	CCP 2-2
oil coolers (Hose 'C')	FP-1-1303 to CCW-1-700	FP-1-1303 to CCW-1-199	FP-2-1303 to CCW-2-700	FP-2-1303 to CCW-2-199

3.4 Connect the 1"x12' hose from the CCP oil coolers to the floor drain:

NOTE 1: This hose has a quick disconnect fitting at one end.

NOTE 2: The CCP 1-1 pump oil cooler drain hose is 1"x27'.

	CCP 1-1	CCP 1-2	CCP 2-1	CCP 2-2
oil coolers (Hose 'D')	CVCS-1-488A to drain	CCW-1-195 to drain	CCW-2-582 to drain	CCW-2-195 to drain

SRO QUESTION 80

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	_____	3
Group #	_____	1
K/A #	APE 054 AA2.01	
Importance		4.4

Proposed Question:

While at 45% power, a main feed reg valve failure causes level in one steam generator to rise to 80%.

Which of the following actions should be implemented by the SFM?

A. Direct the operator to trip the reactor and enter E-0, Reactor Trip or Safety Injection.

B. Enter AP-25, Rapid Load Reduction.

C. Enter AP-29, Main Turbine Malfunction.

D. Enter AP-2, Full Load Rejection.

Proposed Answer:

A. Direct the operator to trip the reactor and enter E-0, Reactor Trip or Safety Injection.

Explanation:
A correct, exceeding P-14, causes a turbine trip, the running MFP to trip and FWI. The loss of feed will cause levels in the other steam generators to drop, the reactor will have to be tripped.
B incorrect, cannot reduce load, must trip the reactor.
C incorrect, the turbine will trip but the reactor must be tripped.
D incorrect, the turbine trip will cause a loss of load, but the reactor must be tripped.

Technical Reference(s): B-6A, Reactor Protection

Proposed references to be provided to applicants during examination: none

Learning Objective: 7373, 8181, 8182, 3308, 3302, 3301 - State the setpoints for all Reactor Trips, control and protection interlocks and ESFAS actuation signals.

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 ______

55.43 43.5

Comments:

K/A: APE 054 AA2.01 - Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Occurrence of reactor and/or turbine trip

Turbine Trip

Purpose
Obj 11, 12

The purpose of the Turbine Trip/Reactor Trip is to sense a mismatch between turbine load and reactor power to minimize the resultant thermal transient if the reactor was not tripped.

Description, setpoint
Obj 13

A turbine trip is sensed by a decrease in turbine auto stop oil pressure or by a limit switch on the full closure of the turbine stop valves. A [two out of three pressure switches sensing low auto stop oil pressure], ≤ 50 psig, or [four of four turbine stop valves full closed] is required to trip the reactor.

- This trip is blocked below 50% reactor power.

Logic

The following logic applies to the Turbine Trip/Reactor Trip.

IF	AND	THEN
low auto stop oil pressure is sensed on 2 of 3 pressure switches	power is above P-9, 50% reactor power	a Turbine Trip/Reactor Trip signal is generated to trip the reactor. • Any switch reading low pressure will activate an alarm to indicate a switch is tripped.
	power is below P-9	the reactor will not trip.
full closed is sensed on all four turbine stop valves	power is above P-9, 50% reactor power	a Turbine Trip/Reactor Trip signal is generated to trip the reactor. • Any valve position switch reading full closed will activate an alarm to indicate a switch is tripped.
	power is below P-9	the reactor will not trip.

Continued on next page

Feedwater Isolation Actuation

Purpose
Obj 25

The purpose of the Feedwater Isolation (FWI) Actuation signal is to provide isolation of feedwater within 7 seconds to the S/Gs to prevent:[1]

- S/G overfeeding.
- excessive RCS cooldown and positive reactivity addition.
- feeding a high energy line break inside the containment.

Description, setpoint
Obj 13

FWI Actuation signal is generated by:

- P-14, S/G High-High level of [≥ 75%] sensed by two of three level detectors in any one S/G.
- any SI signal.
- P-4, Reactor Trip with T_{avg} [≤ 554°F] sensed in two of four loops.

Controls
Obj 32, 33, 34

The following controls are available for the FWI Actuation signal.

In the Control Room on VB-3:

Control	Operation
RESET FWI Train A	pushbutton
RESET FWI Train B	

Logic
Obj 27, 29, 36

The following logic applies to the FWI Actuation signal.

IF	Then
P-14 OR any SI signal is generated	• the Main Turbine is tripped. • both Main FW Pump Turbines are tripped. ♦ activated by SSPS Train B only. • all Feedwater Isolation valves are closed. ♦ activated by SSPS Train A only. • all Feedwater Regulating valves are closed. • all Feedwater Bypass valves are closed. • PK09-11, Feedwater Isolation will alarm. • Red light for above Monitor Light Box C for FWI and S/G come on.[2]

Continued on next page

SRO QUESTION 81

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	______	1
Group #	______	1
K/A #	APE 057 AA2.14	
Importance		3.6

<u>Proposed Question</u>:

Unit 1 is at full power.

AR PK19-19, UPS Failure alarms. The CO reports inputs 1503 and 1505 are in.

Which of the following describes the status of inverter 1-4 and the vital AC instrument bus 14?

A. Both the inverter and the vital AC instrument bus are OPERABLE.

B. The inverter is OPERABLE; the vital AC instrument bus is inoperable.

C. The inverter is inoperable; the vital AC instrument bus is OPERABLE.

D. Both the inverter and vital AC instrument bus are inoperable. The vital AC instrument bus will be OPERABLE if the inverter can be transferred to its backup.

Proposed Answer:

C. The inverter is inoperable; the vital AC instrument bus is OPERABLE.

Explanation:
A incorrect, for the inverter to be OPERABLE, it must be powered from its normal source or from the DC source.
B incorrect, the inverter has transferred to its backup supply (input 1505), therefore the vital AC instrument bus is OPERABLE.
C correct, the vital instrument bus is powered from a class 1E CVT (backup supply) and OPERABLE, the inverter is inoperable.
D incorrect, the vital AC bus is currently energized.

Technical Reference(s):
Tech Spec 3.8.7
Tech Spec Bases B3.8.7
AR PK19-19

sro tier 1 group 1_81.doc

Proposed references to be provided to applicants during examination:
Tech Spec 3.8.7
Tech Spec Bases B3.8.7
AR PK19-19

Learning Objective: 9697 - Identify Technical Specification LCOs

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 ______
55.43 43.2

Comments:

K/A: APE 057 AA2.14 - Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus: That substitute power sources have come on line on a loss of initial ac

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters-Operating

LCO 3.8.7 Four Class 1E Vital 120 V UPS inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 -----------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital 120 V AC bus de-energized. ---------------------------------- Restore inverter to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage and alignment to required AC vital buses.	7 days

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The Class 1E UPS inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the FSAR, Chapter 7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and

b. A worst case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The Class 1E UPS inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

(continued)

BASES

LCO (continued)	Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters ensure an uninterruptible supply of AC electrical power to the 120 VAC vital buses even if the 4.16 kV safety buses are de-energized. Operable inverters require the associated 120 VAC vital bus to be powered by the inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC station battery. Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply.
APPLICABILITY	The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that: a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA. Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."
ACTIONS	<u>A.1</u> With a required inverter inoperable, its associated 120 VAC vital bus becomes inoperable until it is re-energized from its Class 1E constant voltage source transformer. For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the 120 VAC bus is re-energized within 2 hours. Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC vital bus is powered from its constant voltage source, it is relying upon interruptible

(continued)

BASES

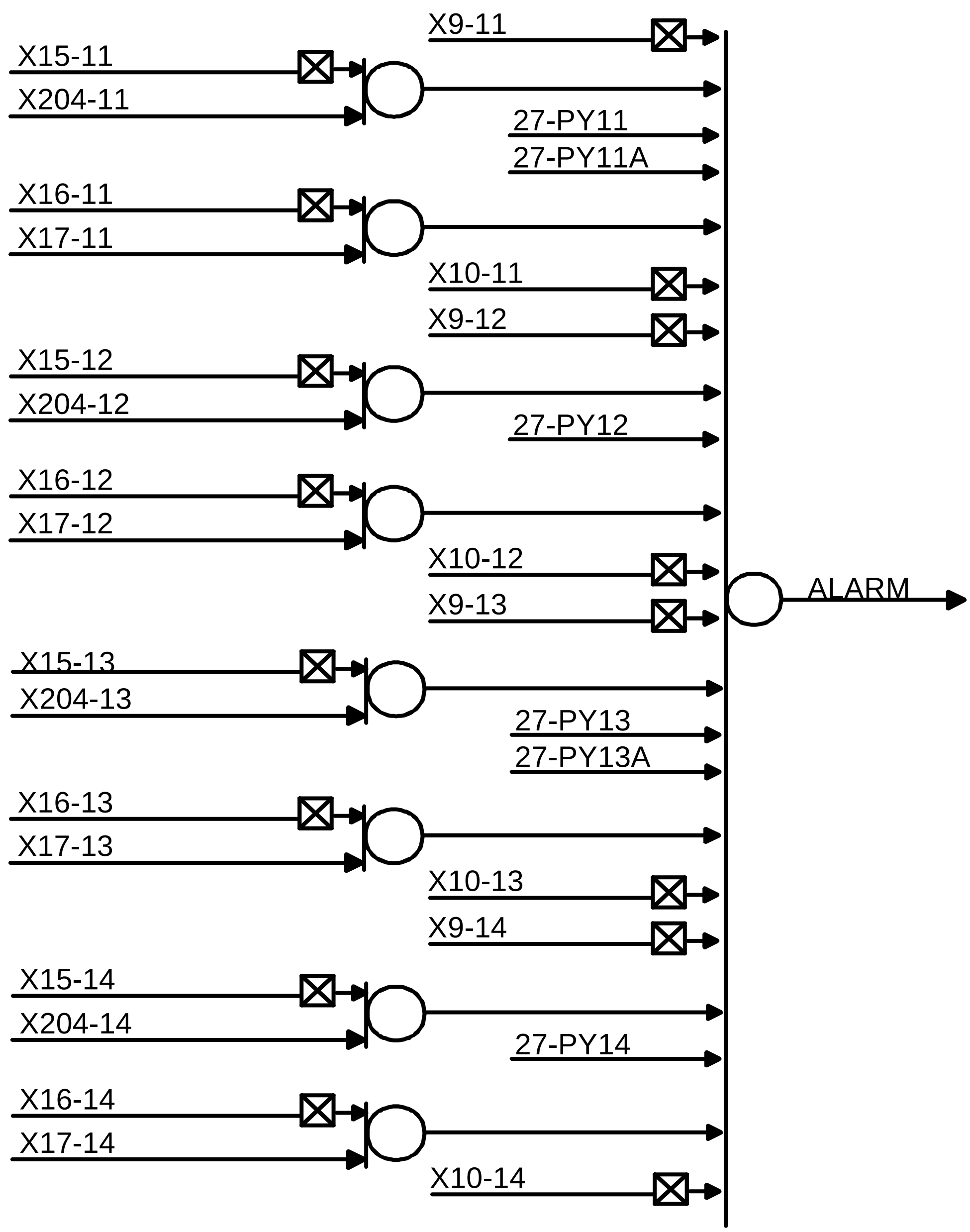
ACTIONS	A.1 (continued) AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC vital buses is the preferred source for powering instrumentation trip setpoint devices. B.1 and B.2 If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.8.7.1 This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.
REFERENCES	1. FSAR, Chapter 7. 2. FSAR, Chapter 6. 3. FSAR, Chapter 15.

TITLE: VITAL UPS FAILURE

02/04/00
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
27PY11	1482	Instr AC Dist Panel 1-1 Undervoltage	
27PG11A	1483	Instr AC Dist Panel 1-1A Undervoltage	
X10-11	1484	Instr AC UPS 1-1 Inverter Failure	
X15-11	1485	Instr AC UPS 1-1 Loss of AC Output	
X16-11	1486	Instr AC UPS 1-1 on Bypass	
X9-11	1487	Instr AC UPS 1-1 Loss of Battery Source	
27PY12	1489	Instr AC Dist Panel 1-2 Undervoltage	
X10-12	1490	Instr AC UPS 1-2 Inverter Failure	
X15-12	1491	Instr AC UPS 1-2 Loss of AC Output	
X16-12	1492	Instr AC UPS 1-2 on Bypass	
X9-12	1493	Instr AC UPS 1-2 Loss of Battery Source	
27PY13	1495	Instr AC Dist Panel 1-3 Undervoltage	
27PY13A	1496	Instr AC Dist Panel 1-3A Undervoltage	
X10-13	1497	Instr AC UPS 1-3 Inverter Failure	
X15-13	1498	Instr AC UPS 1-3 Loss of AC Output	
X16-13	1499	Instr AC UPS 1-3 on Bypass	
X9-13	1500	Instr AC UPS 1-3 Loss of Battery Source	
27PY14	1502	Instr AC Dist Panel 1-4 Undervoltage	
X10-14	1503	Instr AC UPS 1-4 Inverter Failure	
X15-14	1504	Instr AC UPS 1-4 Loss of AC Output	
X16-14	1505	Instr AC UPS 1-4 on Bypass	
X(-14	1506	Instr AC UPS 1-4 Loss of Battery Source	

3. PROBABLE CAUSE

3.1 PY panel de-energized.

3.2 PY panel energized from the Backup Regulating Transformer.

3.3 Loss of output voltage from a UPS.

3.4 Loss of DC input to an inverter.

3.5 Inverter failure.

4. AUTOMATIC ACTIONS

4.1 Power supply to PY panel transferred to Backup Regulating Transformer.

5. OPERATOR ACTIONS

5.1 INSTRUMENT AC DIST PANEL UNDERVOLTAGE

5.1.1 If reactor trips, go to EOP E-0.

5.1.2 If loss of instrument AC System, go to OP AP-4.

5.1.3 Refer to Tech Spec. 3.8.2.1 and 3.8.2.2 (ITS 3.8.9 and 3.8.10).

5.2 INSTRUMENT AC UPS LOSS OF AC OUTPUT

5.2.1 If reactor trips, go to EOP E-0.

5.2.2 If loss of instrument AC System, go to OP AP-4.

5.2.3 Refer to Tech Spec. 3.8.2.1 and 3.8.2.2 (ITS 3.8.7 and 3.8.8).

5.3 INSTRUMENT AC UPS ON BYPASS

NOTE: The UPS output will transfer to Bypass supply if a fault is sensed on the UPS output. The UPS output should then transfer back to the UPS output. The red light P202 BYPASS SOURCE SUPPLYING LOAD light will clear when the UPS output transfers back the inverter, but the control room alarm will stay latched in until alarm reset button S2 is pressed.

5.3.1 Determine why UPS output is on Bypass supply.

5.3.2 Re-energize the UPS output from the Inverter supply when problem is corrected.

5.3.3 Refer to Tech Spec 3.8.2.1 and 3.8.2.2 (ITS 3.8.7 and 3.8.8).

5.4 INSTRUMENT AC UPS LOSS OF BATTERY SOURCE

5.4.1 Determine why the UPS lost its DC input.

5.4.2 Restore the DC input.

NOTE: To prevent damage to the inverter, the inverter should not be operated without its DC input.

5.4.3 If the DC Input cannot be restored, then transfer the UPS output to the Bypass supply per OP J-10:IV.

5.4.4 Refer to Tech Spec 3.8.2.1 and 3.8.2.2 (ITS 3.8.7 and 3.8.8).

5.5 INSTRUMENT AC UPS INVERTER FAILURE

5.5.1 Verify UPS output transferred to Bypass Supply. If the UPS output did not transfer, then go to OP AP-4.

5.5.2 Notify Maintenance Services to troubleshoot and correct the problem with the inverter.

5.5.3 Refer to Tech Spec 3.8.2.1 and 3.8.2.2 (ITS 3.8.7 and 3.8.8).

5.5.4 Re-energize the UPS output from the inverter when the inverter is repaired.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	______	1
	Group #	______	2
	K/A #	APE 061 AA2.05	
	Importance		4.2

Proposed Question:

Unit 2 is performing a core offload.

AR PK11-21 alarms. The SFM is determining if the alarm is valid to determine if containment should be evacuated per AP-21, Irradiated Fuel Damage.

Which of the following describes why the SFM should order containment evacuation? Assume all readings are valid.

A. Containment Area monitor RE-2 reads 30 mR/hr.

B. Containment Air Radiogas monitor RE-12 reads 5.2 $x10^3$ cpm.

C. Containment Ventilation Exhaust monitor RE-44A reads 7.6 $x10^{-5}$ µCi/cc.

D. Containment Ventilation Exhaust monitor RE-44B reads 1.7 $x10^{-4}$ µCi/cc.

Proposed Answer:

B. Containment Air Radiogas monitor RE-12 reads 5.2 x103 cpm.

Explanation:
A incorrect, this is above the modes 2-5 setpoint but less than the mode 6 setpoint.
B correct, unit 2 high alarm setpoint is 5.1 $x10^3$ cpm.
C incorrect, above alert but below high alarm setpoint.
D incorrect, above alert but below high alarm setpoint.

