RO Question 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	EPE 007 EK2.03	
	Importance Rating:	3.5	3.6

Proposed Question:

Unit 1 is at 6% power.

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

A. A single Pressurizer Pressure low bistable trips.

B. Two Pressurizer Pressure low bistables trip.

C. One Pressurizer Pressure channel fails high.

D. Two Pressurizer Pressure channels fail high.

Proposed Answer:

D. Two Pressurizer Pressure channels fail high.

Explanation:

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

B incorrect, trip is blocked below 10%

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

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

Technical Reference(s): PK04-12

Proposed references to be provided to applicants during examination: None

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

- Annunciator Windows
- Status Lights
- Monitor Light Boxes

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

REACTOR TRIP INITIATE

08/31/05
Effective Date

QUALITY RELATED

CONTAINS REACTOR TRIP CRITERIA

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
16	Pzr Lo Press and P7 2/4 React Trip	K0202	< 1950 psig	2.1
17	Pzr Hi Press 2/4 React Trip	K0203	> 2385 psig	2.1
18	Pzr Hi Lvl and P7 2/3 React Trip	K0204	> 90% of span	2.1
24	Stm Gen 1-1 Lo-Lo Lvl 2/3 React Trip	K0210	< 15%	2.1
25	Stm Gen 1-2 Lo-Lo Lvl 2/3 React Trip	K0211	< 15%	2.1
26	Stm Gen 1-3 Lo-Lo Lvl 2/3 React Trip	K0212	< 15%	2.1
27	Stm Gen 1-4 Lo-Lo Lvl 2/3 React Trip	K0213	< 15%	2.1
29	RCP Busses UF and P7 1/2 React Trip	K0301	< 54 hz	2.1
490	Manual React Trip	1RT/CS 2RT/CS		2.1

2. OPERATOR ACTIONS

2.1 General Actions **(All Inputs)**

2.1.1 IF plant conditions require a Reactor trip
AND an automatic Reactor trip has NOT occurred,
THEN manually TRIP the Reactor. [ ]

2.1.2 IF at any time a reactor trip or safety injection occurs
OR has been initiated,
THEN GO TO EOP E-0, "Reactor Trip or Safety Injection". [ ]

2.1.3 Probable Causes

- The required number of reactor trip bistables are activated

3. AUTOMATIC ACTIONS

3.1 Reactor trip

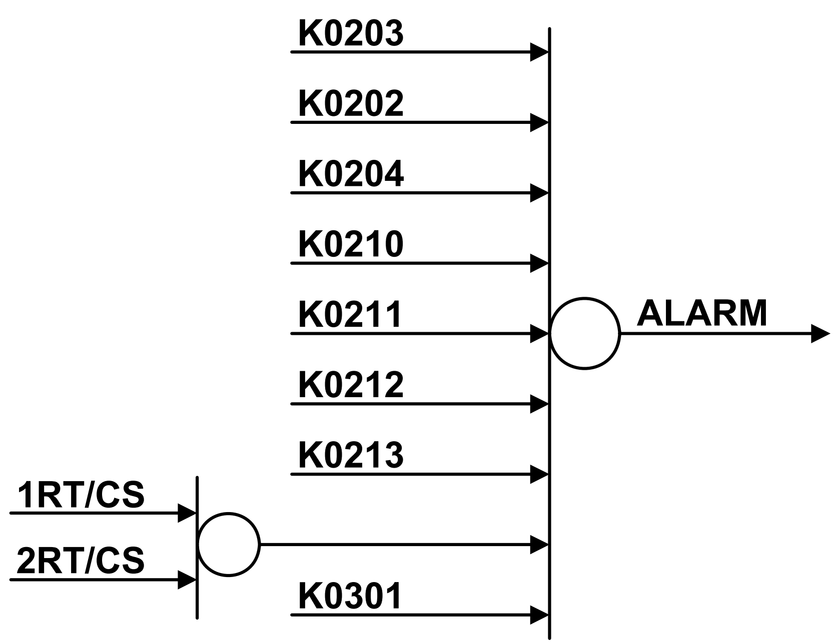
3.2 Turbine - generator trip

3.3 Steam dump activation

4. REFERENCES

4.1 501125, "Electrical Schematic Diagram – Main Annunciator" (Electrical Drawing Section 8)

5. LOGIC DIAGRAM



RO Question 2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1__	_____
	Group:	_1___	_____
	K/A:	APE 008 AA1.02	
	Importance Rating:	4.1	3.9

Proposed Question:

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

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

Which of the following actions should be taken?