Technical Reference(s):
OP AP-21, Irradiated Fuel Damage
Vol. 9B tables:

- T-IIC-2 (page II-5-53)
- T-IIC-2 (page II-5-58)
- T-IIC-2 (page II-5-85a)
- T-IIC-2 (page II-5-86b)

Proposed references to be provided to applicants during examination:

- T-IIC-2 (page II-5-53)
- T-IIC-2 (page II-5-58)
- T-IIC-2 (page II-5-85a)
- T-IIC-2 (page II-5-86b)

Learning Objective: 6543 - Explain the actions for a containment evacuation alarm

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 ______
55.43 43.5

Comments: unit difference

K/A: APE 061 AA2.05 - Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Need for area evacuation; check against existing limits

CHANNEL: 2-RM-2

DESCRIPTION: Containment Area Radiation Monitor

	HIGH ALARM SETPOINT MODES 1&6	HIGH ALARM SETPOINT MODES 2-5	NET CHECK SOURCE READING
MAX	50 mR/hr	24 mR/hr	12 mR/hr
NOM	21 mR/hr	10 mR/hr	6.0 mR/hr
MIN	9 mR/hr	4.2 mR/hr	3.0 mR/hr

NOMINAL RO HV 540 Vdc
NOMINAL ESVM HV 535 Vdc

CALIBRATION DATE 02/10/04-02/13/04
CALIBRATION WO R0237286

DETECTOR TYPE/SER # LND 71416 / 71608

CABLE INTEGRITY CHECK

Loop Resistances	pins 5 and 6	- 5.616 Ω
	pins 8 and 9	- 3.901 Ω
	pins 4 and 7	- 3.420 Ω
Insulation Resistance	pins 5 and 6	- 7.3×10^{12} Ω
Capacitance	pins 5 and 6	- 2.65 nF

BENCHMARK WO C0127173 DATE 05/26/94

COMPUTER ALARM LIMITS

	ADDRESS	HALM MODES 1&6	HALM MODES 2-5	
PPC	R0002A	21	10	mR/hr
	R0002R			mR/hr
SPDS	RM2A	21	10	mR/hr

REFERENCES: STP I-107A and STP I-107B3

EFFECTIVE DATE 3·5·04

CHANNEL: 2-RM-12

DESCRIPTION: Containment Air Radiogas Radiation Monitor

HIGH ALARM SETPOINT		NET CHECK SOURCE READING	
MAX	5.1×10^3 cpm	MAX	1.7×10^3 cpm
NOM	2.6×10^3 cpm	NOM	8.6×10^2 cpm
MIN	1.3×10^3 cpm	MIN	4.3×10^2 cpm

NOMINAL RO HV 1030 Vdc
NOMINAL ESVM HV 1025 Vdc

CALIBRATION DATE 06/10/03-06/11/03
CALIBRATION WO R0228463

CABLE INTEGRITY CHECK

Loop Resistances	pins 5 and 6	- ______ Ω
	pins 8 and 9	- ______ Ω
Insulation Resistance	pins 5 and 6	- ______ Ω
Capacitance	pins 5 and 6	- ______ nF

BENCHMARK WO ________ DATE ________

PPC ALARM LIMIT

ADDRESS	HALM	
R0012A	2.6×10^3	cpm
R0012R	N/A	cpm

REFERENCES: STP I-100A and STP I-100B3

APPROVAL:
ICE MANAGER/SR ENG [signature] DATE 8-28-03

CHANNEL: 2-RM-44A

DESCRIPTION: Containment Ventilation Exhaust Radiation Monitor

T1 CONSTANT	REQUIRED KEYSTROKES (TEST POINTS)	VALUE	UNITS	DESCRIPTION/REMARKS
HVDMM	(LRP 1000:1 & GND)	1.011	Vdc	High Voltage at LRP 1000:1 test points
HVESVM	N/A	1025	Vdc	High Voltage at detector
NETCS	N/A	217	cpm	In-situ NET Check Source reading
HASP	CH 0 FUNC 0 DISP	2.07E-04	μCi/cc	High Alarm Set Point (3.0 x 10^4 cpm)
AASP	CH 0 FUNC 1 DISP	6.21E-05	μCi/cc	Alert Alarm Set Point (9.1 x 10^3 cpm)
TGTCNT	CH 0 FUNC 3 DISP	9.99E19	Counts	Target Counts
BRSUB	CH 0 FUNC 4 DISP	1.50E02	cpm	Background Subtraction value
DTIME	CH 0 FUNC 8 DISP	1.01E-08	Mins/Count	Detector Dead Time Constant
CC	CH 0 FUNC 9 DISP	6.86E-09	μCi/cc/cpm	Conversion Constant (detector efficiency)

DETECTOR TYPE/SER# 943-25T / 163

CALIBRATION DATE 08/05/03-08/06/03
CALIBRATION WO R0224930

CABLE INTEGRITY CHECK

HV Cable RTR = 1.14 Ω
HV Cable IR = 1.82E11 Ω (@ 1000 Vdc - GC TP J-46480-02A dated 03/22/93)

SIG Cable RTR = 0.41 Ω
SIG Cable IR = 9.2E11 Ω (@ 1000 Vdc - GC TP J-46480-02A dated 03/22/93)

PPC ALARM LIMIT

ADDRESS	HALM
R0044AR	2.1E-04 μCi/cc

REFERENCES: STP I-39-R44A.A and STP I-39-R44A.B

EFFECTIVE DATE 11-24-04

CHANNEL: 2-RM-44B

DESCRIPTION: Containment Ventilation Exhaust Radiation Monitor

T1 CONSTANT	REQUIRED KEYSTROKES (TEST POINTS)	VALUE	UNITS	DESCRIPTION/REMARKS
HVDMM	(LRP 1000:1 & GND)	1.132	Vdc	High Voltage at LRP 1000:1 test points
HVESVM	N/A	1150	Vdc	High Voltage at detector
NETCS	N/A	224	cpm	In-situ NET Check Source reading
HASP	CH 0 FUNC 0 DISP	2.07E-04	μCi/cc	High Alarm Set Point (3.0 x 10^4 cpm)
AASP	CH 0 FUNC 1 DISP	6.21E-05	μCi/cc	Alert Alarm Set Point (9.1 x 10^3 cpm)
TGTCNT	CH 0 FUNC 3 DISP	9.99E19	Counts	Target Counts
BRSUB	CH 0 FUNC 4 DISP	1.50E02	cpm	Background Subtraction value
DTIME	CH 0 FUNC 8 DISP	1.10E-08	Mins/Count	Detector Dead Time Constant
CC	CH 0 FUNC 9 DISP	6.86E-09	μCi/cc/cpm	Conversion Constant (detector efficiency)

DETECTOR TYPE/SER# 943-25T / 150

CALIBRATION DATE 08/07/03-08/09/03
CALIBRATION WO R0224935

CABLE INTEGRITY CHECK

HV Cable RTR = 2.16 Ω
HV Cable IR = 1.33E11 Ω (@ 1000 Vdc - GC TP J-46480-02A dated 03/22/93)

SIG Cable RTR = 0.72 Ω
SIG Cable IR = 5.10E9 Ω (@ 1000 Vdc - GC TP J-46480-02A dated 03/22/93)

PPC ALARM LIMIT

ADDRESS	HALM
R0044BR	2.1E-04 μCi/cc

REFERENCES: STP I-39-R44B.A and STP I-39-R44B.B

EFFECTIVE DATE 11.24.04

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	2
	K/A #	W/E02 EA2.1	
	Importance		4.2

Proposed Question:

A steam break occurs inside containment on Steam Generator 2-2. The crew is performing the steps of E-0, Reactor Trip or Safety Injection.

Current plant conditions:

- Containment pressure – 14 psig (peak at 25 psig)
- Steam Generator 2-2 early isolation performed
- Total AFW throttled to 500 gpm
- RCS pressure – 1850 and increasing
- Steam Generator pressures:
  - o 2-1 – 900 psig, stable
  - o 2-2 – 0 psig, stable
  - o 2-3 and 2-4 – 1000 psig, stable
- Pressurizer Level – 35%, increasing
- RCS Subcooling 50ºF

The procedural flowpath should be E-0 to …

A. E-1.1

B. E-2 to E-1.1

C. FR-Z.1 to E-1.1

D. FR-Z.1 to E-2 to E-1.1

Proposed Answer:

B. E-2 to E-1.1

Explanation:

A incorrect, at step 11 of E-0, (No Steam Generator completely depressurized), the crew will be directed to E-2.
B correct, crew will go to E-2, confirm/complete the S/G isolation then go to E-1.1
C incorrect, the condition for transitioning to Z.1 no longer exists.
D incorrect, the condition for transitioning to Z.1 no longer exists.

Technical Reference(s):
E-0, F-0

Proposed references to be provided to applicants during examination: none

Learning Objective: 3552 - Identify entry conditions for the EOPs

Question Source: Bank #
Modified Bank # DCPP P-51520
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 _____
55.43 43.5

Comments:

K/A: W/E02 EA2.1 - Ability to determine and interpret the following as they apply to SI Termination: Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

1 P-51520 Points: 1.00 Multiple Choice

A main steamline break, upstream of the MSIV and outside containment, occurs on S/G 1-2. The expected procedural flowpath is:

A. E-0, E-2, E-1, E-1.1

B. E-0, E-1, E-2, E-1.2

C. E-0, E-2, E-1, E-1.2

D. E-0, E-1, E-2, E-1.1

Answer: A

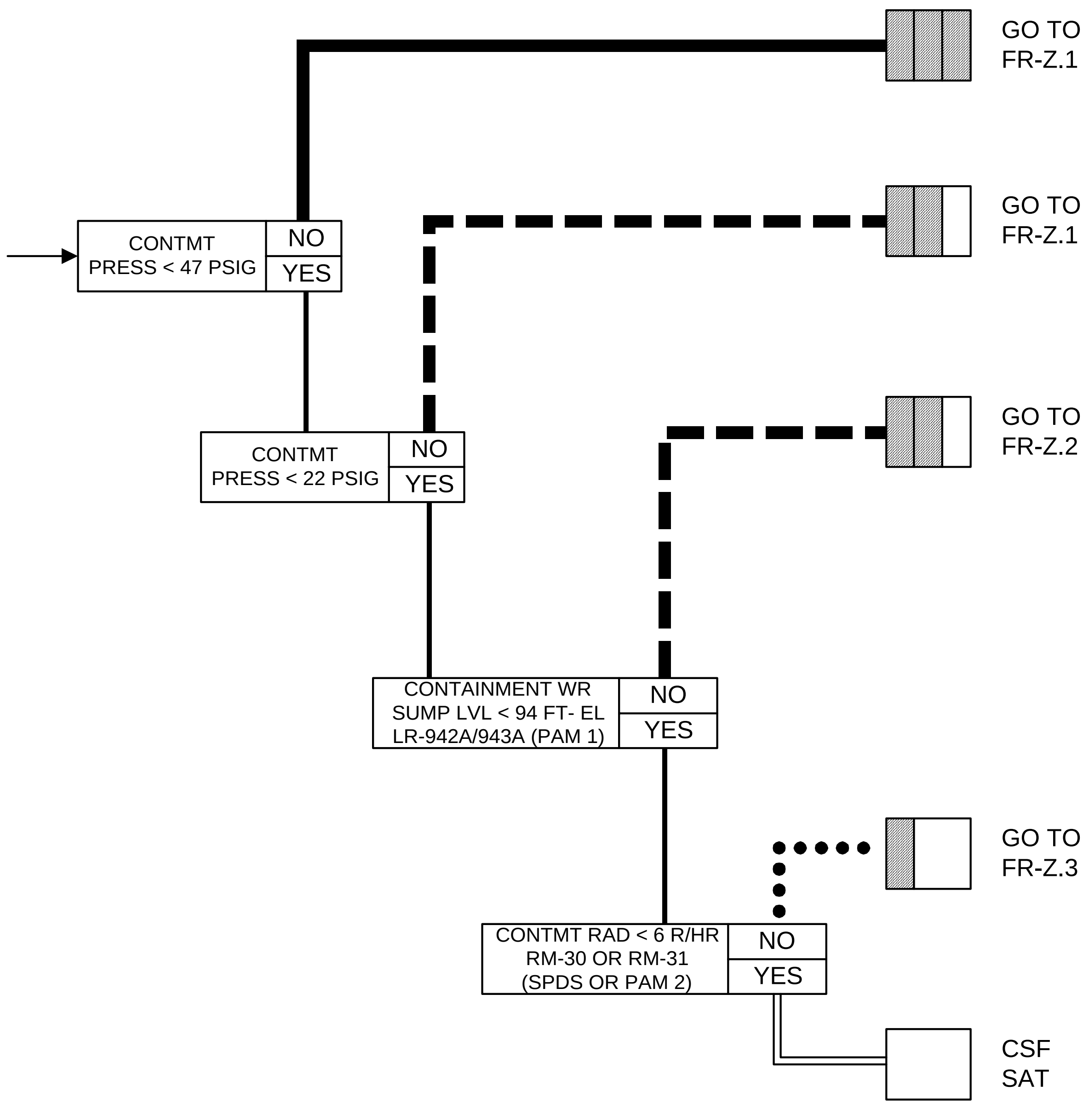
ASSOCIATED INFORMATION:

Associated objective(s):

3552	Identify entry conditions for the EOPs

Reference Id: P-51520
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 2.00
Time to complete: 2
Topic: LPE1A EOP flowpath for MSLB
Cross Reference: LPE-1A OBJ 2, 3
Comment:

F-0.5 CONTAINMENT



#

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **CHECK S/Gs - NOT FAULTED**	GO TO EOP E-2, FAULTED STEAM GENERATOR ISOLATION
o NO S/G Pressure decreasing in an uncontrolled manner	
o NO S/G Completely depressurized	- -
12. **Check S/Gs - NOT RUPTURED**	
a. Secondary Radiation Monitors - NORMAL	a. IF A valid alarm exists THEN GO TO EOP E-3, STEAM GENERATOR TUBE RUPTURE
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	b. IF A valid upward trend or spike exists, THEN GO TO EOP E-3, STEAM GENERATOR TUBE RUPTURE
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)	c. GO TO EOP E-3, STEAM GENERATOR TUBE RUPTURE

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. SJAE Radiation - NORMAL o PPC trend indicates NO upward trend or spike on RM-15 or 15R - SJAE Monitors o NO valid alarm on PK11-06, SJAE 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.
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.

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
8. **CHECK If ECCS Flow Should Be Reduced:**	
a. RCS Subcooling GREATER THAN 20°F (SCMM or Appendix C)	a. GO TO Step 9.
b. Secondary heat sink satisfied:	b. GO TO Step 9.
o Total feed flow to intact S/Gs – GREATER THAN 435 GPM	
OR	
o NR Level in at least one intact S/G – GREATER THAN 6% [16%]	
c. RCS Pressure – STABLE OR INCREASING	c. GO TO Step 9
d. PZR Level – GREATER THAN 12% [36%]	d. GO TO Step 9
e. GO TO EOP E-1.1, SI TERMINATION	
9. **GO TO EOP E-1, LOSS OF REACTOR OR SECONDARY COOLANT**	

- END -

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	2
	K/A #	W/E09 EA2.1	
	Importance		3.8

Proposed Question:

The following plant conditions exist:

- A loss of offsite power has occurred.
- The crew is performing the actions of E-0.2, Natural Circulation Cooldown.
- Low steamline pressure and low PZR pressure SI signals are blocked.
- RCS pressure is 1850 psig and trending down slowly.
- Pressurizer Level is 55% and increasing slowly
- CST level is 35%
- RCS temperature is 535ºF
- The crew performing a cooldown at 25ºF per hour.
- RVLIS is unavailable.
- All CRDM fans are running

Based on the current plant conditions, which of the following procedures should the crew be in when the plant reaches cold shutdown?

A. E-0.2, Natural Circulation Cooldown

B. E-0.3, Natural Circulation Cooldown With Steam Void in the Vessel (With RVLIS)

C. E-0.4, Natural Circulation Cooldown With Steam Void in the Vessel (Without RVLIS)

D. OP L-7, Plant Stabilization Following a Reactor Trip or OP L-5, Plant Cooldown From Minimum Load to Cold Shutdown.

Proposed Answer:

C. E-0.4, Natural Circulation Cooldown With Steam Void in the Vessel (Without RVLIS

Explanation:
A incorrect, based on graph IB-3, insufficient CST level means a faster cooldown is needed than is accomplished in E-0.2
B incorrect, RVLIS is not available.
C correct, based on CST level and no RVLIS, crew should go to E-0.4
D incorrect, these are the procedures used if a RCP can be started.

Technical Reference(s):
E-0.2, Natural Circulation Cooldown
Vol 9, Section IF, Figure IB-2, (page IF-2)

Proposed references to be provided to applicants during examination:
Pages IF-1, IF-2, IF-3

Learning Objective: 5433 - Identify exit conditions for the EOPs

Question Source:
New X

Question History: Last NRC Exam N/A
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

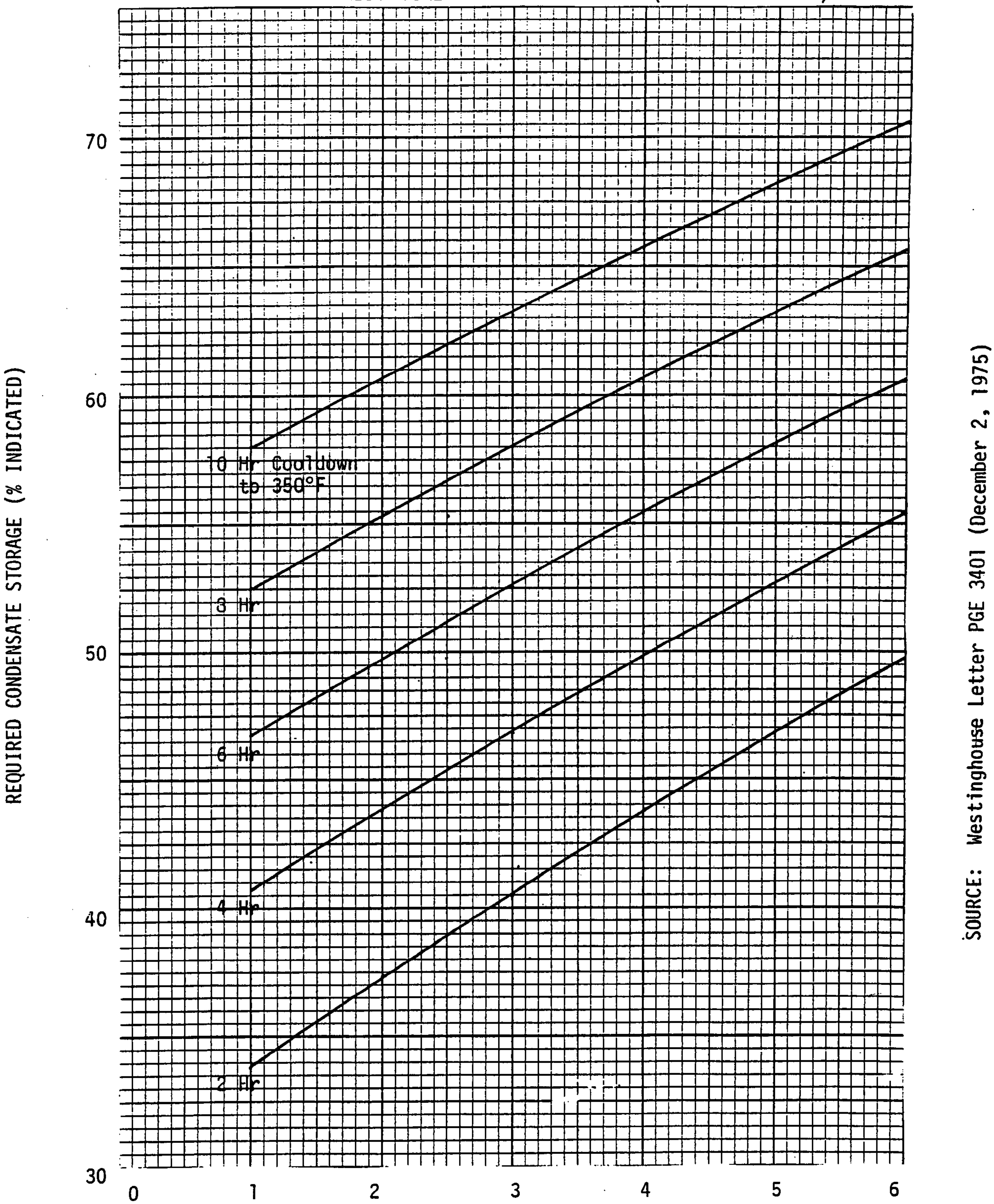
10 CFR Part 55 Content: 55.41 ______
55.43 43.5

Comments: is E-0.2 necessary to answer the question?