A. Restart the RHR pumps.

B. Immediately restart the Charging pump

C. Proceed to the next step to determine if an SI pump can be secured.

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

Proposed Answer:

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

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

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

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

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED

15. CHECK If One CCP Should Be Stopped:

NOTE 1: If a steam or feed line break has occurred inside the aux building, then loop 1 WR T_h and/or T_c may fail low.

NOTE 2: After stopping any ECCS Pp, RCS Pressure should be allowed to stabilize or increase before stopping another ECCS Pp.

NOTE 3: The CCPs and SI Pps should be stopped on alternate ECCS Trains when possible.

a. Check both CCPs - RUNNING

RNO: a. GO TO step 16 (Page 13).

b. Determine required RCS Subcooling from table:

	RCS SUBCOOLING (°F)	
SI Pump Status	Any RCP Running	No RCP Running
None Running	74°F	87°F
One Running	53°F	53°F
Two Running	53°F	53°F

c. Check RCS Subcooling Based on core exit T/Cs GREATER THAN REQUIRED SUBCOOLING (Subcooled Margin Monitor, YI-31 or Steam Tables)

RNO: c. IF RCS Hot Leg WR Temperatures GREATER THAN 350°F,
THEN GO TO step 24 (Page 19).

IF RCS Hot Leg WR Temperatures LESS THAN 350°F,
THEN Start one RHR Pp in SI mode if none running.

IF At least one RHR Pp CANNOT be operated,
THEN GO TO step 24 (Page 19).

d. Check PZR Level - GREATER THAN 17% [50%]

RNO: d. RETURN TO step 13 (Page 9).

e. Depress Vital 4KV Auto Transfer Relay Resets: Blue Light - OFF

f. Stop one CCP

NOTE: After stopping any SI pump, RCS pressure should be allowed to stabilize or increase before stopping another SI pump.

PURPOSE: To remind the operator that RCS pressure should be allowed to stabilize before stopping another SI pump

BASIS:

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

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

RCS pressure may continue to decrease slowly as the reactor coolant temperature is reduced. However, if subcooling is increasing, the SI reduction criteria are appropriate and the SI flow can be further reduced when such criteria are satisfied.

PLANT-SPECIFIC INFORMATION:

N/A

RO Question 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	EPE 009 EK3.26	
	Importance Rating:	4.4	4.5

Proposed Question:

Given the following conditions:

- The crew is performing the actions of E-1.2, Post LOCA Cooldown And Depressurization.
- Safety Injection pumps have been stopped.
- Normal charging is aligned.

The crew is depressurizing the RCS using normal spray.

Which of the following describes the basis for the depressurization?

A. Reduce RCS break flow

B. Ensure continued RCP operation

C. Prevent a challenge to the Core Cooling Critical Safety Function

D. Maintain Pressurizer level for heater operation during the RCS cooldown

Proposed Answer:

A. Reduce RCS break flow

Explanation:

A Correct. Strategy is to depressurize and attempt to minimize subcooling so that break flow is reduced, due to the minimal makeup provided by charging pumps.

From E-1.2 background: Upon entry to Step 20, RCS injection flow will be provided by normal charging flow alone. Subcooling can then be minimized to reduce break flow and charging flow can be used to maintain pressurizer level.

B incorrect. RCP operation is not required for this event, although desired.

C incorrect. Core cooling should not be challenged at these temps and pressures (this point in the cooldown).

D incorrect. Heater operation may be required to reduce the rate of increase in pressurizer level, but is not the reason for depressurizing.

Technical Reference(s): E-1.2 background step 20

Proposed references to be provided to applicants during examination: None

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

Question Source:
Other Exam

Question History: Last NRC Exam: Beaver Valley 12/2002

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

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

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

Comments:

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

STEP: Depressurize RCS To Minimize RCS Subcooling

PURPOSE: To minimize break flow by reducing RCS subcooling

BASIS:

Upon entry to Step 20, RCS injection flow will be provided by normal charging flow alone. Subcooling can then be minimized to reduce break flow and charging flow can be used to maintain PRZR level. As in the previous depressurization (Step 11), normal PRZR spray is preferred and use of one PRZR PORV has priority over auxiliary spray. If the RCS is highly subcooled, PRZR heaters can be used to limit the PRZR level rise and maintain a steam bubble in the PRZR.