K/A: W/E09 EA2.1 - Ability to determine and interpret the following as they apply to Natural Circulation Operations: Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

DIABLO CANYON POWER PLANT OPERATION DATA

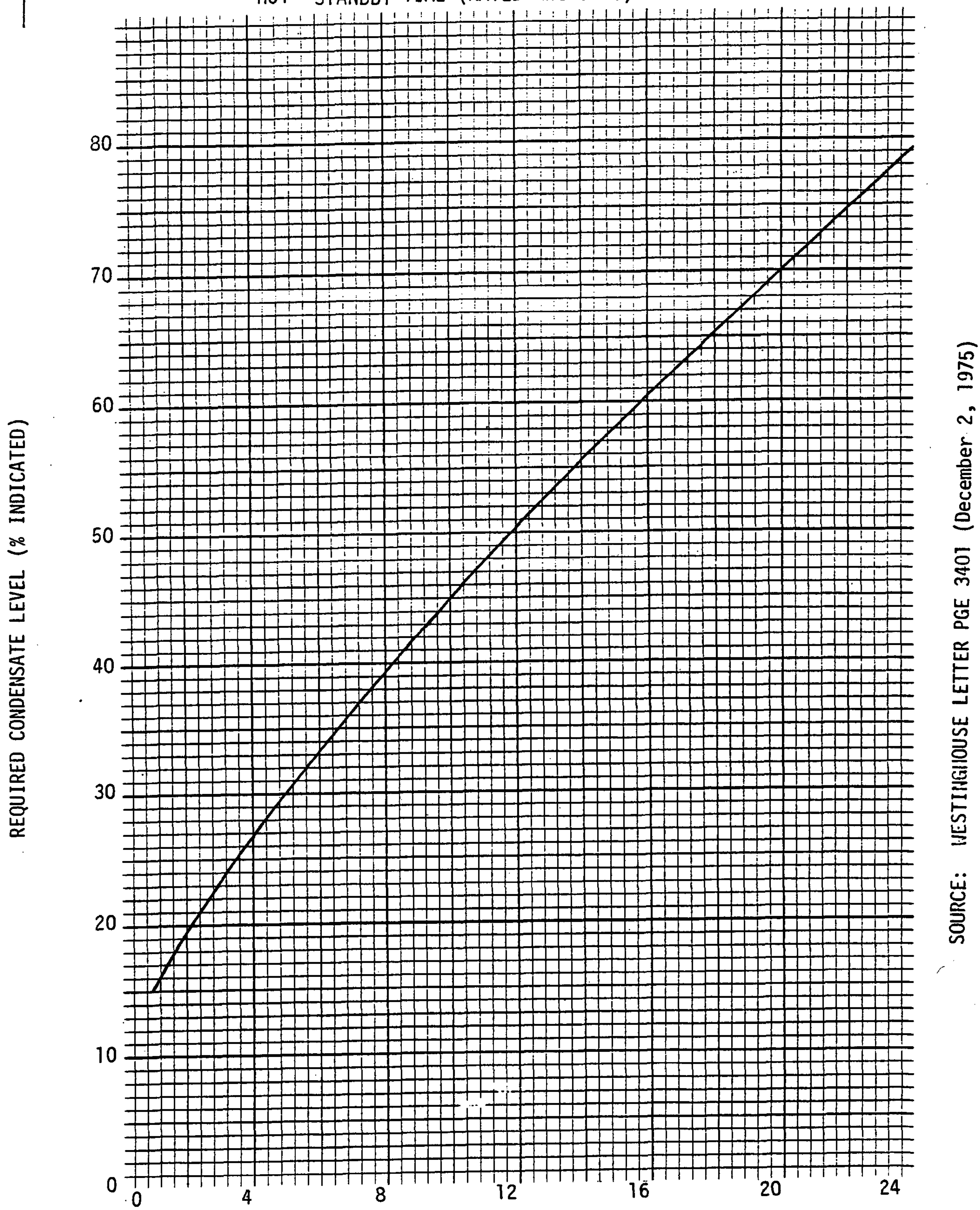
REQUIRED CONDENSATE STORAGE CAPACITY AS A FUNCTION OF HOT STANDBY TIME AND COOLDOWN TIME (RATED MWt=3423)



TIME AT HOT STANDBY (HOURS) FIGURE IB-1

REVISION 1 DATE 4/6/81

DIABLO CANYON POWER PLANT OPERATION DATA
REQUIRED CONDENSATE LEVEL TO MAINTAIN A GIVEN
HOT STANDBY TIME (RATED MWt=3423)



HOT STANDBY TIME (HOURS) FIGURE IB-2

Page IF-2

REVISION 1 DATE 4/6/81

DIABLO CANYON POWER PLANT OPERATION DATA
REQUIRED LEVEL OF CONDENSATE TO COOLDOWN REACTOR SYSTEM TO 350°F AS A FUNCTION OF DURATION OF COOLDOWN FOR VARIOUS HOT STANDBY DURATIONS.

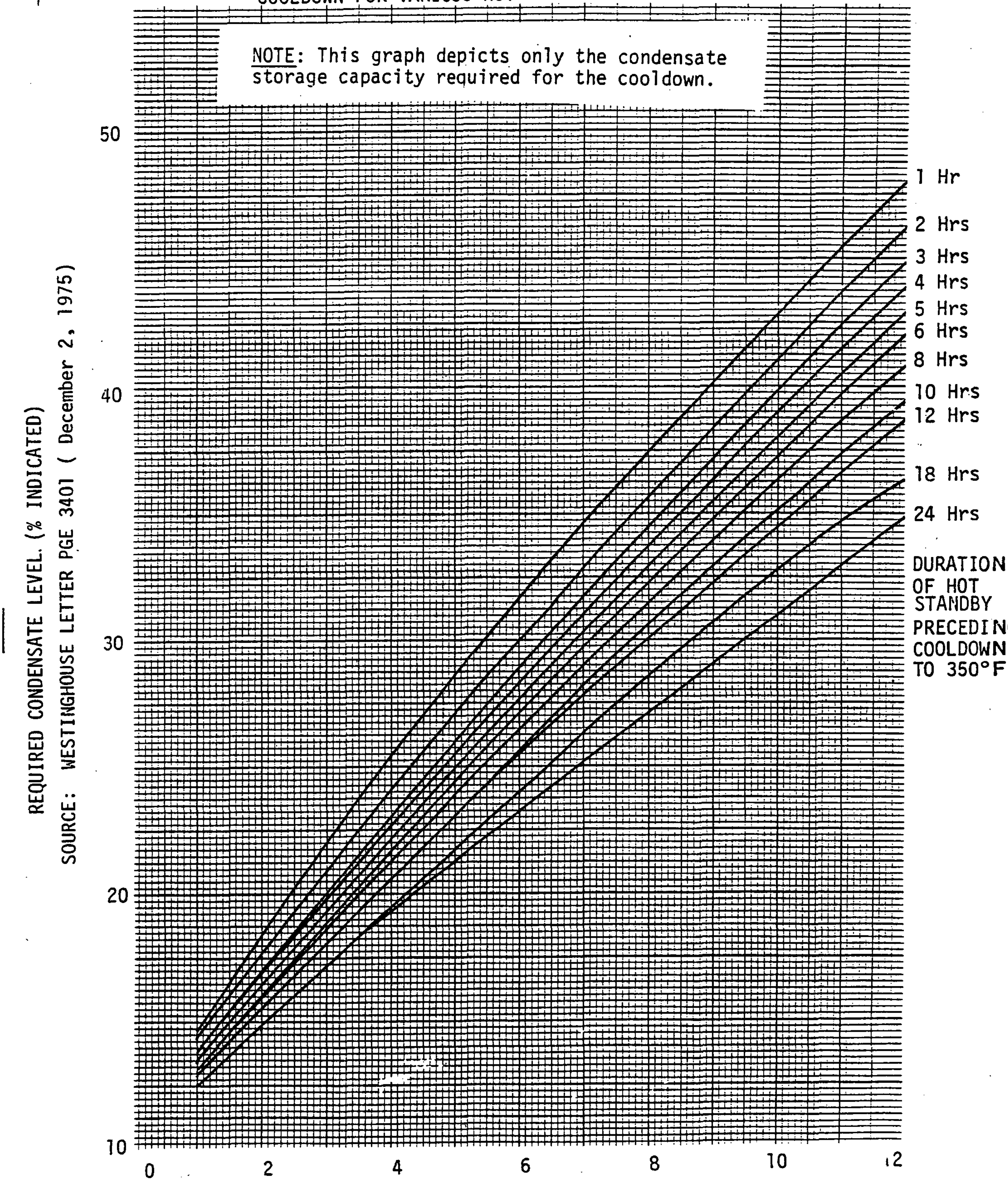


FIGURE IB-3

REVISION 1 DATE 4/6/81

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
15. **INITIATE RCS Depressurization With The RCS Cooldown:**	
NOTE: Refer to Volume 9, Section "IF", for data to evaluate the required cooldown rate considering CST volume.	
a. Maintain 25°F/hr Cooldown Rate	
b. Maintain the cooldown rate and depressurization criteria listed on the Foldout page	b. IF It is determined that the Cooldown and Depressurization must be performed at a rate that may cause formation of a Steam Void in the Vessel, THEN GO TO EOP E-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS) OR GO TO EOP E-0.4, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS). -

THIS STEP CONTINUED ON NEXT PAGE

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	1
	Group #	_____	2
	K/A #	W/E13 G2.4.30	
	Importance		3.6

Proposed Question:

The following conditions exist on Unit 1:

- A spurious closure of all MSIVs occurred while operating at 100% power
- The reactor tripped and immediate actions of E-0, “Reactor Trip or Safety Injection” performed
- During the post trip review, it is discovered that an overpressure condition on 13 SG with pressure at 1240 psig existed.
- There is no indication that any of the safeties on the 13 steam generator operated
- All other steam generators and plant safety systems functioned as designed
- During subsequent repairs the unit has been holding in MODE 3 for the past 20 hours

The condition of the 13 steam generator is reportable to the NRC because:

A. The plant exceeded a safety limit.

B. A loss of two fission barriers was imminent.

C. Challenges occurred to the safety valves.

D. The plant is in a condition prohibited by Technical Specifications.

Proposed Answer:

D. The plant is in a condition prohibited by Technical Specifications.

Explanation:
A incorrect, the safety limits are RCS pressure and Reactor Core (the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure).
B incorrect, no evidence of a loss of a fission barrier.
C incorrect, the challenge of the safety valves is not the reportable event, the loss of the safety valves is the reportable event.
D correct, steam generator safeties are necessary in MODES 1 thru 3. All safeties are inoperable. The unit should have been in MODE 4 in 12 hours.

Technical Reference(s):
Technical Specification 2.1 and 3.7.1
XI1.ID2, Regulatory Reporting Requirements and Reporting Process, attachment 8.5 page 4 of 8

Proposed references to be provided to applicants during examination:
Technical Specification 2.1 and 3.7.1

Learning Objective: 9697 - Identify Technical Specification LCOs

Question Source: Bank # INPO 21506

Question History: Last NRC Exam Braidwood 7/2002
Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 43.2

Comments: print TS 3.7.1, 2.1, Eplan G-1 and Reporting Requirements, att. 8.4

K/A: W/E13 G2.4.30 – Steam Generator Overpressure - Knowledge of which events related to system operations/status should be reported to outside agencies.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

UNITS 1 & 2

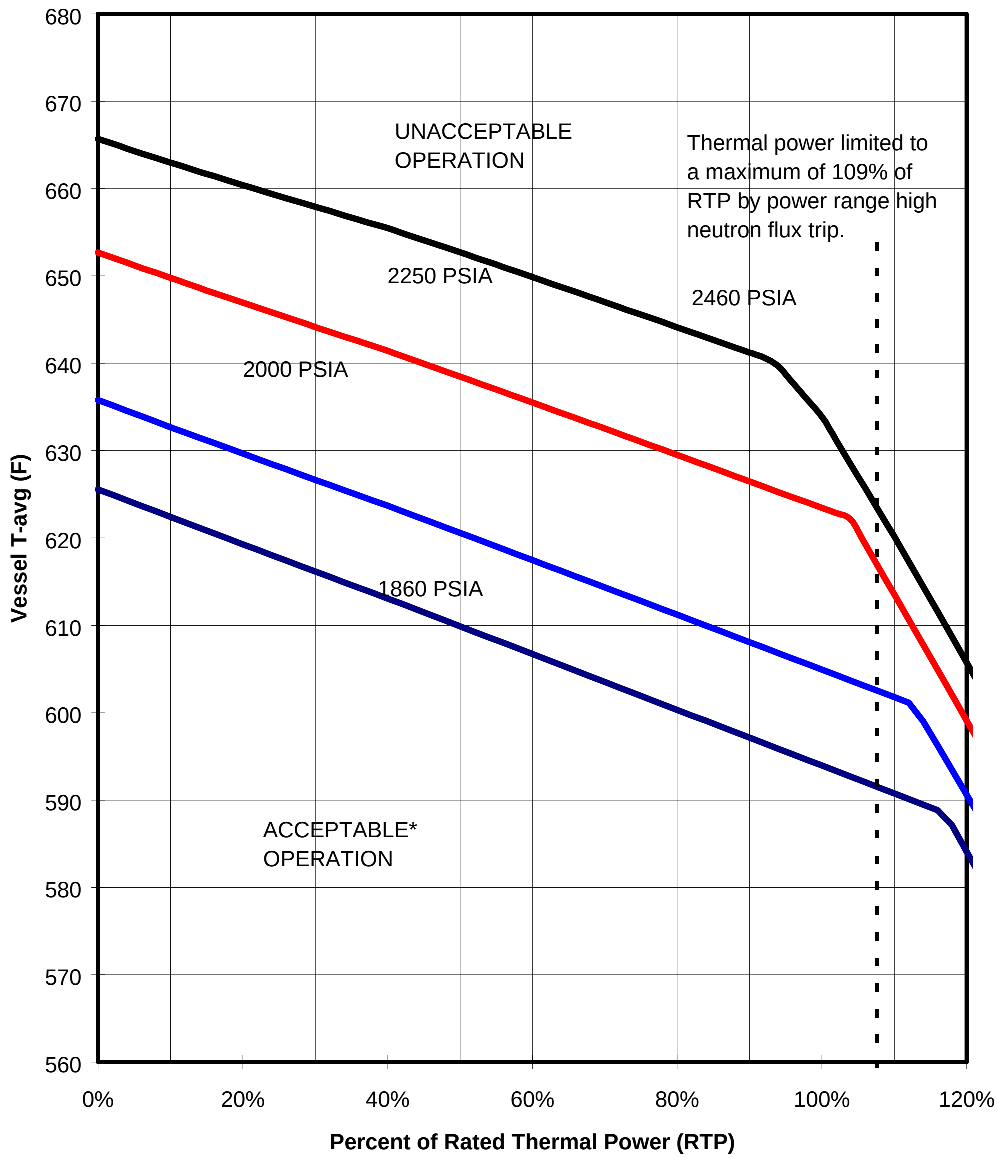


Figure 2.1.1-1
REACTOR CORE SAFETY LIMIT

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each MSSV.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSSVs inoperable.	A.1 Reduce the Power Range High Neutron Flux trip setpoint per Table 3.7.1-1.	4 hours
B. Required Action and associated Completion Time not met. OR One or more steam generators with less than two MSSVs OPERABLE.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	----------------------------NOTE--------------------------- Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ± 1%.	In accordance with the Inservice Testing Program.

Table 3.7.1-1 (page 1 of 1)
Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable MSSVs

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT %RTP
4	87*
3	47*
2	29*

* Unless the reactor trip system breakers are in the open position.

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFT SETTING (psig)
STEAM GENERATOR				
#1	#2	#3	#4	
RV-3	RV-7	RV-11	RV-58	1065 (-2%, +3%)
RV-4	RV-8	RV-12	RV-59	1078 (± 3%)
RV-5	RV-9	RV-13	RV-60	1090 (± 3%)
RV-6	RV-10	RV-14	RV-61	1103 (± 3%)
RV-222	RV-223	RV-224	RV-225	1115 (± 3%)

SRO QUESTION 86

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	______	2
	Group #	______	1
	K/A #	003 A2.05	
	Importance		2.8

Proposed Question:

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.

SRO QUESTION 87

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	______	2
Group #	______	1
K/A #	005 G2.4.45	
Importance		3.6

Proposed Question:

The crew is performing the actions of E-1, Loss of Reactor or Secondary Coolant.

PK01-22, RHR SUCT VLV CHMBR LEVEL HI alarms.

Which of the following describes whether or not action should be taken by the SFM to investigate the alarm?

A. Action should be taken. The alarm indicates seat leakage is occurring which could cause the valve(s) to bind when opened.

B. Action should be taken. The alarm indicates a potential leak, which if left uncorrected could lead to motor damage.

C. No action necessary. The valves are designed to operate in adverse conditions.

D. No action necessary. The alarm is expected as water from the containment sump fills the piping up to the valve.

Proposed Answer:

B. Action should be taken. The alarm indicates a potential leak, which if left uncorrected could lead to motor damage.

Explanation:

A incorrect, this is not an indication of seat leakage.
B correct, If chamber level gets too high, valves 8982A or B may get water into the valve motor and render the valve inoperable.
C incorrect, the valves may not operate in this condition.
D incorrect, if piping is intact, this alarm will not actuate.

Technical Reference(s):

sro tier 2 group 1_87.doc

PK01-22, RHR SUCT VLV CHMBR LEVEL HI

Proposed references to be provided to applicants during examination: none

Learning Objective: 7036 - State the RHR parameters that produce alarms.

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 _____
55.43 43.5

Comments:

K/A: 005 G2.4.45 - Residual Heat Removal - Ability to prioritize and interpret the significance of each annunciator or alarm.

#

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
ANNUNCIATOR RESPONSE

NUMBER AR PK01-22
REVISION 8A
PAGE 1 OF 1
UNIT

1

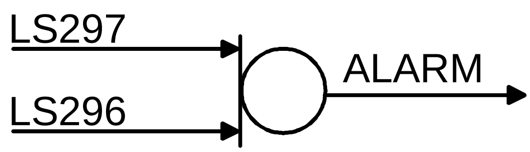
TITLE: **RHR SUCT VLV CHMBR LEVEL HI**

12/17/99
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
LS 297	444	RHR Suction Vlv Chamber 1-1 Lvl High	GT 11" From Bottom of Chamber
LS 296	445	RHR Suction Vlv Chamber 1-2 Lvl High	GT 11" From Bottom of Chamber

3. PROBABLE CAUSE

   3.1 Packing leak on RHR suction valves from sump.

   3.2 Leak in recirculation sump penetration.

4. AUTOMATIC ACTIONS

   None

5. OPERATOR ACTIONS

****

CAUTION: If chamber lvl. gets too high, valves 8982A or B may get water into the valve motor and render the valve inoperable.

****

   5.1 Check the recirc. chamber through the view port for visible leakage or water level.

   5.2 Drain the chamber to the miscellaneous equipment drain tank.

   5.3 File an Action Request so the condition can be repaired as soon as practical.

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	______	2
Group #	______	1
K/A #	022 A2.04	
Importance		3.2

Proposed Question:

High ocean temperature and fouling of CCW heat exchanger 11 has caused the following conditions to occur on Unit 1:

- PK01-16, CONTMT ENVIRONMENT – PPC is in alarm
- Containment Temperature ;
  - 100 ft elevation between crane wall and containment wall — 122 °F
  - 100 ft elevation between S/Gs — 115 °F
  - 140 ft elevation — 115 °F
  - 184 ft elevation — 120 °F
- CFCUs 11, 12, 13 and 15 are running in FAST
- All CRDM fans are running
- Amps on CFCUs 11, 12, and 13 are 330
- Amps on CFCU 15 is 340
- Containment pressure is above the alarm setpoint and is currently 1.0 psig and increasing slowly

Based on the current plant conditions, which of the following actions should the SFM direct the operator to perform?

A. Direct a start of CFCU 14, and consider venting containment.

B. Direct a start of CFCU 14 and restore pressure within limits within the next 4 hours or be in MODE 3 in the next 6 hours.

C. Direct a start of start CFCU 14, reduce containment temperature to less than 120ºF within the next 8 hours or be in MODE 3 in the next 6 hours and consider venting containment.

D. Direct a start of start CFCU 14, restore pressure within limits within the next 4 hours or be in MODE 3 in the next 6 hours and consider venting containment.

Proposed Answer:

A. Start CFCU 14, and consider venting containment.

Explanation:

A correct, CFCU 14 should be started. Containment temperature (average) is less than TS 3.6.5 limit of 120ºF. Consideration of venting containment should be considered since amps on CFCU 15 are above 330.
B incorrect, containment pressure is not above the limit (1.2 psig).
C incorrect, Containment temperature (average = 118F) is less than TS 3.6.5 limit of 120ºF.
D incorrect, no guidance to stop CFCU 14.