ACTIONS:

- Determine if no PRZR PORV is available for RCS depressurization
- Determine if PRZR level is greater than (D.08)% [(D.09)% for adverse containment]
- Determine if RCS subcooling based on core exit TCs is less than (R.08)°F [(R.09)°F for adverse containment]
- Depressurize RCS using normal PRZR spray
- Depressurize RCS using one PRZR PORV
- Depressurize RCS using auxiliary spray
- Turn on PRZR heaters as necessary
- Stop RCS depressurization

INSTRUMENTATION:

- PRZR level indication
- RCS pressure indication
- Core exit TCs temperature indication
- Normal PRZR spray valve position indication
- PRZR PORV position indication
- Auxiliary spray valve position indication
- PRZR heaters status indication

Given the following conditions:

"A Small Break LOCA has occurred.
"The operating crew is performing the actions of ES-1.2, Post LOCA Cooldown And Depressurization.
"Safety Injection pumps have been stopped.
"Normal charging is aligned.
"The crew is depressurizing the RCS using normal spray.

Which one of the following describes the strategy and the basis for the continuing depressurization?

Minimize subcooling to reduce RCS break flow.

Maximize subcooling to ensure continued RCP operation.

Maximize subcooling to prevent a challenge to the Core Cooling Critical Safety Function.

Minimize subcooling to ensure Pressurizer level remains above the lower limit to allow heater operation during the RCS cooldown.

Correct. Strategy is to depressurize and attempt to minimize subcooling so that break flow is reduced, due to the minimal makeup provided by charging pumps.

Incorrect. RCP operation is not required for this event, although desired.

Incorrect. Core cooling should not be challenged on loss of subcooling at these temps and pressures (this point in the cooldown).

Incorrect. Heater operation may be required to reduce the rate of increase in pressurizer level, but is not the reason for minimizing subcooling.

RO Question 4

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 015/017 AK1.01	
	Importance Rating:	4.4	4.6

Proposed Question:

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

Ten minutes after the trip, the following conditions exist:

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

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

A. Heat removal via natural circulation is not established at this time.

B. Heat removal is via forced circulation and the condenser steam dumps.

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

D. Heat removal is via natural circulation and both the condenser and 10% atmospheric steam dumps.

Proposed Answer:

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

Explanation:

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

B incorrect. RCPs are tripped in AP-11.

C correct. Per E-0.1 for natural circulation.
1) RCS subcooling based on core exit T/Cs GREATER THAN 20°F.

2) S/G Pressures stable or decreasing.
3) RCS Hot Leg Temperatures stable or decreasing.
4) Core Exit T/Cs stable or decreasing.
5) RCS Cold Leg Temperatures at saturation temperature for S/G Pressure.
The MSIVs should still be open, therefore, the condenser steam dumps are providing heat removal.

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

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

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

Learning Objective: 5810 - Interpret indication of natural circulation

Question Source:
Other Exam modified

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

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

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

Comments:

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

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
10. CHECK RCP 2 - RUNNING	Try to start RCP(s) to provide forced cooling and normal spray: a. IMPLEMENT APPENDIX B to start an RCP. b. Start RCP 2. IF RCP 2 can NOT be started, THEN Try to start other RCP(s) as necessary to provide forced cooling and normal spray. IF No RCP can be started THEN Verify Natural circulation based on: 1) RCS subcooling based on core exit T/Cs GREATER THAN 20°F. 2) S/G Pressures stable or decreasing. 3) RCS Hot Leg Temperatures stable or decreasing. 4) Core Exit T/Cs stable or decreasing. 5) RCS Cold Leg Temperatures at saturation temperature for S/G Pressure. IF Natural Circulation NOT Verified, THEN Increase Dumping Steam. -
11. CHECK Secondary Systems Status:	
a. Check Mn Fdwtr Pps Tripped o Mn Fdwtr Pps No. 1 and No. 2 Green Trip Light - ON	a. IF NOT needed for control of feedwater, THEN Trip the Mn Fdwtr Pps -
b. Stop ALL but one Condensate and Booster Pump set	

THIS STEP CONTINUED ON NEXT PAGE

015 AK1.01 5/12/2005

Beaver Valley Unit 1 R

The Unit is operating at 100% power, NSA with the exception of [PT-447], Turbine First Stage Pressure Transmitter, which is in the process of being removed from service due to erratic operation. The condenser steam dumps are in the Tavg mode. A loss of reactor plant component cooling water then results in a manual reactor trip. Ten minutes after the trip, the following conditions exist during the performance of ES-0.1, Reactor Trip Response:

SG	Pressure
1A	1000 psig and stable
1B	1005 psig and stable
1C	995 psig and stable

" All RCP's are OFF.
" RCS pressure is 2230 psig and stable.
" Thot is approximately 575 degrees F in all three (3) loops and slowly lowering.
" Core Exit TC's indicate approximately 580 F and stable.
" Tcold is approximately 555 degrees F in all three (3) loops and stable.
"Reactor Trip Breaker "A" failed to open.