Technical Reference(s):
PK01-16
Tech Spec 3.6.4
Tech Spec 3.6.5

Proposed references to be provided to applicants during examination:
PK01-16
Tech Spec 3.6.4
Tech Spec 3.6.5

Learning Objective: 9697F - Identify 3.6 Technical Specification LCOs

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
55.43 43.2

Comments:

K/A: 022 A2.04 - 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 service water

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be $\geq$ -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. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be $\leq$ 120°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. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	24 hours

TITLE: CONTMT ENVIRONMENT - PPC

09/04/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM

Y0405C ⟶ ALARM

2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT	PPC ADDRESS
Y0405C	437	Contmt Environment PPC		

The inputs to device Y0405C are as follows:

1. CONTMT PRESS, PT 937	P1000A
2. CONTMT PRESS, PT 936	P1001A
3. CONTMT PRESS, PT 935	P1002A
4. CONTMT PRESS, PT 934	P1003A
5. VESSEL SE SHIELD WALL TEMP, TT-720	T0720A
6. VESSEL SW SHIELD WALL TEMP, TT-721	T0721A
7. VESSEL NW SHIELD WALL TEMP, TT-722	T0722A
8. VESSEL NE SHIELD WALL TEMP, TT-723	T0723A
9. CONTMT TEMP (SELECTED)	Y0701A

3. PROBABLE CAUSE

3.1 High air temperature/pressure due to:

3.1.1 Steam leak

3.1.2 Missing insulation

3.1.3 Fan cooler problems

3.1.4 Instrument air/N_2 leakage

3.2 High shield wall temperatures due to:

3.2.1 Inadequate fan cooler flow

3.2.2 High CCW temperature

3.2.3 Inadequate CRDM fan cooler flow

4. AUTOMATIC ACTIONS

None

5. OPERATOR ACTIONS

5.1 Call up PK01-16 on the PPC and check for an out of limit conditions.

5.2 Check the CCW system for normal heat exchanger outlet temperature.

5.3 Check the CCW flow normal through each fan cooler unit.

5.4 If necessary place another fan cooler unit in service.

5.5 Verify all in service or start additional CRDM fan cooler.

5.6 If Contmt pressure is in alarm, refer to Technical Specification 3.6.1.4 (ITS 3.6.4) and consider venting containment if Contmt press is 0.8 psig or any CFCU in fast speed indicates greater than 330 amps.

5.7 If Contmt temp GT 120°F, refer to Technical Specification 3.6.1.5 (ITS 3.6.5).

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #	076 G2.2.22	
	Importance		4.1

Proposed Question:

Units 1 and 2 are at full power.

Both trains of ASW on Unit 2 are declared inoperable. Cross Tie valve FCV-601 has been opened to provide ASW to Unit 2.

Which of the following describes when, if at all, the units must be in MODE 3?

A. Unit 2 must be in MODE 3 in 6 hours. Unit 1 can remain at power indefinitely.

B. Unit 2 must be in MODE 3 in 6 hours. Unit 1 must be in MODE 3 in 78 hours.

C. Unit 2 must be in MODE 3 in 7 hours. Unit 1 can remain at power indefinitely.

D. Unit 2 must be in MODE 3 in 7 hours. Unit 1 must be in MODE 3 in 78 hours.

Proposed Answer:

D. Unit 2 must be in MODE 3 in 7 hours. Unit 1 must be in MODE 3 in 78 hours.

Explanation:

Per TS Bases B3.7.8 - In the event of a total loss of ASW in one unit, the capability to cross-tie units ensures the availability of sufficient redundant cooling capacity for the affected unit. If the unit cross-tie capability were used, <u>the unit with no operable ASW train would enter LCO 3.0.3, and the unit from which ASW was being provided would be in a 72-hour action with the cross-tie then declared inoperable</u>.

A incorrect, the time for Unit 1 is 7 hours, (3.0.3), Unit 2 must shutdown, starting in 72 hours, due to one train inoperable (because it is providing unit 1 ASW)
B incorrect, the time for Unit 1 is 7 hours, (3.0.3)
C incorrect, Unit 2 must shutdown, starting in 72 hours, due to one train inoperable (because it is providing unit 1 ASW)

D correct, per TS 3.0.3, Unit 1 must be in MODE 3 in 7 hours. Unit 2 can remain at power for 72 hours (TS 3.7.8 ACTION A) but then must be in MODE 3 in 6 hours (ACTION B) - 78 hours total.

Technical Reference(s):
Tech Spec 3.7.8
Tech Spec Bases B3.7.8

Proposed references to be provided to applicants during examination:
Tech Spec 3.7.8
Tech Spec 3.0.3

Learning Objective: 9697 – Identify Technical Specification LCOs

Question Source: Bank # DCPP R-55056
Modified Bank # ________
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 _____
55.43 43.2

Comments:

K/A: 076 G2.2.22 – Service Water - Knowledge of limiting conditions for operations and safety limits.

1 R-55056 Points: 1.00 Multiple Choice

Both trains of Unit 1 Auxiliary Saltwater System are INOPERABLE. The Unit 2 Auxiliary Saltwater System is cross-tied to provide cooling for Unit 1 loads.

How long can each unit operate in this configuration before MODE 3 must be achieved?

A. 7 hours for Unit 1, and 78 hours for Unit 2

B. 1 hour for Unit 1, and 72 hours for Unit 2.

C. 7 hours for Unit 1, and indefinitely for Unit 2.

D. 78 hours for both Unit 1 and Unit 2

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

9697	Identify Technical Specification LCOs

Reference Id: R-55056
Must appear: No
Status: Active
User Text: 9697.13ALLN
User Number 1: 3.30
User Number 2: 4.10
Difficulty: 2.00
Time to complete: 3
Topic: Two ASW Trains OOS, xtied to other unit
Cross Reference: ITS 3.7.8, LCO 3.0.3
Comment: This question generated for R996C10, RLF1,12/07/99

3.7 PLANT SYSTEMS

3.7.8 Auxiliary Saltwater (ASW) System

LCO 3.7.8 Two ASW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ASW train inoperable.	A.1 ------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, “RCS Loops - MODE 4,” for residual heat removal loops made inoperable by ASW. ----------------------------------- Restore ASW train to OPERABLE status	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify each ASW manual and power operated, valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.2	Verify each ASW power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, can be moved to the correct position.	In accordance with the Inservice Test Program.
SR 3.7.8.3	Verify each ASW pump starts automatically on an actual or simulated actuation signal.	24 months

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours; b. MODE 4 within 13 hours; and c. MODE 5 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.
LCO 3.0.4	When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or c. When an allowance is stated in the individual value, parameter, or other Specification. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

(continued)

3.0 LCO APPLICABILITY (continued)

LCO 3.0.5	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.
LCO 3.0.6	When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCO 3.1.8, allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per ..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

B 3.7 PLANT SYSTEMS

B 3.7.8 Auxiliary Saltwater System (ASW)

BASES

BACKGROUND	The ASW system provides a heat sink from the Pacific Ocean for the removal of process and operating heat from the CCW system. The CCW system then provides cooling to safety-related components during all modes of operation, including a DBA, and also to various non safety-related components during normal operation and shutdown. The ASW consists of two, 100% capacity, safety related, cooling water trains. Each train consists of one 100% capacity pump, one component cooling water (CCW) heat exchanger, piping, valving, and instrumentation. The pumps are automatically started upon receipt of a safety injection signal or 4kV automatic transfer. Normal configuration is for one train operation with the second pump cross-tied in stand-by and the second heat exchanger valved out-of-service except when the UHS temperature is 64°F or higher; therefore no valve realignment occurs with a safety injection signal. Manual and remote manual system realignment provides for utilization of the second CCW heat exchanger, for use of the standby pump on the same unit, for cross-tying the standby ASW pump from opposite unit, and for train separation for long term cooling. The ASW unit cross-tie valve (FCV-601) allows one ASW pump on one unit to supply the CCW heat exchanger(s) on the other unit. In the event of a total loss of ASW in one unit, the capability to cross-tie units ensures the availability of sufficient redundant cooling capacity for the affected unit. If the unit cross-tie capability were used, the unit with no operable ASW train would enter LCO 3.0.3, and the unit from which ASW was being provided would be in a 72-hour action with the cross-tie then declared inoperable. FCV-601 is controlled by ECG 17.1. Additional information about the design and operation of the ASW system, is presented in the FSAR, Section 9.2.7 (Ref. 1). The principal safety related function of the ASW system is the removal of decay heat from the reactor via the vital CCW System.
APPLICABLE SAFETY ANALYSES	The design basis of the ASW system is for one ASW train, in conjunction with the CCW System and the containment cooling systems, to remove accident generated and core decay heat following a design basis LOCA as discussed in the FSAR, Section 6.2 (Ref. 2). The ASW system can be re-configured to maintain the CCW temperature to within its design bases limits. The ASW system is designed to perform its function with a single failure of any active

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

component, with or without the loss of offsite power. This assumes a maximum ASW temperature of 64°F occurring simultaneously with maximum heat loads on the system. The ASW system, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of ASW pumps, CCW heat exchangers, and RHR heat exchangers that are operating. One ASW train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. However, in the split-train configuration during post-accident operation, operator action may be required to realign the ASW and CCW systems to prevent loss of all cooling to containment and safety-related systems following specific active failure scenarios.

The ASW system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ASW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ASW train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. The pump is OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

This requires that at least one vacuum relief valve be OPERABLE. Each ASW train has a vacuum relief system consisting of two vacuum relief valves (check valves) which function to prevent water hammer in the system piping during an ASW pump trip and restart transient. Check valves are passive components and, unless otherwise specified, are not considered in meeting the single failure criterion. The second vacuum relief valve on each header ensures reliability of the function. If both vacuum relief valves on a single header are inoperable, water hammer during an ASW pump trip and restart transient could affect both ASW trains unless the ASW header cross-tie valve is closed and the ASW pump breaker or dc control power switch is opened for the affected ASW train, precluding the potential for water hammer in the train. See ECG 17.4, "ASW Pump Discharge Vacuum Relief Valves."

(continued)

BASES

LCO (continued)

Both cross-tie valves FCV-495 and FCV-496 are required to be open to support single active failure criteria. The valves may be closed in post-accident long-term phase to support passive failure criteria, if system integrity is a concern. With one or both ASW trains in service with the cross-tie valves closed, a single active failure could result in a significant reduction or loss of heat removal capability. With both ASW trains in service, approximately one-half of the total CCW flow is routed through each CCW heat exchanger. In the event of a postulated ASW pump failure in this configuration, with the cross-tie valves closed, only one ASW pump will be operating and providing heat removal to one-half of the total CCW flow via its associated in-service CCW heat exchanger. In this situation, the ASWS heat removal capability is limited and may not meet the requirements of the system to maintain the CCW supply temperature within its design limits.

c. The associated pump vault drain check valve is OPERABLE. The ASW pump vault check valves prevent flooding of the ASW pump vaults during design flood events.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ASW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ASW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ASW system are determined by the systems it supports.

(continued)

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #	010 A2.03	
	Importance		4.2

Proposed Question:

PLANT CONDITIONS:

- Unit 1 is at 35% in a normal lineup
- All pressurizer heaters are "ON"
- Green lights for RCS Spray valves, PCV-455A and PCV-455B are "LIT"
- RCS pressure is 1945 psig and decreasing slowly
- Pressurizer tailpipe temperature is 135ºF
- Green and Red lights for Pressurizer PORVs PCV-474 are "LIT" with the switch in CLOSE

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

A. Direct the operator to close block valve 8000A, power should be left to 8000A.

B. Direct the operator to trip the reactor and enter E-0, Reactor Trip or Safety Injection and close block valve 8000A.

C. Direct the operator to close block valve 8000A and within one hour remove power from 8000A.

D. Direct the operator to manually initiate Safety Injection and enter E-0, Reactor Trip or Safety Injection.

Proposed Answer:

B. Direct a reactor trip and enter E-0, Reactor Trip or Safety Injection and close block valve 8000A.

Explanation:
A incorrect, this is done if the PORV cannot be isolated.
B correct, RCS pressure is at the trip setpoint, the reactor should be tripped, as part of isolation, the block valve should be closed in an effort to stop the depressurization.
C incorrect, this action would be if the reactor was not tripped and the PORV was inoperable for causes other than seat leakage.
D incorrect, correct if only the PORV had to be isolated.

Technical Reference(s):
OP AP-13, Malfunction of Reactor Pressure Control System
OIM B-6-4b
TS 3.4.11

Proposed references to be provided to applicants during examination: none

Learning Objective: 4587 – Explain the effect of failures on the Pressurizer

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
55.43 43.2

Comments:

K/A: 010 A2.03 - 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: PORV failures

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable solely due to excessive seat leakage.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable for reasons other than excessive seat leakage.	B.1 Close associated block valve.	1 hour
	AND	
	B.2 Remove power from associated block valve.	1 hour
	AND	
	B.3 Restore the Class I PORV to OPERABLE status.	72 hours
C. One block valve inoperable.	----------------NOTE------------------- Required Actions do not apply when block valve is inoperable solely as result of complying with Required Actions B.2 or E.3. ---	1 hour
	C.1 Place associated PORV in manual control.	
	AND	(continued)

TITLE: Malfunction of Reactor Pressure Control System

1 AND 2

06/04/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. SCOPE

1.1 This procedure provides instructions in the event of a Reactor Pressure Control malfunction.

2. SYMPTOMS

2.1 Pressurizer pressure unexpectedly high or low.

2.2 Possible Main Annunciator Alarms:

2.2.1 PZR PRESSURE LOW (PK05-17)

2.2.2 PZR PRESSURE HIGH (PK05-16)

2.2.3 PZR SAFETY OR RELIEF LINE TEMP (PK05-23)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **STOP any load changes in progress**	
2. **CHECK PZR PORVs - CLOSED** • PCV-474 • PCV-455C • PCV-456	IF PZR pressure LESS THAN 2335 psig, THEN Close PORV(s). IF Any PORV can NOT be closed, THEN Close its block valve. 8000A for PCV-474 8000B for PCV-455C 8000C for PCV-456 IF Block valve can NOT be closed, THEN Manually Initiate Safety Injection AND GO TO EOP E-0.
3. **CHECK PZR Safety Relief Valves**: a. Sonic flow indicators - 0 b. Tail pipe temp ≤185°	IF A PZR safety has lifted and will not reseat, THEN Manually Initiate Safety Injection AND GO TO EOP E-0 IF A PZR safety is leaking, THEN Refer to AR PK05-23.

Reactor Trip Signals (Continued)

Pressurizer System Signals

Trip	Setpoint**	Coincidence	Interlock	Protection Afforded
Low Pressure	1950psig	2/4	P-7	* DNB
High Pressure	2385psig	2/4		* RCS Integrity
High Level	90%	2/3	P-7	Prevent water relief RCS Integrity

Secondary System Signals

Trip	Setpoint**	Coincidence	Interlock	Protection Afforded
Steam Generator Low-Low Level	15%***	2/3 Sensors of 1/4 Loops		* Loss of Heat Sink
Turbine Trip	Auto Stop Oil <50psig -OR- Stop Valves Closed	2/3 4/4	P-9	* Limit temperature/pressure transients on the RCS

Miscellaneous Signals

Trip	Setpoint**	Coincidence	Interlock	Protection Afforded
Manual		1/2		Operator Judgement
Safety Injection		Any "S" Signal		* Limit the consequences of accidents
Seismic	0.30g	2/3 sensors (In the same direction)		Trip the reactor in the event of a double design earthquake
SSPS General Warning	Train A&B SSPS simultaneously	2/2		Trip RX in event of failure of protection system

* Protection assumed in FSAR accidents.

** Reset values are approximately 1% different than the setpoint value (see the appropriate setpoint documentation for the exact values).

*** An increasing time delay to trip is allowed, from 50% power to 0% power (based on loop delta-T) from approximately 27.8 to 464.1 seconds (non-linear), once SG level drops under 15%. There is no time delay from 50% to 100% power. The Steam Generator level transmitters do not correlate directly to the associated RCS loop delta-T used to generate the timing signal (i.e., LT-529, LT-539, and Loop 1 delta-T are used to generate the error signal for Protection Set I).

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	2
	K/A #	014 G2.4.10	
	Importance		3.1

Proposed Question:

While a load rejection is occurring, the operator reports PK03-21, DRPI FAILURE/ROD BOTTOM has alarmed and there is no indication on the DRPI panel. Indicated rod speed is 72 steps per minute.

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

A. Direct the operator to place rods in MANUAL and implement AP-6, Emergency Boration.

B. Direct the operator to monitor Group Step demand counters.

C. Direct a reactor trip and enter E-0, Reactor Trip or Safety Injection.

D. Direct the actions of AP-25, Rapid Load Reduction while performing the actions of the annunciator response as time permits .

Proposed Answer:

C. Direct a reactor trip and enter E-0, Reactor Trip or Safety Injection.

Explanation:
Per PK03-21:
5.2 If DRPI power failure:
- 5.2.1 Place Rod Control in MANUAL.
- 5.2.2 Refer to ITS 3.1.7.
- 5.2.3 Place DRPI on backup per OP A 3:I, Section 6.1.
- 5.2.4 If power cannot be restored:
  - a. Continue attempts to restore power to DRPI.
  - b. Notify Operations Management.
  - c. If evidence of a plant transient occurs:
    1. With Shift Foreman concurrence, trip the Reactor, and
    2. GO TO EOP E-0.

C correct, due to the rods moving at max speed and a plant transient in progress, the reactor should be tripped.

Technical Reference(s):
PK03-21, DRPI FAILURE/ROD BOTTOM

Proposed references to be provided to applicants during examination: none

Learning Objective:
4918 – State the DRPI parameters that produce alarms.

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 ______
55.43 43.6

Comments:
K/A: 014 G2.4.10 – Knowledge of annunciator response procedures.

TITLE: DRPI FAILURE/ROD BOTTOM

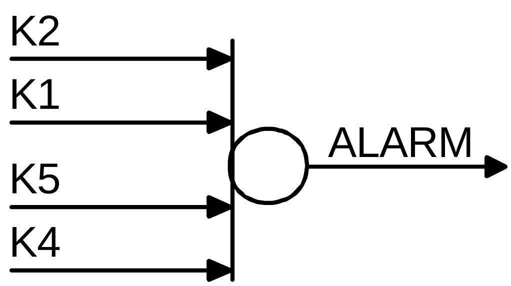
1

02/09/01
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
K2	478	Rod Pos Ind Rod Bottom	LT 6 steps
K1	551	Rod Pos Ind Rods at Bottom	LT 6 steps
K5	552	Rod Pos Ind Non-Urgent	Loss of RPI data A or B
K4	553	Rod Pos Ind Urgent	Loss of RPI data A & B

3. PROBABLE CAUSE

3.1 One or more dropped control rods.

3.2 Rod position channel data coil failure.

3.3 DRPI power transient.

#

#

4. AUTOMATIC ACTIONS

4.1 Probable reactor trip if one or more rods are dropped at power.

4.2 Loss of capability of every other position indicating light on the control rod with a failed data coil.

4.2.1 Flashing general warning light above affected rod(s).

4.2.2 Flashing data channel A or B failure lights.

4.2.3 Loss of all position indicating lights on a control rod with a failure of both data coils.

a. Rod urgent alarm light.

b. Rod bottom alarm and light.

c. Flashing general warning light.

d. Flashing data channel A and B failure lights.

5. OPERATOR ACTIONS

5.1 If reactor trips, go to EOP E-0.

5.2 If DRPI power failure:

5.2.1 Place Rod Control in MANUAL.

5.2.2 Refer to ITS 3.1.7.

5.2.3 Place DRPI on backup per OP A-3:I, Section 6.1.

5.2.4 If power cannot be restored:

a. Continue attempts to restore power to DRPI.

b. Notify Operations Management.

c. If evidence of a plant transient occurs:

1. With Shift Foreman concurrence, trip the Reactor, and

2. GO TO EOP E-0.

#

#

5.3 GO TO OP AP-12C for a dropped rod without a reactor trip.

5.4 For loss of one data channel (Rod Pos Ind Non-Urgent), the SFM may choose to select the OPERABLE channel (A ONLY or B ONLY) with switch S106 on the back of the DRPI panel while initiating troubleshooting and repair of the failed channel. This will place DRPI in "half accuracy" for all control rod positions.

5.5 If BOTH A and B data channels on one or more rods in any bank have failed, place Rod Control in MANUAL and refer to ITS 3.1.7.

NOTE: Per DRPI tech. manual, if both channels are indicated failed it is possible due to the electronic system not knowing which is the good channel. This may be remedied by selecting first one channel, then the other to see which works. Regardless, whenever loss of both data channels are encountered, action step 5 above must be followed.

SRO QYESTION 92

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034 A2.01	
	Importance		4.4

Proposed Question:

PLANT CONDITIONS:

- Core offload is in progress
- The manipulator crane is transferring a spent fuel assembly from the reactor vessel to the transfer canal
- The assembly is currently close to the transfer canal.

CVI actuates due to high radiation.

Which of the following describes what the Refueling SRO should direct be done with the spent fuel assembly?

A. Return it to its original core position.

B. Leave the assembly in the mast of the manipulator crane.

C. Place the assembly in the containment side upender and lower the upender.

D. Place the assembly in the containment side upender, lower the upender, and send the assembly into the fuel handling building.

Proposed Answer:

C. Place the assembly in the containment side upender and lower the upender.

Explanation:
A incorrect, if closer to the core, the fuel assembly is placed any open core position.
B incorrect, this is not an option.
C correct, if close to the upender, AP-21 placed in the upender and lowered.
D incorrect, not a required action.

Technical Reference(s): AP-21, Irradiated Fuel Damage.

Proposed references to be provided to applicants during examination: None

Learning Objective: 6619 Explain the actions for fuel damage, actual or suspected
Question Source: Bank # DCPP P-1729

Question History: Last NRC Exam DCPP RO 10/94

Question Cognitive Level:

Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.13
55.43 _____

Comments:

K/A: 034 A2.01 – Fuel Handling Equipment – Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped fuel element.

1 P-1729 Points: 1.00 Multiple Choice

PLANT CONDITIONS:

- Refueling is in progress
- The manipulator crane is transferring a spent fuel assembly from the reactor vessel to the transfer canal
- The assembly is currently close to the transfer canal.

RE-44A and 44B go into high alarm and CVI actuates.

Which ONE of the following describes what should be done with the spent fuel assembly?

A. Place the assembly in the containment side upender and lower the upender

B. Leave the assembly in the mast of the manipulator crane

C. Place the assembly in the containment side upender

D. Place the assembly in the containment side upender, lower the upender, and send the assembly into the fuel handling building

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

6619	Explain the actions for fuel damage, actual or suspected

Reference Id: P-1729
Must appear: No
Status: Active
User Text: 6619.110341
User Number 1: 3.00
User Number 2: 4.10
Difficulty: 2.00
Time to complete: 3
Topic: LPA-21 Fuel handling accident actions
Cross Reference: LPA21, OP AP-21
Comment: DCPP RO EXAM 10/94
modified due to RE-7 going into high alarm is a remote chance given the condition of being in refueling. Changed to RE-44A and B to drive home a problem in containment atmosphere.

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 <u>If in Unit 1</u>

CONTMT RADIATION (PK11-19)

2.5.5 <u>If in Unit 2</u>

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.)

ACTION / EXPECTED RESPONSE		RESPONSE NOT OBTAINED	
1. SUSPEND Core Alteration/Fuel Movement			
2. CHECK Fuel Assembly (F/A) Is In Safe Position: • F/A is resting in a core position OR • F/A is resting in a Spent Fuel Pool grid position OR • F/A is in the Upender and the Upender has been lowered		Place the F/A in a safe position per direction of the refueling SRO.	
3. DIRECT Refueling Crew Members To Get Off Manipulator Crane Or Spent Fuel Pool Bridge Crane, As Applicable			
4. CHECK Radiation Monitors – NOT in alarm			
a. CONTMT Radiation **–** NORMAL		IF	Containment radiation alarms are determined to be valid by the Refueling SRO or SFM,
– Containment Area Monitor RE-2 NORMAL		THEN	1) Activate CONTMT evacuation alarm
– Containment Radiogas Monitor RE-12 NORMAL			2) Evacuate personnel from Containment
– Containment Vent Monitor RM44A/RM44B			3) Initiate Containment Closure (AD8.DC54)
– PORTABLE Cams			4) VERIFY Containment Ventilation Isolation • Containment Purge Exhaust (RCV-11,12) – Closed • Containment Purge Supply (FCV-660,661) – Closed • Containment Pressure Vacuum Relief (FCV-662, 663, 664) – Closed • Containment Radiogas Sample Valves (FCV-678, 679, 681) – Closed

– THIS STEP CONTINUED ON NEXT PAGE –

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. CHECK Radiation Monitors – NOT in alarm (Continued)	5) Start fan E-15 and E-16, CONTMT Iodine Removal Units 6) Secure fans E-3/S-3. 7) Observe plant vent Rad monitors for any potential off-site releases. IF Readings indicate GREATER THAN NORMAL, THEN Perform EP R-2 calculations.
b. FHB Radiation – NORMAL • RE-58 AND RE-59 • Portable CAMS	b. Perform the following: 1) Evacuate the FHB. 2) Select Iodine Removal for the FHB Vent system. 3) Observe plant vent rad monitors for any potential off-site releases. IF Readings indicate GREATER THAN NORMAL, THEN Perform EP R-2 calculations.
5. VERIFY Ventilation Alignments:	
Damper failure panel lights – OFF	Select S Signal Test on POV-1 AND POV-2: IF Damper failure panel lights are still on, THEN Perform the following: a. On RCV-1 AND RCV-2, select POV-1 CONTROL AND POV-2 CONTROL switches – OFF. b. Select ONE exhaust AND ONE supply fan to ON to restore the Aux Bldg Vent System: 1) E-1 OR E-2 2) S-31 OR S-32 c. Select ONE exhaust AND ONE supply fan to ON to restore the Fuel Handling Bldg Vent System: 1) E-5 OR E-6 2) S-1 OR S-2

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. CONFIRM With Radiation Protection that Radiological Condition of The Affected Building Is Acceptable For Occupancy	IF Radiological conditions are NOT acceptable, THEN Prohibit reentry of personnel to the building until condition becomes acceptable for occupancy as verified by Radiation Protection – OR – As permitted and directed by Radiation Protection via an SWP if reentry is required.
7. DETERMINE Extent Of Fuel Damage: a. Check for torn or missing grid strap(s) b. Check for excessive fuel rod misalignment c. Check for loose fuel pellets	
8. IMPLEMENT Long Term Recovery Actions prescribed by Plant Management	

– END –

3. APPENDICES

None

4. ATTACHMENTS

None

5.2.2 In the event of a required halt in fuel movement, fuel assemblies should not be left suspended, but moved to a suitable and safe location, such as a supported core location or the upender. A corner baffle is the preferred core location.

5.2.3 In the event of a refueling cavity leak indicated by rapidly decreasing level, the Refueling SRO shall ensure that the fuel element in transit is stored at the lowest elevation possible.

a. If the fuel element is in the upender, lower the upender.

b. If the fuel element is in the manipulator and over the core area, lower the element into the core.

c. Close the fuel transfer tube gate valve (SFS-50), shut the SFP swing gate and inflate the gate seal.

5.2.4 If a containment evacuation alarm occurs, CORE ALTERATIONS shall be suspended immediately and all personnel shall assemble in the main airlock. The Refueling SRO shall inquire about the cause of the alarm and determine the response to be taken. If it is determined that no hazards to personnel exist, evacuation need not proceed any further.

5.2.5 If a portable radiation monitor alarm occurs on the manipulator crane while handling fuel, CORE ALTERATIONS shall be suspended immediately and refueling personnel shall commence evacuation after placing the fuel in a safe condition. The Refueling SRO shall make a determination of the cause of the alarm and, if the alarm appears valid, the Control Room shall be notified to sound the Containment Evacuation Alarm. If it is determined that no hazards to personnel exist, evacuation need not proceed any further.

5.2.6 If the Refueling SRO or power production engineer (Nuclear) suspects that continued operation will involve undue risk to personnel or equipment or will compromise the Technical Specifications or license provisions, operations will be suspended pending resolution.

5.2.7 If tornado or severe weather warnings go into effect, fuel handling activities shall be suspended and action taken immediately to close the equipment hatch.

SRO QUESTION 93

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	______	2
Group #	______	2
K/A #	011 A2.05	
Importance		3.7

Proposed Question:

A load rejection occurs on Unit 1.

The crew has entered AP-25, Rapid Load Rejection. RCS pressure is 2260 psig. AR PK 05-22, PZR LEVEL HI/LO CONTROL is in alarm.

When checking Pressurizer heaters, the CO reports all backup pressurizer heaters are OFF.

What action should the SFM take?

A. Direct the heaters energized to ensure mixing of RCS and pressurizer boron.

B. Direct the heaters energized to heat the subcooled water entering the pressurizer.

C. Continue with the procedure, high RCS pressure is the cause for the heaters not being energized.

D. Continue with the procedure, heaters should not be energized to prevent possibly lifting a pressurizer PORV.

Proposed Answer:

B. Direct the heaters energized to heat the subcooled water entering the pressurizer.

Explanation:
A incorrect, this is true when borating or diluting
B correct, heaters should turn on when level is +5 above program (as during a load rejection), this heats the subcooled water entering the pressurizer.
C incorrect, normally heaters would be off and this would be appropriate for normal plant conditions.
D incorrect, heaters should be energized.

Technical Reference(s):
OP AP-25, Rapid Load Rejection,
STG A4A, Pressurizer
AR PK 05-22

Proposed references to be provided to applicants during examination: none

Learning Objective:
4514 - Explain the bases for Pressurizer Level Control operations.

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
55.43 43.5

Comments:

K/A: 011 A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of PZR heaters

ACTION / EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
1. REDUCE Turbine Load:	
a. IF Turbine runback or programmed ramp is in progress, THEN Go to step 2	
b. Place DEH MW and IMP feedbacks in service	
c. Set TARGET to desired load	

NOTE: DFWCS should satisfactorily control S/G levels in AUTO **IF** the ramp rate is kept below 225 MW/Minute.

ACTION / EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
d. Set desired RAMP RATE per SFM discretion	
e. Push GO	

NOTE: Maintain rods above RIL and AFD within the doghouse.

ACTION / EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
2. VERIFY Control Rods Inserting in AUTO	Manually insert control rods to maintain TAVG and TREF within 5°F.
3. VERIFY PZR Backup Heaters - ON	
4. VERIFY at Least One CCP In Service	
5. VERIFY DFWCS Controlling S/G Levels in AUTO:	
a. Verify MFW control valves in AUTO	a. Control S/G levels in MANUAL.
b. MFP speed controllers in AUTO	b. Control FW/STM HDR ΔP in MANUAL.
6. BORATE RCS: Refer to the Reactivity Handbook to determine the quantity of boric acid to add	

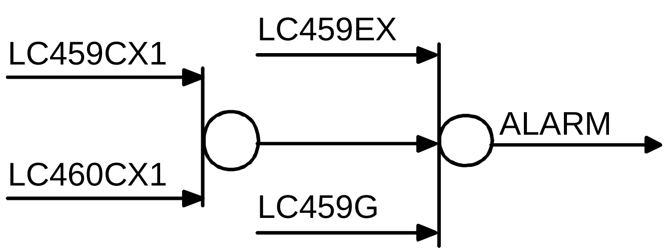
TITLE: PZR LEVEL HI/LO CONTROL

09/15/99
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

#

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
LC 459EX	543	PZR Lvl Dev High From REF Backup Htrs On	+ 5% GT LREF
LC 459CX1	544	PZR Lo Lvl Letdn Iso All Htrs Off	LT 17%
LC 460CX1	544	PZR Lo Lvl Letdn Iso All Htrs Off	LT 17%
LC 459G	315	PZR High Charging Flow Demand	8% Below Level Program

3. PROBABLE CAUSE

3.1 Reduction or increase in plant load.

3.2 Malfunction of rod control system.

3.3 Malfunction of pressurizer level control system.

3.4 Malfunction of pressurizer pressure control system.

3.5 T_{avg} signal failing high.

3.6 Overfeeding or underfeeding the steam generators.

3.7 RCS loss of coolant accident, steam generator tube rupture, or loss of secondary coolant accident.

3.8 RCS leak.

3.9 Insufficient charging flow.

3.10 Failure of charging or letdown line.

Level Control Circuit, Continued

Bases for energizing heaters on +5% level deviation
Obj 42

- The level deviation-high is assumed to be an insurge produced by a decrease in load. Subcooled water enters the Pressurizer.
- If a subsequent outsurge were to occur, the subcooled water would not assist in maintaining pressure by flashing to steam.
- It is conservatively assumed that a subsequent outsurge will occur. The backup heaters are energized as an anticipatory measure.
- This feature also helps during other insurge transients as the backup heaters will be needed to heatup the larger than normal amount of cold water entering the pressurizer to normal saturation temperature (whether outsurge occurs or not, the heaters will turn on in anticipation of saturation temperature dropping).

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	______	3
	Group #	______	1
	K/A #	G2.1.10	
	Importance		3.9

Proposed Question:

The crew is responding to a reactor trip initiated by an earthquake tripping all 3 channels of the seismic monitors.

Which of the following describes the action the crew must take, if any, regarding the fire detection instrumentation to meet the requirements of ECG 18.3, Fire Detector Instrumentation?

A. Patrols of fire zones listed in ECG 18.3 must be initiated within 2 hours.

B. Inspection for fires in fire zones listed in ECG 18.3 must be completed within 2 hours.

C. Patrols of fire zones with equipment in operation and listed in ECG 18.3 must be initiated within 2 hours.

D. No action is required, the threshold as stated in ECG 18.3 has not been met.

Proposed Answer:

B. Inspection for fires in fire zones listed in ECG 18.3 must be completed within 2 hours.

Explanation:

A incorrect, inspection must be complete in 2 hours.
B correct, per clarification sent 10/18/2004, "..all zones listed in ECG 18.3 have to be inspected for fires within 2 hours. ..It is a requirement of our facility license to have this inspection complete in 2 hours."
C incorrect, all zones listed must be patrolled.
D incorrect, the threshold (>0.02 g) has been exceeded, trip setpoint is 0.03 g.

Technical Reference(s):
ECG 18.3
Email from Mark Lemke dated 10/18/2004

Proposed references to be provided to applicants during examination: ECG 18.3

Learning Objective: 66056 - Discuss the requirements of System 18 ECGs.

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 ______
55.43 43.2

Comments:
K/A: G2.1.10 – Knowledge of conditions and limitations in the facility license.

18.0 FIRE PROTECTION

18.3 Fire Detection Instrumentation

ECG 18.3 The fire detection instrumentation for each fire detection zone shown in Table 18.0-3 shall be OPERABLE*.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE*.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. With less than the minimum number of required instruments operable per Table 18.0-3, for one or more zones outside containment.	A.1	Establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s).	1 hour AND Once per hour thereafter.
B. With less than the minimum number of required instruments operable per Table 18.0-3, for one or more zones inside containment.	B.1	Establish a fire watch patrol to inspect the containment.	8 hours AND Once per 8 hours thereafter.
	OR		
	B.2	Monitor the containment air temperature at locations TE-85 or TE-86[1], TE-87 or TE-88[2], TE-89 or TE-90[3], and TE-91 or TE-92[4].	1 hour AND Once per hour thereafter.

(continued)

1 Approximately 100 ft elevation between crane wall and containment wall.

2 Approximately 100 ft elevation between steam generators.

3 Approximately 140 ft elevation near equipment hatch or stairs at 270°, respectively.

4 Approximately 184 ft elevation on top of steam generator missile barriers away from steam generators.

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Following a seismic event in excess of 0.02g.	C.1 Inspect the zones in Table 18.0-3 for fires.	2 hours
	AND	
	C.2 Perform an engineering evaluation to verify the OPERABILITY* of the Fire Detection System.	72 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 18.3.1	Perform a CHANNEL FUNCTIONAL TEST* of each of the required fire detection instruments which are accessible during plant operation.	6 months
SR 18.3.2	Perform a CHANNEL FUNCTIONAL TEST* of each of the required fire detection instruments which are not accessible during plant operation.	During each COLD SHUTDOWN* exceeding 24 hours unless performed in the previous 6 months.
SR 18.3.3	Demonstrate the OPERABILITY* of the supervision circuitry associated with the detector alarms of each of the required fire detection instruments.	6 months
SR 18.3.4	For SSPS room detectors connected to panel POFC, verify the system (detectors, control room alarm and panel) actuate automatically upon receipt of a simulated test signal (STP M-19B).	18 months

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

BASES

The OPERABILITY* of the detection instrumentation ensures that adequate warning capability is available for prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage of safety-related equipment and is an integral element in the overall facility Fire Protection Program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY*.

Action C.1 requires inspecting the zones in Table 18.0-3 for fires within 2 hours following a seismic event in excess of 0.02g. The basis for the 2-hour Completion Time is that fire detectors are nonseismic and cannot be relied upon to detect fires after an earthquake. Since safe shutdown systems are protected by barriers rated at two hours or more, any fire after an earthquake should be detected by this inspection before redundant safe shutdown equipment is affected (Reference 7). For the purposes of this inspection, entry into a closed room or enclosure, including containment, is sufficient to meet the requirement to "inspect for fires." Presence of a fire (smoke, odor, flame, heat) would be self evident upon entry.

Some fire detectors are located in areas not normally accessible during plant operation. Detectors not normally accessible during plant operations, such as in containment, will be tested (channel functional test) during each COLD SHUTDOWN* exceeding 24 hours unless the surveillance has been performed in the previous six months. Under these conditions, surveillance requirement 18.3.2 must be satisfied prior to leaving COLD SHUTDOWN* and re-entering MODE 4.

Table 18.0-3 includes detectors credited to protect safety-related equipment and equipment required for safe shutdown. The detectors which protect safe shutdown equipment are credited in the 10 CFR 50, Appendix R Safe Shutdown Analysis and, if credited, in the technical basis of approved deviations from 10 CFR 50, Appendix R, Section III.G. Some of the detectors are also credited in the technical basis for Fire Hazards Appendix R Evaluations.

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

(continued)

BASES (continued)

REFERENCES

1. FSAR Update Appendices 9.5A and 9.5H
2. Emergency Plan Implementing Procedure, EP M-4, "Earthquake"
3. License Amendment Request 90-11, "Revision of Fire Protection License Conditions, Relocation of Fire Protection Technical Specifications, and Clarification of AFW Water Sources"
4. Supplement No. 23 to Safety Evaluation Report (SSER 23) for Diablo Canyon Power Plant, Approved Deviations from the Requirements of Section III.G of Appendix R of Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50) for Unit 1.
5. SSER No. 31 for Diablo Canyon Power Plant, Approved Deviations from the Requirements of Section III.G of 10 CFR 50, Appendix R, for Unit 2.
6. Calculation M-928, 10 CFR 50 Appendix R Safe Shutdown Analysis
7. SSER No. 8 for Diablo Canyon Power Plant, Section 9.6.1, Fire Protection System - Seismically Induced Fires.