Which ONE of the following describes the condition of the RCS and the preferred method of heat removal for the current plant conditions?

A. Natural circulation exists. The condenser steam dumps are maintaining heat removal.

B. Natural circulation does NOT exist. Heat removal will be established by opening the condenser steam dumps.

C. Natural circulation exists. SG atmospheric steam dump valves are maintaining heat removal.

D. Natural circulation does NOT exist. Heat removal will be established by opening the SG atmospheric steam dump valves.

B. Correct. Condenser steam dumps are the preferred method of heat removal in ES-0.2.

A. Incorrect. Natural circulation does not exist because Tcold temperatures are too high and stable.

C. Incorrect. Natural circulation does not exist because Tcold is too high for SG pressure, and not trending down. No indications that SG ADV's are open for heat removal.

D. Incorrect. Tcold is above saturation for each SG and not trending down indicating natural circulation does not exist. SG ADV's are not the preferred method of heat removal.

RO Question 5

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1__	_____
	Group:	_1__	_____
	K/A:	APE 022 AK1.02	
	Importance Rating:	2.7	3.1

Proposed Question:

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

A. To prevent passing two phase flow through the safety valves.

B. To ensure PTS limits will not be exceeded when the reactor is tripped and cools down.

C. To minimize primary-to-secondary leakage in case of a SGTR, until other recovery actions can be taken.

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

Proposed Answer:

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

Explanation:

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:
Bank – DCPP P-0619

Question History: Last NRC Exam: N/A

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

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

Comments:

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

STEP: Initiate Emergency Boration of RCS

PURPOSE: To add negative reactivity to bring the reactor core subcritical

BASIS:

After control rod trip and rod insertion functions, boration is the next most direct manner of adding negative reactivity to the core. The intended boration path here is the most direct one available, not requiring SI initiation, but using normal charging pump(s). Pump miniflow lines are assumed to be open to protect the pumps in the event of high RCS pressure.

Several plant specific means are usually available for rapid boration and should be specified here in order of preference. Methods of rapid boration include emergency boration, injecting the BIT, and safety injection actuation. It should be noted that SI actuation will trip the main feedwater pumps. If this is undesirable, the operator can manually align the system for safety injection. However, the RWST valves to the suction of the SI pumps should be opened first before opening up the BIT valves. If a safety injection is already in progress but is having no effect on nuclear flux, then the BIT and RWST are not performing their intended function, perhaps due to blockage or leakage. In this case some other alignment using the BATs and/or non-safeguards charging pump(s) is required.

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

The following plant conditions exist:

The reactor has failed to trip both automatically and manually.
Control rods are being inserted continuously.
The turbine is tripped.
The AFW pumps are running.
Emergency boration is in progress.
All PORVs are closed and their block valves open.
RCS pressure is 2400 psig.
The operating crew opens one PORV and reduces PZR pressure below 2135 psig.

Which one of the following describes why the crew opens the PORV?

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

B. To prevent a challenge to the safety valves due to the rapid overpressurization transient associated with an ATWS.

C. To minimize primary-to-secondary leakage in case of a SGTR, until other recovery actions can be taken.

D. To ensure PTS limits will not be exceeded during the subsequent RCS cooldown.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7920	Explain basis of emergency procedure step

Reference Id: P-0619
Must appear: No
Status: Active
User Text: 7920.130494
User Number 1: 0000003.00
User Number 2: 0000003.40
Difficulty: 3.00
Time to complete: 2
Topic: Why PORVs are opened in FR-S.1.
Cross Reference: LPE-S

RO Question 6

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 025 AK2.03	
	Importance Rating:	2.7	2.7

Proposed Question:

GIVEN:

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

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

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

A. Isolated to both heat exchangers.

B. Reduced to both heat exchangers.

C. Unchanged to both heat exchangers.

D. Reduced to one heat exchanger and unchanged to other.

Proposed Answer:

C. Unchanged to both heat exchangers.

Explanation:

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

B incorrect. Flow is unchanged to both heat exchangers.