11/02/04
Effective Date

TABLE 18.0-3

FIRE DETECTION INSTRUMENTS

PANEL A

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE	MINIMUM INSTRUMENTS OPERABLE HEAT OR FLAME
1.	Cable Spreading Room	10	4[(1)]
3.	4kV Switchgear Bus F Room	1	N.A.
	4kV Switchgear Bus G Room	1	N.A.
	4kV Switchgear Bus H Room	1	N.A.
	4kV Switchgear Ventilation Fan Room	1	N.A.
	Exciter Field Breaker Room	2	N.A.
4.	4kV Switchgear Bus F Cable Spreading Area	1	N.A
	4kV Switchgear Bus G Cable Spreading Area	1	N.A.
	4kV Switchgear Bus H Cable Spreading Area	1	N.A.
	107' Corridor at Outside Wall (North/South)	1	N.A.
5.	12 kV Switchgear Room	8	N.A.
6.	73' Turb Bldg, 4/12kV Cable Spreading Room	8	N.A.
7.	Containment Electrical Penetration Zone 7	4	N.A.
8.	Containment Electrical Penetration Zone 8	3	N.A.
9.	Rod Control Programmer/Reactor Trip Breaker Area	3	N.A.
	Battery No. 1 Charger Room	1	N.A.
	Battery No. 2 Charger Room	1	N.A.
	Battery No. 3 Charger Room	1	N.A.
10.	480 Volt Bus F Area	1	N.A.
	480 Volt Bus G Area	1	N.A.
	480 Volt Bus H Area	1	N.A.
	Hot Shutdown Panel	1	N.A.
12.	Inside Containment	17[(2)]	N.A.
13.	Control Room Ventilation Return/Exhaust	1	N.A.
14.	Control Room Ventilation Return/Exhaust	1	N.A.
15.	Outside the Auxiliary Salt Water Pump Room	1	N.A.

TABLE 18.0-3 (Continued)

FIRE DETECTION INSTRUMENTS

PANEL B

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE	MINIMUM INSTRUMENTS OPERABLE HEAT OR FLAME
1.	Residual Heat Removal Pump No. 1 Room	1	N.A.
	Residual Heat Removal Pump No. 2 Room	1	N.A.
2.	Component Cooling Water Pump No. 1 Room	1	N.A.
	Component Cooling Water Pump No. 2 Room	1	N.A.
	Component Cooling Water Pump No. 3 Room	1	N.A.
	Charging Pump No. 1 Area	1	N.A.
	Charging Pump No. 2 Area	1	N.A.
	Charging Pump No. 3 Area	1	N.A.
	Containment Spray Pump No. 1 Area	1	N.A.
	Containment Spray Pump No. 2 Area	1	N.A.
	73' Aux Bldg East-West Corridor	5	N.A.
3.	Safety Injection Pump No. 1 Room	1	N.A.
	Safety Injection Pump No. 2 Room	1	N.A.
	Boric Acid Evaporator Area	1	N.A.
4.	Chemistry Lab and adjacent Rooms	10	N.A.
	CCW Heat Exchanger Room	2	N.A.
	Electrical Raceway Space G Bus	2	N.A.
	Electrical Raceway Space H Bus	1	N.A.
5.	Auxiliary Feedwater Pump No. 1 Area	1	N.A.
	Auxiliary Feedwater Pumps Nos. 2 & 3 Area	1	N.A.
	Boric Acid Transfer Pumps Area	1	N.A.
6.	Fire Pumps Area(5)	1	N.A.
	Unit 2 Auxiliary Building Supply Fan Room	1	N.A.
	Control Room Ventilation Equipment Room	1	N.A.
7 & 8.	Auxiliary Building Ventilation System Charcoal Filter Bank, EFC-1	12	6
11.	Fuel Handling Building Ventilation System Charcoal Filter Bank, EFC-5	6	3
12.	Fuel Handling Building Ventilation System Charcoal Filter Bank, EFC-6	6	3

TABLE 18.0-3 (Continued)

FIRE DETECTION INSTRUMENTS

PANEL B

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE	MINIMUM INSTRUMENTS OPERABLE HEAT OR FLAME
13.	Control Room - Control Console	3	N.A.
	Control Room Board	10	N.A.
14.	PPC/SSPS/SFM Office	4	N.A.
15.	Control Room - Radiation Monitoring	2	N.A.
	Control Room Nuclear Instrumentation	3	N.A.
16. (1)	Unit 1 Auxiliary Building Supply Fan Room	1	N.A.
	Control Room Ventilation Equipment Room	1	N.A.
16. (2)	Boric Acid Tanks Area	1 (3)	N.A.
Not Assigned to Zone	Diesel Generator No. 1 Room	N.A.	2 (1)
	Diesel Generator No. 2 Room	N.A.	2 (1)
	Diesel Generator No. 3 Room	N.A.	2 (1)
Not Assigned to Zone	Circulating Water Pump 1-1	N.A.	2 (1)
	Circulating Water Pump 1-2	N.A.	2(1)
	Circulating Water Pump 2-1	N.A.	2 (1)
	Circulating Water Pump 2-2	N.A.	2 (1)
Not Assigned to Zone	Solid State Protection System Room (panel POFC)	3 (4)	N.A.

TABLE 18.0-3 (Continued)

FIRE DETECTION INSTRUMENTS

PANEL D

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE	HEAT OR FLAME
6.	Access Control and Adjacent Rooms	28	N.A.
	Electrical Raceway Space H Bus	1	N.A.
	Electrical Raceway Space G Bus	2	N.A.

TABLE NOTATIONS

(1) Heat sensors actuate CO_2 flooding and are tested per ECG 18.5.1.3, 18.5.1.4, and 18.5.2.5. ECG 18.3.1, 18.3.2, and 18.3.3 do not apply.

(2) The fire detection instruments located within the containment are not required to be OPERABLE* during the performance of Type A Containment Leakage Rate Tests.

(3) Unit 1 Boric Acid Tank Detectors are in Zone 16, Unit 2.

(4) Smoke sensors actuating POFC and alarms are tested per ECG 18.3.4. ECG 18.3.1, 18.3.2, and 18.3.3 do not apply.

(5) The fire pumps 0-1 and 0-2 are common to both units and are located on the Unit 1 side and on the Unit 1 Fire Detection Instrument Panel only.

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

Fortier, Ronald

rom: Burns, David
Sent: Monday, October 18, 2004 1:58 PM
To: DCPP LRN Operations Training
Subject: FW: Fire Inspections Following an Earthquake

FYI

-----Original Message-----
From: Lemke, Mark S
Sent: Monday, October 18, 2004 1:47 PM
To: DCPP OPS Shift Managers; Gouveia, David
Cc: Zawalick, Maureen R; Becker, James; Roller, Paul James; Tomkins, James; Ketelsen, Stan C; Haynes, Joseph; Radford, James
Subject: Fire Inspections Following an Earthquake

Shift Managers, Dave Gouveia;

There has been much discussion in the last few months surrounding earthquakes and the requirement to establish a fire watch patrol within two hours of an earthquake of 0.02g or greater. The question is whether this means "initiate" the patrols, or have them "completed" in 2 hours. ***The answer is that all zones listed in ECG 18.3 have to be inspected for fires within 2 hours.*** This is more than just containment. It's many zones in the plant. This means we are looking for evidence of fires (smoke, flames, heat). The following is a discussion from Reg Services with regards to the requirements. We all know how challenging doing this in 2 hours can be. But it is a ***requirement*** of our facility license to have this inspection ***complete*** in 2 hours. It's not optional and it is not an "administrative requirement" that we can change. We will have to figure out a way, as a group, to be able to accomplish this. Both the IFOs and the Operators will need to work together to be able to get this done.

The point of this email is to clarify the need to do this. Please read the discussion below and apply it to the next time we have an earthquake. CP M-4 is being revised to include lessons learned from the last Parkfield earthquake. This requirement is being clarified in there.

Mark Lemke

DISCUSSION

A review of the licensing history of ECG 18.3 states in part:

b. Within 2 hours following a seismic event in excess of 0.02 G, each zone shown in Table 3.3-11 shall be inspected for fires...

The basis for TS 3/4.3.3.8 Action b is addressed in Supplemental Safety Evaluation Report No. 8, dated November 1978, (page 9-12):

> "Since the fire detectors are nonseismic and cannot therefore be relied upon to detect fires after an earthquake, we will require in the plant Technical Specifications that the applicant inspect the plant for fires within two hours following an earthquake. Since safe shutdown systems are protected by barriers rated at two hours or more, any fire after an earthquake should be detected by this inspection before redundant safe shutdown would be affected. Based on our review and the Technical Specifications requirements, we find the procedures for detection of fires after an earthquake acceptable."

Conditions	Required Action	Completion Time
C. Following a seismic event in excess of 0.02 g.	C.1 Establish a fire watch patrol to inspect the zones in Table 1.0-3 for fires	2 hours

NOTE

In general, safe shutdown systems are protected by barriers rated at 2 hours or 3 hours. These zones could be prioritized requiring the 2-hour zones to be inspected first, followed by the 3-hour zones. On this basis, the Completion Time for the 3-hour zones could be extended to 3 hours. Additionally, fire protection may be able to create a walkdown path that directs the fire watch patrol in the most efficient sequence.

SRO QUESTION 95

Question Worksheet

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	______	3
Group #	______	1
K/A #	G2.1.22	
Importance		3.3

Proposed Question:

Of the 3 conditions listed below, which of the following causes a MODE change from MODE 6 to MODE 5?

1. RCS temperature increases above 200ºF.
2. One reactor vessel head closure bolt is fully tensioned.
3. The last reactor vessel head closure bolt is fully tensioned.

A. Condition 2 only.

B. Condition 3 only.

C. Condition 1 or condition 2.

D. Condition 1 or condition 3.

Proposed Answer:

B. Condition 3 only.

Explanation:
A incorrect, this is the mode change from 5 to 6.
B correct, to transition from mode 6 to mode 5, all reactor vessel head closure bolts are fully tensioned.
C and D incorrect, no mode change if temperature increases above 200ºF

Technical Reference(s): DCPP Tech Specs section 1.1, Definitions.

Proposed references to be provided to applicants during examination: none

Learning Objective:
9696 – Define Technical Specification items found in the Definition Section

Question Source:
Modified Bank # DCPP S-52672

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 _____
55.43 43.2

Comments:

K/A: G2.1.22 - Ability to determine Mode of Operation

Table 1.1-1 (page 1 of 1)

MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	$350 > T_{AVG} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1 S-52672 Points: 1.00 Multiple Choice

Which one of the following events, at the end of refueling, would require a log entry to signify the transition from MODE 6 to MODE 5?

A. ALL reactor vessel head bolts fully tensioned

B. Steam Generator nozzle dams removed

C. Reactor vessel head in place with NO reactor vessel head bolts tensioned

D. First reactor vessel head bolt tensioned

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

9696	Define Technical Specification items found in the Definitions Section

Reference Id: S-52672
Must appear: No
Status: Active
User Text: 9624.13ALLN
User Number 1: 2.70
User Number 2: 3.90
Difficulty: 2.00
Time to complete: 2
Topic: Criteria for Mode 6 to 5 transition
Cross Reference: M-8 OBJ 6
Comment: Obj. 9696

SRO QUESTION *96*

Question Worksheet

Examination Outline Cross-Reference:

	Level	RO	SRO
	Tier #	______	3
	Group #	______	2
	K/A #	G2.2.27	
	Importance		3.5

Proposed Question:

Which of the following identifies the FIRST core alteration activity requiring the presence of the Refueling SRO in Containment?

A. Unlatching RCCAs

B. Lifting the upper internals

C. Moving the first fuel assembly

D. Lifting the reactor vessel head

Proposed Answer:

A. Unlatching RCCAs

Explanation:

A correct, per OP B-8DS2, the Refueling SRO is responsible for supervising Core Alterations. RCCA unlatching and moving of fuel assemblies are core alterations. Unlatching is performed prior to removal of the fuel assembly.
B incorrect, performed after unlatching RCCAs.
C incorrect, core alteration after unlatching.
D incorrect, prior to unlatching, but not a core alteration.

Technical Reference(s):
OP L-6, Refueling
OP B-8DS1, Core Unloading
OP B-8D attachment 9.1 Core Unloading Prerequisites Checklist

Proposed references to be provided to applicants during examination: none

Learning Objective:
6497 - State the responsibilities and duties of Refueling SRO

5827 – Explain which activities are considered core alterations and which are not.

Question Source: Bank #
Modified Bank # INPO 23199
New ______

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 _____
55.43 43.6

Comments:

K/A: G2.2.27 – Knowledge of the refueling process

		DATE/TIME/INITIALS
6.2.23	Notify maintenance and establish communications with maintenance personnel at Seal Table and check for leaks during flooding of Reactor Cavity.	____/____/____
6.2.24	If the RCS is to be drained after the Core is fully offloaded, begin voiding the S/G U-tubes per OP A-2:IV.	____/____/____
6.2.25	Obtain NSSS asset team leader concurrence that refueling cavity ready for flood up. ______________________________ NSSS ATL Signature	____/____/____
6.2.26	Flood the Refueling Cavity per OP B-2:II.	____/____/____
6.2.27	If PCV-135 was bypassed to increase flow for RCS cleanup, return to normal per OP B-1A:XVII, step 6.5. N/A [ ]	____/____/____
6.2.28	Remove tags and verify open RHR-1005 and RHR-933, RHR-8703 equalizing line isolation valves per Attachment 9.2.	____/____/____
6.2.29	Complete the OP L-0 Transition checklist for MODE 6 to Core Alterations.	____/____/____

NOTE: Unlatching RCCA drive shafts IS considered a Core Alteration.

6.2.30	When RCCA unlatching requirements are met per OP B-8D, authorize maintenance to uncouple the RCCA drive shafts.	____/____/____
6.2.31	Verify Refueling Cavity level is greater than 137'8" elevation.	____/____/____
6.2.32	Verify the Outage Safety Checklist requirements are satisfied for Mode 6 internals removal.	____/____/____
6.2.33	Authorize maintenance to remove the Upper Internals.	____/____/____
6.2.34	When the upper internals move is complete, the equipment hatch and personnel hatch doors may be opened and administrative controls established over containment penetrations for core alterations per TS 3.9.4.	____/____/____

ID: 23199 A refueling outage is in progress on Unit 1.

Which one of the following choices correctly identifies the FIRST activity requiring presence of the Refueling SRO in Containment?

Ans Lifting the upper internals

D1 De-tensioning the first reactor head stud

D2 Lifting the reactor vessel head.

D3 Moving the first fuel assembly.

AbbrevLocName	ExamDate	Vendor	Type	ExamType	Cog Level	ExamLevel	RefMaterial	ParentId
Salem Unit 1	11/4/2002	WEC	PWR	ILO	1	R		

QuestionComment: Explanation (Optional): A., B. - Head can be removed; D. Upper internals come out after "head is removed".

Distract1Comment: Explanation (Optional): A., B. - Head can be removed; D. Upper internals come out after "head is removed".

Distract2Comment: Explanation (Optional): A., B. - Head can be removed; D. Upper internals come out after "head is removed".

Distract3Comment: Explanation (Optional): A., B. - Head can be removed; D. Upper internals come out after "head is removed".

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
..G2.2.27			G2	2	27	

TITLE: Core Unloading

08/24/04
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure describes core unloading for Units 1 and 2.

2. DISCUSSION

2.1 This procedure provides step by step guidance for core unloading. Core loading and insert shuffling and are discussed in OP B-8DS2, PEP R-8DS2, OP B-8DS3 and PEP R-8DS3 respectively.

3. RESPONSIBILITIES

3.1 Shift foreman for operation of the plant and plant equipment.

3.2 The Refueling SRO is responsible for coordinating and supervising the following activities.

3.2.1 All fuel handling operations.

3.2.2 Core Alterations.

3.2.3 Safe and orderly evacuation of the refueling crew in the event of a high radiation alarm at a refueling station.

3.2.4 Determining the cause of the high radiation alarm.

3.2.5 Training the shift's upender operators, to the satisfaction of the SRO, that they can operate the upender station safely.

3.3 The Refueling SRO may delegate supervisory responsibilities at the other duty stations outside of containment to a designated operations representative.

3.4 Senior power production engineer (operations) for refueling procedures.

3.5 PPE (Nuclear) for technical guidance.

3.6 Control Room operator to assist with Control Room activities associated with fuel movements. This includes:

- In accordance with OP1.DC12, "Conduct of Routine Operations," assuring three-way communication of correct fuel assembly location prior to latching or unlatching;
- and providing a verification of fuel assembly locations when moving fuel.

DIABLO CANYON POWER PLANT
OP B-8D
ATTACHMENT 9.1

1 AND 2

TITLE: Core Unloading Prerequisites Checklist

ITEM NO.	DESCRIPTION	RESPONSIBLE DEPT INITIALS/DATE	REVIEWED (SFM) INITIALS/DATE
	C = Chemistry Manager; ANSS = NSSS Asset Team Leader		
	RP = Radiation Protection Manager; ACRE = Control Room/Electrical Asset Team Leader		
	RX Eng = Reactor Engineering; ASUP = Maintenance Support Asset Team Leader		
	Eng = Engineering; OP = Shift Manager/SFM		
	Prior to RCCA unlatching, verify Items 1 through 10 are current.		
1.	Temporary nozzle dams removed, or use approved by PSRC. N/A [ ]	OP ______/______	_____/______
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.	(FHB) ANSS ______/______	______/______
		(CONT) ANSS ______/______	______/______
3.	Verify STP M-42 was performed within 100 hours of Rx head removal. (ECG 42.3)	ENG ______/______	______/______
4.	After the Refueling Cavity has been filled, initiate a separate SFM Clearance and tag the following valves closed. CR NUMBER ____________________	OP ______/______	______/______
	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, <u>or</u> 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.	RP ______/______	______/______

OP B-8D (UNITS 1 AND 2)
ATTACHMENT 9.1

TITLE: Core Unloading Prerequisites Checklist

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 refueling cavity 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 Evacuation Alarm.	1 hour	OP ___/___	____ ____/____
10.	SRO for fuel handling operations only and licensed RO required for continuous monitoring of count rate data during CORE ALTERATIONS.		OP ___/___	____ ____/____

RCCA Unlatching Commenced ______/______ ______
TIME DATE SFM

RCCA Unlatching Completed ______/______ ______
TIME DATE SFM

TITLE: Core Unloading Prerequisites Checklist

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 computer graphical display or boards 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.		
	• Fuel rack absorber specimen tree located in proper storage rack cell.		
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 ______/______	______/______

TITLE: Core Unloading Prerequisites Checklist

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	ANSS ___/___	____ ____/____	6 mo.	____	____ ____/____
20.	LT 13-2, Spent Fuel Pool Temperature Channel TIC-651 Calibration.	31 days	ANSS ___/___	____ ____/____	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 refueling cavity water level >23 feet over the reactor vessel flange. (elev. 137 ft. 8 in.)	2 hours	OP ___/___	____ ____/____			
24.	ECG 42.2, Establish direct communications between the control room and the refueling stations.	1 hour	OP ___/___	____ ____/____	12 hours		See STP I-1A
	a. Test Evacuation Alarm.						C. O. Logs
25.	SRO for fuel handling operations only and licensed operator required for continuous monitoring of count rate data during CORE ALTERATIONS.	MODE 6	OP ___/___	____ ____/____	ALL TIME		

Core Unloading Commenced ______/______ ______
TIME DATE SFM

Core Unloading Completed ______/______ ______
TIME DATE SFM

OP B-8D (UNITS 1 AND 2)
ATTACHMENT 9.2

TITLE: Core Loading Prerequisites Checklist

ITEM NO.	DESCRIPTION	PRIOR TO LOADING. COMPLETE WITHIN	RESPONSIBLE DEPARTMENT INIT./DATE	REVIEWED (SFM) INIT. HOUR/DATE	RETEST FREQUENCY	NEXT DUE DATE	(SFM) RETEST COMPLETED INIT. HOUR/DATE
9.	High Flux at Shutdown						
a.	Alarm setpoint calculated for Mode 6 (STP R-28).	N/A	RX ENG ___/___	___ ____/____			
b.	Alarm setpoint current for Mode 6.	N/A	ACRE ___/___	____ ____/____			
c.	Hi flux at shutdown switch on NIS Panel in NORMAL.	N/A	OP ___/___	____ ____/____	7 days	_____	_____ ____/___

OP B-8D (UNITS 1 AND 2)
ATTACHMENT 9.2

TITLE: Core Loading Prerequisites Checklist

ITEM NO.	DESCRIPTION	RESPONSIBLE DEPT INITIALS/DATE	REVIEWED (SFM) INITIALS/DATE
10.	Fuel accountability computer graphical display or boards for control of fuel assembly and insert locations during loading operations are available for use in the appropriate areas (See TS6.ID2).	RX ENG ______/______	______/______
11.	Core loading sequence completed and issued by reactor engineering to ensure the core is neutronically coupled to the two source range detectors to be used.	RX ENG ______/______	______/______
12.	All critical personnel participating in the core loading have been adequately trained for their part in the fuel handling operations including nuclear engineering/operations/fuel handling.	RX ENG ______/______	______/______
		OP ______/______	______/______
13.	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.	RP ______/______	______/______
14.	Underwater TV cameras with video taping capability and adequate underwater lighting for post-loading verification:	RX ENG ______/______	______/______
	a. in operation on the reactor guide pin, OR b. available to be mounted onto the manipulator crane after reload.		
15.	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. Completing 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.		