C correct. Loads affected are:

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

D incorrect. Flow to both heat exchangers is unchanged.

Technical Reference(s): OP SD-4

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

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

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **GO TO Applicable Steps of this Procedure for Existing Problem with CCW System:** • HIGH CCW System Temp - steps 2, 3, 4, 5, 0 and 7 • LOW CCW System Flow - steps 7, 8, 9 and 10 • CCW System OUT-Leakage - steps 11, 12, 13 and 14 • CCW System IN-Leakage - steps 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26 and 27	
2. **PLACE a Second Train of ASW/CCW In Service per OP E-5:II Section 6.1**[T31387]	
3. **VERIFY ASW Flow to In Service CCW HX - ADEQUATE:** • ASW Pp Amps - GREATER THAN 51 Amps	GO TO OP AP SD-3, LOSS OF AUXILIARY SALT WATER, unless this procedure was entered From OP AP SD-3 or it has been determined that ASW pumps are not capable of being placed in service, then implement Appendix E, Instructions for Loss of Ultimate Heat Sink and continue with step 4.
4. **REDUCE CCW Heat Loads:**[T31387] • SHUT DOWN the BA Evap • STOP ALL RCPs • ISOLATE Letdown • SWAP ASDR to Other Unit	

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. ESTABLISH Alternative Methods of Heat Transfer From The CCW System: T34915	
a. Heat Transfer to Containment:	
1) START AT LEAST Two CFCUs in Fast Speed WITH GREATER THAN 2000 GPM CCW Flow to ANY Running CFCU	1) RUN ANY AVAILABLE CFCU
2) ESTABLISH Containment Purge Using Fans E-3 AND S-3	2) IF Containment Closure is NOT in Effect, THEN OPEN the Equipment Hatch.
b. Heat Transfer to the Spent Fuel Pool:	
1) VERIFY CCW Flow to the Spent Fuel Pool HX - AT LEAST 2800 GPM (FI-197, 100' El Penetration area)	
NOTE 1: Full flow for Pp 1 is 55 PSID.	
NOTE 2: Full flow for Pp 2 is 39 PSID.	
2) VERIFY Spent Fuel Pp at MAXIMUM Flow	
3) MONITOR Spent Fuel Pool Temperature (TIC-651, 140' El FHB)	
6. ESTABLISH BACKUP Cooling to a CCP:	
IMPLEMENT Appendix C, BACKUP COOLING TO A ECCS CENTRIFUGAL CHARGING PUMP	

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. VERIFY Pps RUNNING – AT LEAST ONE CCW Pp AND ONE ASW Pp RUNNING	REDUCE Heat Input to CCW System: • STOP ALL RCPs. • ISOLATE Letdown. • SHUT DOWN the BA Evap. • IMPLEMENT Appendix C BACKUP COOLING TO A CCP. • SWAP ASDR to Other Unit. • SHUT DOWN the Waste Gas Compressor. • GO TO OP AP SD-0, LOSS OF OR INADEQUATE DECAY HEAT REMOVAL, step 7.
8. VERIFY CCW Pps - AT LEAST TWO RUNNING	VERIFY CCW FLOW - LESS THAN 9200 GPM AND GO TO step 9.
9. VERIFY CCW Temperature: • STABLE OR DECREASING AND • LESS THAN 75°F	GO TO step 1.
10. RETURN to Procedure AND Step In Effect	
11. CHECK CCW System Makeup Status:	
a. VERIFY Makeup Water System available • Makeup Water Transfer Pump ON • Adequate source of makeup water available	a. Refer to OP F-2:VII, "Alternate Makeup Water Sources to the CCW System."
b. Verify Makeup Valves maintaining Surge Tank Level	b. OPEN Makeup Bypass Valves: • CCW-62 (LCV-69) • CCW-65 (LCV-70)

RO Question 7

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 027 AK3.01	
	Importance Rating:	3.5	3.8

Proposed Question:

GIVEN:

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

A loss of PY-11 occurs.

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

A. Address the loss of the running charging pump.

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

C. Perform the immediate actions of E-0, Reactor Trip or Safety Injection due to the energizing of a Source Range.

D. Perform the actions of AP-29, Main Turbine Malfunction, due to the failure of the PT-505 and PT-506.

Proposed Answer:

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

Explanation:

A incorrect, the running charging pump does not trip.

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

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

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

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

Proposed references to be provided to applicants during examination: none

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Pressurizer Level Channel Failures

Controlling Channel Failure	Indications	Alarms	Plant Response
Controlling PZR level fails HI	PZR level meter goes Hi PZR level recorder (if selected) goes High	PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06, PZR LEVEL HI/LO PK05-21 PZR LEVEL HI/LO CONTROL PK05-01 RCP Seal Alarms PK05-01	Backup heaters will initially energize (+5% deviation) Charging flow will decrease, causing actual level to drop, which eventually will cause letdown to isolate at 17%, and heaters to turn off. After letdown is isolated, actual Pressurizer level will start to increase and eventually cause a high level reactor trip, if the PDP is in service. If the CCP is in service FCV-128 can fully close stopping the increase in PZR level. This response of FCV-128 would also cut off flow to the RCP seals.
Controlling PZR level fails Low (or loses power)	PZR level meter goes Low PZR level recorder (if selected) goes Low	PZR LEVEL HI/LO PK05-21 PZR LEVEL HI/LO CONTROL PK05-22	Charging flow will increase. Letdown will isolate and the heaters will turn off. Actual level will increase, which will eventually cause a high level reactor trip.
Backup PZR level fails low	PZR level recorder (if selected) goes Low	PZR LEVEL HI/LO CONTROL PK05-22	Letdown will isolate and the heaters will turn off
Backup or Non-selected PZR level fails high	PZR level recorder (if selected) goes High	PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06	None

Note: No effects other than indication if LT-462 fails or loses power.

RO Question 8

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	EPE 029 EK2.06	
	Importance Rating:	2.9	3.1

Proposed Question:

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

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

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

A. Reactor Trip Breaker "B" Shunt Trip coil failed to deenergize.

B. Reactor Trip Breaker "B" Undervoltage Trip coil failed to energize.

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

D. Reactor Trip Bypass Breaker "B" Undervoltage Trip coil failed to deenergize.

Proposed Answer:

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

Explanation:

A incorrect, shunt trip coils must energize to trip.

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

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

D incorrect, bypass breaker A is closed.

Technical Reference(s): STG B6A, RPS

Proposed references to be provided to applicants during examination: None

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

Question Source:
INPO

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

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

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

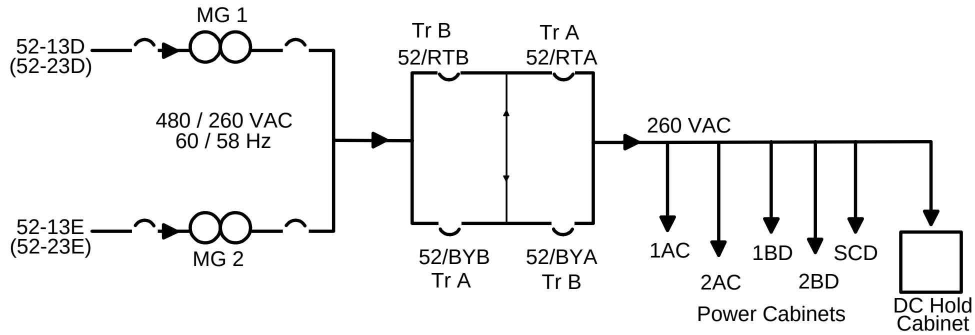
Comments:

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

Reactor Trip and Bypass Breakers, Continued

Physical description
Obj 13

The Reactor Trip and Bypass Breakers are described below.



RPS-24

Part	Function
MG Sets	Two MGs are available for converting 480 VAC power to 260 VAC required for CRDM operation.

Rx Trip Breakers	52/RTA and 52/RTB are the Reactor Trip Breakers.
	• Normally closed
	• 48 VDC must be supplied by SSPS and maintained on UV coil to maintain the breakers closed.
	  • 52/RTA supplied by SSPS Logic Train A.
	  • 52/RTB supplied by SSPS Logic Train B.
	• Breakers can also be tripped by energizing the shunt coil (125 VDC to the trip coil).