TITLE: Core Loading Prerequisites Checklist

ITEM NO.	DESCRIPTION	PRIOR TO LOADING. COMPLETE WITHIN	RESPONSIBLE DEPARTMENT INIT./DATE	REVIEWED (SFM) INIT. HOUR/DATE	RETEST FREQUENCY	NEXT DUE DATE	(SFM) RETEST COMPLETED INIT. HOUR/DATE
16.	Verify an active SFM clearance exists on the source range speaker in containment.		OP ___/___	____ ____/____			
17.	STP I-4C, Calibration of Audio Count Rate/Scaler Timer Channel, current (should be performed in conjunction with STP I-4A).	N/A	ACRE ___/___	____ ____/____	18 months		
18.	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 ___/___	____ ____/____	ALL TIME		
19.	Verify all OP L-0 items for Core offload to Mode 6/Core Alterations are complete.	2 hours	OP ___/___	____ ____/____			
20.	ECG 42.2, Communications, establish direct communications between the Control Room and the refueling stations.	1 hour	OP ___/___	____ ____/____	12 hours		See STP I-1A
	a. Test Evacuation Alarm.						C. O. Logs
21.	SRO for refueling operations only and licensed operator required for continuous monitoring of count rate data during CORE ALTERATIONS.	MODE 6	OP ___/___	____ ____/____	ALL TIME		

Core Loading Commenced ______/______ ______
TIME DATE SFM

Core Loading Completed ______/______ ______
TIME DATE SFM

TITLE: Core Loading Prerequisites Checklist

ITEM NO.	DESCRIPTION	PRIOR TO LOADING. COMPLETE WITHIN	RESPONSIBLE DEPARTMENT INIT./DATE	REVIEWED (SFM) INIT. HOUR/DATE	RETEST FREQUENCY	NEXT DUE DATE	(SFM) RETEST COMPLETED INIT. HOUR/DATE

NOTE: Prior to RCCA Latching, verify Items 1, 2, 5, 8, 9, 13, and 21 are all current.

22.	ECG 42.5, Verify refueling cavity water level >23 feet above irradiated fuel assemblies within the reactor. (elev. 126 ft. 6 in).	2 hours	OP ___/___	____ ____/____	ALL TIME		See STP I-1B

RCCA Latching Commenced ______/______ ______
TIME DATE SFM

RCCA Latching Completed ______/______ ______
TIME DATE SFM

SRO QUESTION *97*

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, DCPP Emergency Exposure Guidelines

sro tier 3 group 3_97.doc

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: DCPP 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:

1. 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.

2. 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.

3. 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.)

4. 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.

ID: 20049 Given the following information for a rad worker qualified operator:
- Age 25 yrs.
- Total Lifetime exposure 3800 mrem TEDE
- Current Year exposure 800 mrem TEDE

A Site Area Emergency has been declared due to a LOCA Outside Containment with limited makeup to the RWST available. The above 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. What is the MAXIMUM exposure the operator may receive while performing this action?

Ans 25000 mrem TEDE.

D1 1200 mrem TEDE.

D2 4200 mrem TEDE.

D3 24200 mrem TEDE.

AbbrevLocName	ExamDate	Vendor	Type	ExamType	Cog Level	ExamLevel	RefMaterial	ParentId
Braidwood 1	10/20/2001	WEC	PWR	ILO		R		

QuestionComment	
Distract1Comment	
Distract2Comment	
Distract3Comment	

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
..2.3.1			2	3	1	

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	3
	K/A #	G2.3.8	
	Importance		3.2

<u>Proposed Question</u>:

A discharge of Gas Decay Tank 1-2 is planned.

The current conditions exist:

- Current time and date - 2200, 17 July
- RE-22 declared inoperable at 0100 on 3 July
- The planned discharge will take 4 hours
- 2 samples have been independently drawn and analyzed

Which of the following describes whether the planned discharge may or may not occur?

A. The planned discharge may not occur until RE-22 is restored to OPERABLE status.

B. The planned discharge may proceed in its entirety.

C. The discharge may occur, but only for 3 hours, then it must be terminated.

D. The discharge may not proceed because during the discharge the allowable time RE-22 may be inoperable will expire.

Proposed Answer:

A. The planned discharge may not occur until RE-22 is restored to OPERABLE status.

Explanation:

A correct, as of 0100 on 17 July, the 14 days allowed by ECG (and procedure OP G-2:V) has been exceeded. A discharge is not allowed.
B incorrect, 14 days exceeded.
C incorrect, 14 days already exceeded.
D incorrect, if there is time available, the discharge could have continued for a short period of time.

Technical Reference(s):
ECG 39.4
OP G-2:V, Gaseous Radwaste System – Gas Decay Tank Discharge

Proposed references to be provided to applicants during examination:
OP G-2:V
ECG 39.4

Learning Objective:
7428 - State gaseous radwaste system administrative controls
66068 - Discuss the requirements of System 39 ECGs.

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 ______
55.43 43.2

Comments:
K/A: G2.3.8 - Knowledge of the process for performing a planned gaseous radioactive release.

TITLE: Gaseous Radwaste System - Gas Decay Tank Discharge

07/30/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

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1. SCOPE

1.1 This procedure is intended to provide a method of safely discharging the contents of a Gas Decay Tank. The primary goal is to prevent an inadvertent or unmonitored release.

2. DISCUSSION

2.1 This procedure should be performed in conjunction with CAP A-6.

3. RESPONSIBILITIES

3.1 Chemistry Engineer/Foreman is responsible for preparation of the Authorization for Gas Decay Tank (GDT) discharge.

3.2 The Shift Foreman is responsible for the following:

3.2.1 Reviewing Discharge Authorization and verifying compliance with ECG's 24.3 and 39.4.

3.2.2 Authorizing and issuing the Discharge Authorization.

3.2.3 Reviewing completed Discharge Authorization and forwarding it to Chemistry Section.

3.3 The Auxiliary Building Sr and/or Auxiliary Building NO are responsible for completing the Discharge Authorization and returning it to the Shift Foreman.

4. PREREQUISITES

4.1 The Gaseous Radwaste System is aligned per OP G-2:I, Gaseous Radwaste System - Make Available.

4.2 The Auxiliary Building Ventilation Exhaust Treatment System shall be operable at all times. Refer to Tech Spec 3.7.12.

4.3 An Authorization for Gas Decay Tank Discharge, (Form 69-9351), has been prepared and signed by Chemistry Section Engineer/Foreman.

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5. PRECAUTIONS AND LIMITATIONS

5.1 Do not commence a discharge until the Discharge Authorization is approved and issued by Chemistry.

5.2 RE-22 should be operable for all GDT discharges. If RE-22 is inoperable, discharges may occur so long as additional precautions are taken as required by ECG 39.4. These precautions are detailed in this procedure.

5.3 Ventilation dilution flow should be maintained throughout the discharge.

5.4 Only one GDT can be discharged (vented) at a time.

5.5 Only a tank NOT selected for FILL or STANDBY can be discharged (vented).

5.6 A GDT discharge should NOT be performed if the actual tank pressure is NOT in agreement with the tank pressure indicated on the Discharge Authorization, (Form 69-9351).

5.7 Valve alignment verification is performed and documented on the Discharge Authorization, (Form 69-9351).

6. INSTRUCTIONS

6.1 Review the Prerequisites and Precautions and Limitations sections of this procedure.

6.2 Verify RE-22 operable for monitoring GDT release.

NOTE: If the Discharge Authorization requires HASP to be readjusted prior to the discharge, contact the Shift MS Tech and request the setpoint be changed to the value prescribed by the Chemistry Engineer/Foreman.

6.2.1 Check RE-22 Operable by doing the following:

a. Check that the instrument calibration has not expired.

b. Perform a CHANNEL CHECK as follows:

1. Power light is ON.
2. Operation Selector set to OPERATE.
3. Range Selector set to WIDE.
4. Normal meter reading with slight movement of needle.
5. Red Low Alarm lamp is OFF.

c. Perform a source check on RE-22.

1. Place the operation selector switch to CHECK SOURCE position.
2. Read the SOURCE count rate.
3. Place the operation selector switch to RESET and then to OPERATE.

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d. Log the SOURCE count rate on Part 3 of the Discharge Authorization.

e. If RE-22 is operable, proceed to Step 6.3.

6.2.2 If RE-22 is inoperable, proceed as follows:

****
CAUTION: Gaseous Radwaste discharges may continue for up to 14 days after RE-22 is declared inoperable if the applicable ACTION statements in ECG 39.4 are followed.
****

a. Note the date and time that RE-22 is declared inoperable in the Shift Log.

b. Review the applicable portions of ECG 39.4.

c. Attach a CAUTION Tag to the control switch for the key operated discharge valve, 1-FCV-410, made out to the Shift Foreman. Indicate on the tag, the date and time of the expiration of the 14 day period when termination of all releases, via this pathway, is required.

d. Request Chemistry Section to obtain and analyze a second independent GDT sample.

e. Initiate an Action Request (AR) to initiate repair of RE-22.

f. Fill out the appropriate sections of the Discharge Authorization.

6.2.3 If RE-22 has been inoperable for less than or equal to 14 days:

****
CAUTION: Gaseous Radwaste discharges may continue for up to 14 days after RE-22 is declared inoperable if the ACTION statements in ECG 39.4 are followed.
****

a. Review the applicable portions of ECG 39.4.

b. Request Chemistry Section to obtain and analyze a second independent GDT sample.

c. Ensure that the release rate calculations are verified by at least two qualified staff members.

d. If the tank activity is below the limits stated in CAP A-6, fill out the appropriate sections of the Discharge Authorization.

6.2.4 If RE-22 has been inoperable for greater than 14 days:

****
CAUTION: Gaseous Radwaste discharges may NOT continue when RE-22 has been declared inoperable for greater than 14 days.
****

a. Review the applicable portions of ECG 39.4.

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b. Terminate the Gaseous Radwaste discharge process.

c. Initiate an Admin Tagout to the Shift Foreman to attach a CAUTION Tag to the control switch for the key operated discharge valve, 1-FCV-410.

d. Notify SFM that RE-22 inoperability must be resolved.

6.3 The Shift Foreman should report off the Admin. Tagout used for placing the GDT in holdup.

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CAUTION: If Items 6.4 or 6.5 are not met, discontinue this procedure and notify the Shift Foreman.

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6.4 At the Auxiliary Control Board, verify actual Gas Decay Tank pressure is consistent with the pressure logged in Part 1 of the Discharge Authorization. Log the prerelease pressure in Part 3 of the Discharge Authorization.

6.5 Verify the following valves CLOSED and Caution tagged as required by OP G-2:11, when originally placing the tank in holdup, then remove Admin Tagout.

6.5.1 GDT Fill Valve

6.5.2 GDT Purge Valve

6.5.3 GDT N_2 Supply Valve

6.5.4 GDT Gas Analyzer Sample has been removed from scan and CAUTION Tag is hung on panel.

NOTE: If GDT 1-3 was on holdup, perform this on the U-2 Gas Analyzer also.

6.5.5 GW-0-FCV-417, IF GDT 1-3 is in holdup.

6.6 Just prior to the start of discharge, notify the Control Operator that the GDT discharge is about to begin.

6.7 Notify the Chemistry Section that the discharge will commence. Log the date and time on Part 3 of the Discharge Authorization.

6.8 Select the GDT to be discharged on the tank vent selector switch and OPEN the appropriate tank vent valve listed below:

6.8.1 GDT 1-1, 1-FCV-404

6.8.2 GDT 1-2, 1-FCV-405

6.8.3 GDT 1-3, 1-FCV-406

6.9 Complete independent verification of valve alignment and record valve numbers used on Part 3 of Discharge Authorization.

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6.10 Obtain authorization from the Shift Foreman to discharge the GDT following his review of the discharge authorization. Both Shift Foreman and Chemistry Engineer/Foreman must have signed the Discharge Authorization to make the authorization valid.

6.11 The Operator who will perform the discharge shall obtain the key for the Gaseous Radwaste discharge valve, 1-FCV-410, and receive any pertinent instructions concerning the discharge from the Shift Foreman.

6.12 Start the discharge by opening the key operated discharge valve, 1-FCV-410. Log time and Batch No. in NO log and document performance of line up and date and time on Part 3 of Discharge Authorization.

6.13 Record the release information required on the Discharge Authorization during and following the release.

6.14 If RE-22 alarms due to an actual high rad condition during the discharge:

6.14.1 Ensure 1-RCV-17 has closed.

6.14.2 Close 1-FCV-410 and remove the key.

6.14.3 Notify the Shift Foreman.

6.14.4 Notify the Chemistry Section.

6.14.5 Complete the Discharge Authorization Form and note in the comments section the reason for discharge termination.

6.15 If RE-22 fails during the discharge due to a component problem, proceed as follows:

6.15.1 Ensure 1-RCV-17 has closed.

6.15.2 Close 1-FCV-410 and remove the key.

6.15.3 Notify the Shift Foreman.

6.15.4 Notify the Chemistry Section.

6.15.5 If RE-22 is repaired, verify that the GDT has not been selected to "FILL" or "PURGE" following the termination of the discharge. To continue the tank discharge, consult the Shift Foreman.

6.16 When the GDT pressure decreases to approx. 5 psig, terminate the discharge by closing the tank vent valve. Log final GDT pressure on Part 3 of Discharge Authorization.

NOTE: If the GDT is being discharged in preparation for clearing, (OP G-2:III), or in conjunction with N_2 purging, (OP G-2:IV), terminate the discharge when the GDT pressure decreases to approx. 0.5 psig.

6.17 Close 1-FCV-410 and remove key. Record date and time of release termination. Complete all portions (Part 3) on the Discharge Authorization. Sign and return the Discharge Authorization and the key for 1-FCV-410 to the Shift Foreman.

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7. REFERENCES

7.1 Technical Specifications 3.7.12, ECG's 24.2 and 24.3.

7.2 Equipment Control Guidelines, ECG 39.4.

7.3 CAP A-6, "Gaseous Radwaste Discharge Management."

8. RECORDS

The authorization for GDT discharge must be routed to the Chemistry Section for update and retention.

9. ATTACHMENTS

None

39.0 INSTRUMENTATION

39.4 Radioactive Gaseous Effluent Monitoring Instrumentation

ECG 39.4 The radioactive gaseous effluent monitoring instrumentation channels for each function shown in Table 39.4-1 shall be OPERABLE with their alarm/trip setpoints set to ensure the limits of the Radioactive Effluent Controls Program (CY2.ID1) are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculations (CAP A-8).

APPLICABILITY: In accordance with Table 39.4-1.

ACTIONS

<table>
<tr><th>CONDITION</th><th>REQUIRED ACTION</th><th>COMPLETION TIME</th></tr>
<tr><td>A. One or more functions listed in Table 39.4-1 with the required channel inoperable.</td><td>A.1 Enter the Condition referenced in Table 39.4-1 for the channel.</td><td>Immediately</td></tr>
<tr><td>B. Required Gaseous Radwaste System Noble Gas Activity Monitor channel inoperable.</td><td>---------------------NOTE (B)--------------------
The contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:
a. At least two independent samples of the tank’s contents are analyzed, and
b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve line up.
Otherwise, comply with Action B.1
---</td><td></td></tr>
<tr><td></td><td>B.1 Suspend release of radioactive effluents via this pathway.</td><td>Immediately</td></tr>
<tr><td></td><td>AND</td><td></td></tr>
<tr><td></td><td>B.2.1 Restore the Monitor to OPERABLE status.</td><td>14 days</td></tr>
<tr><td></td><td>OR</td><td></td></tr>
<tr><td></td><td>B.2.2 Explain in Effluent Release Report pursuant to TS 5.6.3 why this inoperability was not corrected within the time specified.</td><td>Next submittal of the Effluent Release Report</td></tr>
</table>

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C. Two Plant Vent System Noble Gas Activity Monitor channels inoperable.	-------------------NOTE (C)--------------------- Effluent release via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours. ---	
	C.1.1 Implement a sampling program as described in NOTE C.	12 hours
	<u>OR</u>	
	C.1.2 Suspend release of radioactive effluents via the associated pathway.	Immediately
	<u>AND</u>	
	C.2.1 Restore one Monitor channel to OPERABLE status.	30 days
	<u>OR</u>	
	C.2.2 Explain in Effluent Release Report pursuant to TS 5.6.3 why this inoperability was not corrected within the time specified.	Next submittal of the Effluent Release Report
D. Two Plant Vent System Iodine Sampler channels inoperable (cartridge and filter only) <u>OR</u> Two Plant Vent System Particulate Sampler channels inoperable.	-------------------NOTE (D)--------------------- Effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment. ---	
	D.1.1 Implement a sampling program as described in Note D.	2 hours
	<u>OR</u>	
	D.1.2 Suspend release of radioactive effluents via this pathway.	Immediately
	<u>AND</u>	
	D.2.1 Restore one Sampler channel to OPERABLE status.	30 days
	<u>OR</u>	
	D.2.2 Explain in Effluent Release Report pursuant to TS 5.6.3 why this inoperability was not corrected within the time specified.	Next submittal of the Effluent Release Report

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E. Two Plant Vent System Flow Rate Monitor channels inoperable OR Two Plant Vent System Iodine Sampler Flow Rate Monitor channels inoperable.	-------------------NOTE (E)---------------------- Effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours. --- E.1.1 Estimate the flow rate as described in Note E.	4 hours
	OR E.1.2 Suspend release of radioactive effluents via this pathway.	Immediately
	AND E.2.1 Restore one Monitor channel to OPERABLE Status.	30 days
	OR E.2.2 Explain in Effluent Release Report pursuant to TS 5.6.3 why this inoperability was not corrected within the time specified.	Next submittal of the Effluent Release Report

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 39.4.1	Perform CHANNEL CHECK.	Prior to each release.
SR 39.4.2	Perform CHANNEL CHECK.	24 hours
SR 39.4.3	-----------------------------------NOTE----------------------------------- The CHANNEL CHECK shall consist of verifying that the iodine cartridge and particulate filter are installed in the sample holders. --- Perform CHANNEL CHECK.	7 days
SR 39.4.4	Perform SOURCE CHECK.	Prior to each release.
SR 39.4.5	Perform SOURCE CHECK.	31 days
SR 39.4.6	----------------------------------NOTE------------------------------------- The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. --- Perform CHANNEL CALIBRATION.	18 months
SR 39.4.7	Perform CHANNEL CALIBRATION.	18 months
SR 39.4.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 39.4.9	----------------------------------NOTE------------------------------------- The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist: a. Instrument indicates measured levels above the Alarm Setpoint, or b. Circuit failure, or c. Instrument indicates a downscale failure, or d. Instrument controls not set in operate mode. --- Perform CHANNEL FUNCTIONAL TEST.	92 days (continued)

SURVEILLANCE REQUIREMENTS

(continued)	SURVEILLANCE	FREQUENCY
SR 39.4.10	-----------------------------------NOTE-------------------------------------- The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist: a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or b. Relay control circuit failure (isolation only), or c. Instrument indicates a downscale failure (alarm only), or d. Instrument controls not set in operate mode (alarm only). --- Perform CHANNEL FUNCTIONAL TEST.	92 days

Table 39.4-1
Radiological Gaseous Effluent Monitoring Instrumentation

FUNCTION	REQUIRED NUMBER OF CHANNELS	REQUIRED MODE	ECG 39.4 ACTION CONDITION	SURVEILLANCE REQUIREMENTS
1. Gaseous Radwaste System Noble Gas Activity Monitor – Providing alarm and automatic termination of release	1	At all times	B	SR 39.4.1 SR 39.4.4 SR 39.4.6 SR 39.4.10
2. Plant Vent System				
a. Noble Gas Activity Monitor	1 per Unit	At all times	C	SR 39.4.2 SR 39.4.5 SR 39.4.6 SR 39.4.9
b. Iodine Sampler (the cartridge and filter only)	1	At all times	D	SR 39.4.3
c. Particulate Sampler (the cartridge and filter only)	1	At all times	D	SR 39.4.3
d. Plant Vent Flow Rate Monitor	1	At all times	E	SR 39.4.2 SR 39.4.7 SR 39.4.8
e. Iodine Sampler Flow Rate Monitor	1	At all times	E	SR 39.4.2 SR 39.4.7 SR 39.4.8
3. Containment Purge System (In Accordance With TS) Noble Gas Activity Monitor	TS 3.3.6	TS 3.3.6	TS 3.3.6	

BASES

BACKGROUND

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the release of radioactive materials in gaseous effluents during actual or postulated radiological releases. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM-TS 5.5.1) to ensure that the alarm/trip will occur prior to exceeding 10 CFR Part 20 limits.

The gaseous radwaste system gas decay tank noble gas discharge monitor, RM-22 (channel R-22), detects radioactive noble gases and provides an alarm and automatic termination of release. When actuated, the instrument channel provides a control output signal initiating the closure of valve RCV-17 (Reference 6). When activity level is sufficiently low, these gases are released from the gas decay tanks, passed through a HEPA filter, and vented directly to the atmosphere via the plant vent system (Reference 4). The gas decay tanks are common to both units.

The plant vent system includes additional monitoring instrumentation in the release vents. Noble gas monitors RM-14 and RM-14R (channels R-14 and R-14R) provide an alarm when high radiation is detected in the plant vent. The plant vent iodine monitors, RM-24 and RM-24R (channels R-24 and R-24R), sample air exhausted through the plant vent to determine the radioactive iodine concentration. Similarly, the plant vent radioactive particulate monitors, RM-28 and RM-28R (channels R-28 and R-28R), sample the exhaust air for radioactive particulate concentrations. In conjunction, flow rate monitoring channels FT-813 and FT-814, (associated with iodine sampler channels R-24 and R-24R), measure the flow rates, which are used as inputs in determining the iodine activity. Loop flow transmitter FT-12 or its redundant counterpart, FT-12R, feeds a signal to flow recorder FR-12, which monitors the gaseous effluent flow rate in the plant vent. (Reference 7). The containment exhaust radiation monitors, RM-44A and RM-44B (channels R-44A and R-44B), monitor the air exhausted from containment through the containment purge and exhaust lines, and are covered by TS 3.3.6, “Containment Ventilation Isolation Instrumentation.”

The operability and use of this instrumentation is consistent with the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of the ODCM (commitment for operation 6.1.7.1 of CY2.ID1) shall be such that concentrations as low as 1×10^{-5} microcurie/mL are measurable.

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APPLICABLE SAFETY ANALYSES	Several plant features are in place to limit the amount of activity that could be released in the event of a design basis accident (DBA) gas decay tank rupture. Limits on primary coolant activity restrict the total curies present in the gas decay tanks as do the physical dimensions of each individual tank. Radiation monitors allow for the early detection of release and isolation valves allow operators to terminate the release upon detection. The plant vent system is equipped with two channels of radioactive noble gas monitors, iodine monitors, and particulate monitors. Failure of any one of the two channels does not render the monitoring instrumentation inoperable. Sampling programs can be implemented to compensate for a failure of the channel required for operability.
LCO	The radioactive gaseous effluent monitoring instrumentation must be OPERABLE to ensure that radioactive gas from the decay tanks is not released to the atmosphere via the plant vent system until its activity is sufficiently low. The LCO requires that the single gaseous radwaste system noble gas activity monitor channel (R-22) be OPERABLE at all times as a primary means of detection and transmittal of an automatic isolation signal in the event of a gas decay tank rupture. The radiation monitoring channels in the plant vent system release vent detect the concentration of radioactive noble gases, iodine, and particulates. The plant vent system radiation monitoring channels and the plant vent flow rate monitors associated with each function have two channels, but only one is required for operability.
APPLICABILITY	ECG 39.4 is applicable at all times because the effects of an uncontrolled release of a gas decay tank's contents are not dependent on plant operational mode.
ACTIONS	<u>A.1</u> With one or more functions (as listed in Table 39.4-1) with the required channel inoperable, the appropriate actions must be taken to restore operability to the instrumentation in order to ensure that a radioactive gaseous effluent release is not vented to the atmosphere in the case of a gas decay tank rupture. When one or more function is inoperable, the release of gaseous effluents is suspended or a compensatory action is taken to ensure the safe release of gaseous effluent until the equipment is made OPERABLE. <u>B.1</u> Suspending release of radioactive effluents via this pathway ensures that no potentially radioactive gases are released from the decay tanks to the plant vent while the monitoring function is inoperable. As described in Required Action Note B, the contents of the tank may be released for up to 14 days provided that the compensatory analysis and verification are performed.

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ACTIONS

B.2.1 and B.2.2

During the 14-day allotment the channel shall be repaired so that the activity in the gaseous effluent can be determined and the automatic valve closure function is available if the activity exceeds the alarm/trip setpoint. If the channel is not repaired in the allotted time, an explanation is required in the next submittal of the Effluent Release Report. The 14 day completion time is reasonable based on the low probability of uncontrolled release and provides ample time to repair the channel.

C.1.1

The purpose of the sampling program is to manually sample and analyze the gaseous effluent for radiation when the release vent radiation monitoring channels are inoperable. The gaseous radwaste system noble gas activity monitoring channel (R-22) ensures gaseous effluent release to the plant vent system via the gas decay tanks has sufficiently low activity. In addition to the gas decay tanks, several other release pathways are routed to the plant vent system, making it necessary to sample at this frequency.

C.1.2

Suspending release of radioactive effluents via this pathway ensures that no potentially radioactive gases are vented to the atmosphere while the radioactive noble gas monitoring channel in the release vent is inoperable and initiating a sampling program is not feasible.

C.2.1 and C.2.2

After suspending release of effluents via this pathway or initiating the sampling program, 30 days are allotted to repair the channel. The completion time is based on the time necessary to restore operability to the channel and the low probability of radioactive release concurrent with channel inoperability. If the channel is not repaired in the allotted time, an explanation is required in the next submittal of the Effluent Release Report.

D.1.1

To respond to the low flow alarm, determine that a simple fix cannot be made and that an auxiliary sampler is needed, move the sampler in, and hook up and verify operation. A maximum of two hours is considered a reasonable time to initiate the continuous sampling program Over two hours should be considered as exceeding the time limitation of this ECG.

D.1.2

Suspending release of radioactive effluents via this pathway ensures that no potentially radioactive gases are vented to the atmosphere while either the iodine sampler or particulate sampler is inoperable and continuous sampling is not possible.

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ACTIONS	D.2.1 and D.2.2 After suspending release of effluents via this pathway or initiating the sampling program, 30 days are allotted to repair the channel, based on the time demonstrated necessary by experience to restore operability. If the channel is not repaired in the allotted time, an explanation is required in the next submittal of the Effluent Release Report. E.1.1 Effluent release via the plant vent may continue for up to 30 days provided the flow rate is estimated at least once every four hours in order to ensure detection of significant releases. E.1.2 Suspending release of radioactive effluents via this pathway is a conservative measure that ensures no potentially radioactive gases are vented to the atmosphere while the flow rate monitors are inoperable and the flow rate cannot be estimated per the frequency specified in Required Action E.1. E.2.1 and E.2.2 After suspending effluent release, 30 days are allotted to repair the channel If the channel is not repaired in the allotted time, an explanation is required in the next submittal of the Effluent Release Report.
SURVEILLANCE REQUIREMENTS	SR 39.4.1 Performing a CHANNEL CHECK on the gas decay tank noble gas activity monitor (RM-22) ensures normal behavior of the channel prior to each release. By confirming operability, a CHANNEL CHECK also ensures that the instrumentation is able to send a high radiation signal to the isolation valve to terminate gaseous effluent release. SR 39.4.2 A CHANNEL CHECK is performed on the noble gas plant vent monitors (RM-14 and RM-14R), the iodine sampler flow rate monitors, and the plant vent flow rate monitor (flow recorder, FR-12) once every 24 hours. This surveillance ensures reliability of concentration measurements in the case of a gas decay tank rupture. The frequency is based on the low probability of an uncontrolled release. SR 39.4.3 A CHANNEL CHECK is performed every 7 days on the iodine sampler and particulate sampler; the monitors associated with these samplers are RM-24 and RM-24R, and RM-28 and RM-28R, respectively. The CHANNEL CHECK consists of verifying that the iodine cartridge and particulate filter are installed in the sample holders.

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SURVEILLANCE REQUIREMENTS	<u>SR 39.4.4</u> A SOURCE CHECK is the qualitative assessment of channel response when the channel sensor is exposed to a radiological source of known activity. A SOURCE CHECK is performed on the gas decay tank noble gas activity monitor prior to each release to ensure that the radiation monitor (RM-22) is capable of accurate detection and measurement. The surveillance frequency ensures that the radiation monitor is functioning prior to a release and is based on the low probability of an uncontrolled release. <u>SR 39.4.5</u> A SOURCE CHECK is performed on the plant vent system noble gas activity monitors (RM-14 and RM-14R) every 31 days as a routine check of radiation monitor functionality. The surveillance frequency is based on the low probability of an uncontrolled release. <u>SR 39.4.6 and SR 39.4.7</u> A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. CHANNEL CALIBRATION is performed on the gas decay tank noble gas activity monitor (RM-22), the plant vent system noble gas activity monitors (RM-14 and RM-14R), the plant vent system flow rate monitor (FR-12) and the iodine sampler flow transmitters (FT-813 and FT-814). These surveillances are performed every 18 months, which has been shown to be acceptable by experience. <u>SR 39.4.8, SR 39.4.9, and SR 39.4.10</u> A CHANNEL FUNCTIONAL TEST is performed on the plant vent flow recorder and its associated flow transmitters (FR-12 fed from FT-12 and FT-12R), the iodine sampler flow transmitters (FT-813 and FT-814), the plant vent system noble gas activity monitors (RM-14 and RM-14R), and the gas decay tank noble gas activity monitor (RM-22) every 92 days to ensure channel OPERABILITY. The frequency is based on operator experience and the low probability of channel inoperability concurrent with a DBA.
REFERENCES	CY2.ID1, “Radioactive Effluent Controls Program.” CAP A-8, “Offsite Dose Calculations.” DCPP Technical Specifications, Sections 3.3.6, 5.4.1, 5.5.1, and 5.6.2. FSAR, Chapter 11, “Radioactive Waste Management.” FSAR, Chapter 15, “Accident Analyses.” DCM S-39, “Radiation Monitoring System.” DCM T-24, “Design Criteria for DCPP Instrumentation and Controls.”

12/18/2002

Effective Date

SRO QUESTION *99*

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	______	3
	Group #	______	4
	K/A #	G2.4.41	
	Importance		4.1

Proposed Question:

A radiological emergency event is in progress. The release could result in exceeding EPA PAGs near the site boundary.

Which of the following Emergency Event Classifications would be appropriate?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer:

C. Site Area Emergency

Explanation:

C correct, definition of SAE:
Events which are in progress or have occurred involving actual or likely major failures of plant functions needed for protection of the public.

- Reflects conditions where there is a clear potential for radioactive release.
- Radioactive releases or their potential may exceed EPA PAGs, but only near the site boundary

Technical Reference(s): LEP-2, Emergency Plan Procedures

Proposed references to be provided to applicants during examination: none

Learning Objective: 8535 - State the definition of the four emergency event classifications

Question Source: Bank # E-36020

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 _____
55.43 43.4

Comments:
K/A: G2.4.41 - Knowledge of the emergency action level thresholds and classifications.

EP G-1, Emer. Classification and E-Plan Activation, Continued

15 Minute Time Frame
Obj 7

- The time frame for initial notification starts after the SM/ISEC:
  - analyzes the event or conditions, and
  - determines that event classification criteria are met.
- This gives the SM/ISEC a reasonable period of time (15 minutes per NRC guidance and management expectations) to assess and classify an emergency condition once indications are available to the Control Room operators that an EAL has been exceeded.
- Note that in the previous discussions there are two-15 minute times being discussed:
  - Goal: to have classification made within 15 minutes of event initiation.
  - Requirement: to have notifications performed within 15 minutes of classification.
  - Therefore, the time between event initiation and notification completion can be up to 30 minutes[TRP5].

Follow-up Notifications
Obj 5, 6

Follow-up notifications to the county and state should be made approximately every 45 minutes, even if the event classification is unchanged.

- Keeps everyone up-to-date of changing plant conditions
- Notification still required within 15 minutes if event classification changes

Emergency Classifications

DCPP (and the commercial nuclear industry) uses four standard emergency classifications for categorizing events. They are (least to most severe):

- Notification of Unusual Event (NUE) or Unusual Event
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

Unusual Event (NUE)
Obj 8

- Off-normal conditions are in progress OR have occurred which:
  - Indicate a potential *degradation* of level of plant safety *if proper action is not taken*.

    OR
  - If circumstances beyond the control of operating staff, making the situation more serious from a safety standpoint.
- No releases of radioactive material requiring off-site response or monitoring are expected.

Continued on next page

EP G-1, Emer. Classification and E-Plan Activation, Continued

Alert
Obj 8

Events in progress or having occurred, involving an actual or potentially *substantial degradation* of the plant safety level.

- Any radioactive releases that may occur (greater than Technical Specification limits) are expected to be limited to a *small fraction* of EPA Protective Action Guides (PAGs) at the site boundary.
- The lowest classification level where off-site emergency response is anticipated.
- It is expected that for most Alerts, the plant will be placed in a safe condition and releases, if any, will be minimal.

Site Area Emergency (SAE)
Obj 8

Events which are in progress or have occurred involving actual or likely *major failures* of plant functions needed for protection of the public.

- Reflects conditions where there is a clear potential for radioactive release.
- Radioactive releases or their potential may exceed EPA PAGs, but *only near the site boundary*.
- Because the possible release is significant, care must be taken in alerting offsite authorities to distinguish whether the release is merely potential, likely, or actually occurring.
- A core meltdown situation is NOT indicated based on current information.

General Emergency (GE)
Obj 8

Events are in progress OR have occurred which indicate actual or imminent *substantial core damage* with potential for containment loss.

- Radioactive releases can be expected to exceed EPA PAG off-site exposure levels.

Command and Control Hierarchy

Command and control of an emergency begins with the ISEC and, if required, is transferred to the SEC and then to the Recovery Manager (RM).

Only the general responsibilities of these three positions, as they pertain to EP G-1, are listed below.

- Specific duties and responsibilities are discussed later.

Continued on next page

1 E36020 Points: 1.00 Multiple Choice

Which one of the following emergency event classification levels assumes that dose rates at the Site Boundary due to a radioactive release are well below Emergency Action Plan (EAP) Protection Action Guidelines (PAGs)?

A. Notification of Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

ASSOCIATED INFORMATION:

Associated objective(s):

8535	State the definition of the four emergency event classifications

Reference Id: E36020
Must appear: No
Status: Active
User Text:
User Number 1: 0.00
User Number 2: 0.00
Difficulty: 2.00
Time to complete: 2
Topic: LEP2 - Definition of events
Cross Reference: ECP Obj. 7.1
Comment: From O/E database DLH4 6/5/00

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	3
	Group #	_____	4
	K/A #	G2.4.45	
	Importance		3.6

<u>Proposed Question</u>:

A Unit 1 shutdown from full power is in progress at 1%/min.

The following events occur:

- PK15-22 alarms on Unit 2 due to Alarm Input 1531- Main Annun Unit 1 Train A and B Failure.
- On Unit 1 ALL annunciator windows are blank, and the alarm typewriter and CRT have NOT responded for 18 minutes.
- The PPC and SPDS are not responding

What Classification or notification is required to be made?

A. 1 hour Non-emergency report.

B. Unusual Event.

C. Alert.

D. Site Area Emergency.

Proposed Answer:

B. Unusual Event.

Explanation:
A incorrect, threshold for emergency classification exceeded.
B correct, UE 16.
C incorrect, no major transient in progress and PPC and SPDS not responding.
D incorrect, no major transient in progress.

Technical Reference(s):
EP G-1 Attachment 7.1

Proposed references to be provided to applicants during examination:
EP G-1 Attachment 7.1

Learning Objective: 5464 – Explain the classification of emergency conditions

Question Source:

Modified Bank # DCPP B-0621

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 _____
55.43 43.5

Comments:

K/A: G2.4.45 – Ability to prioritize and interpret the significance of each annunciator or alarm.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
VII. LOSS OF POWER OR ALARMS OR ASSESSMENT OR COMMUNICATIONS	12. Loss of <u>all</u> off-site power for greater than 15 minutes <u>AND</u> at least 2 D/Gs are supplying their vital busses (Modes 1-4).	13. Loss of <u>all</u> off-site power for greater than 15 minutes <u>AND</u> only 1 D/G is supplying its vital bus (Modes 1-4).	6. Loss of all on-site <u>AND</u> off-site AC power for > 15 minutes (Modes 1-4).	See General Emergency Condition #5 under LOSS OF ENGINEERED SAFETY FEATURE.
	13. Loss of <u>all</u> off-site power for greater than 15 minutes <u>AND</u> at least 1 D/G is supplying its vital bus (Modes 5 and 6).	14. Loss of <u>all</u> off-site and on-site AC power for greater than 15 minutes in Modes 5 or 6.		
	14. Loss of all vital DC power as indicated by DC Bus 11(21), 12(22), and 13(23) undervoltage for A 15 minutes (Modes 5-and 6)	15. Loss of all vital DC power as indicated by DC Bus 11(21), 12 (22) and 13 (23) undervoltage for < 15 minutes (Modes 1-4).	7. Loss of all vital DC power as indicated by DC Bus 11 (21), 12 (22) and 13 (23) undervoltage for > 15 minutes (Modes 1-4).	
	15. Loss of assessment capabilities as indicated by a total loss of SPDS in the Control Room <u>AND</u> simultaneous loss of all displays for any "Accident Monitoring" variable in Tech Spec Table 3.3.3-1 for > 1 hour while in Modes 1, 2 or 3.			
	16. Main Control Room Annunciators PKs 1 through 5 <u>AND</u> display capabilities <u>AND</u> the seismically qualified annunciator display all do not respond to an alarm condition in Modes 1-4 for over 15 minutes.	16. Main Control Room Annunciators PKs 1 through 5 <u>AND</u> display capabilities <u>AND</u> the seismically qualified annunciator display all do not respond to an alarm condition in MODES 1-4 for over 15 minutes <u>AND</u> the plant is in a significant transient (plant trip, SI, or generator runback A25 Mw/min), nonannunciating systems available.	8. Main Control Room Annunciators PKs 1 through 5 <u>AND</u> display capabilities <u>AND</u> the seismically qualified annunciator display all do not respond to an alarm condition in MODES 1-4 for over 15 minutes <u>AND</u> the plant is in a significant transient <u>AND</u> backup, nonannunciating systems are not available (PPC, SPDS).	

<u>NOTE</u>: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.