Rx Trip Bypass Breakers	52/BYA and 52/BYB are the Reactor Trip Bypass Breakers.
	• Normally open and racked out, racked in and closed to allow for trip testing.
	• 48 VDC must be supplied by SSPS and maintained on UV coil to maintain the breakers closed.
	  • 52/BYA supplied by SSPS Logic Train B.
	  • 52/BYB supplied by SSPS Logic Train A.
	• Breakers can also be tripped by energizing the shunt coil (125 VDC to the trip coil).

Continued on next page

RO Question 9

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	EPE 038 G2.4.31	
	Importance Rating:	3.3	3.4

Proposed Question:

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

While isolating Accumulators at step 34, PK05-07, Subcooling Margin Lo/Lo-Lo alarms. The CO reports the valid alarm inputs are:
- 1151 - Subcooled Mon Lo-Lo Margin From RVLIS Train A (setpoint <20°F)
- 1160 - Subcooled Mon Lo-Lo Margin From RVLIS Train B (setpoint <20°F)

What action should be taken by the crew?

A. Turn on Pressurizer heaters.

B. Continue with the isolation of the Accumulators.

C. Restart ECCS pumps and return to step 9 to re-perform the RCS cooldown and restore subcooling.

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

Proposed Answer:

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

Explanation:

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

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

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

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

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

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

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

7336 - State contents of foldout page

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

1.0 COLD LEG RECIRCULATION SWITCHOVER CRITERION

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

2.0 RESTART SAFEGUARDS EQUIPMENT AFTER LOSS OF OFFSITE POWER

IF Offsite Power is lost AFTER SI RESET

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

3.0 SI REINITIATION CRITERIA

NOTE: THIS FOLDOUT PAGE ITEM IS NOT APPLICABLE UNTIL EOP E-3, STEP 25.

IF EITHER condition listed below occurs:
- RCS subcooling based on Core Exit T/Cs – LESS THAN 20°F (Subcooled Margin Monitor, YI-31 or Appendix C)
- PZR level – CANNOT BE MAINTAINED GREATER THAN 12% [36%]

THEN Manually start ECCS pumps as necessary to restore desired condition

AND

GO TO EOP ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT, SUBCOOLED RECOVERY DESIRED

4.0 MULTIPLE TUBE RUPTURE CRITERIA

IF Any Intact S/G Level increases in an Uncontrolled manner,

OR

Any Intact S/G has Abnormal Radiation

THEN Stabilize the plant and RETURN TO EOP E-3, STEAM GENERATOR TUBE RUPTURE, Step 1

5.0 SECONDARY INTEGRITY CRITERIA

IF Any S/G Pressure is decreasing in an Uncontrolled manner or has completely depressurized, AND has NOT been isolated, unless it is needed for cooldown,

THEN GO TO EOP E-2, FAULTED STEAM GENERATOR ISOLATION.

6.0 AFW SUPPLY SWITCHOVER CRITERION

IF CST Level decreases to LESS THAN 10%

THEN IMPLEMENT OP D-1:V, ALTERNATE AFW SUPPLIES

SUBCOOLING MARGIN LO/LO-LO

06/14/05
Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
1151	Subcooled Mon Lo-Lo Margin From RVLIS Train A	K4 (from SCMM A)	< 20°F	2.1
1152	Subcooled Mon Lo Margin From RVLIS A or B	K3 (from SCMM A)	< 30°F AND below P-10	2.1
		K3 (from SCMM B)	< 30°F AND below P-10	2.1
1600	Subcooled Mon Lo-Lo Margin From RVLIS Train B	K4 (from SCMM B)	< 20°F	2.1

2. OPERATOR ACTIONS

> **NOTE:** Three of four power range channels must be below P-10 to satisfy the alarm condition for Input 1152.

2.1 General Actions **(All Inputs)**

2.1.1 Observe RCS pressures and temperatures to confirm low subcooling margin. [ ]

2.1.2 IF RCS is approaching a saturated condition, THEN perform at least one of the following, using appropriate means, to restore RCS subcooling to normal: [ ]

- Raise RCS pressure
- Lower RCS temperature

RO Question 10

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 040 AA2.01	
	Importance Rating:	4.2	4.7

Proposed Question:

The plant trips from full power due a steam break.

Plant conditions 5 minutes later:

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

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

A. Upstream of Steam Generator 12 MSIV FCV-42.

B. Downstream of Steam Generator 13 MSIV FCV-43.

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

D. On the line to the TDAFW pump, downstream of FCV-95.

Proposed Answer:

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

Explanation:

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

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

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

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

Technical Reference(s): OVID 106704 sheet 4

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

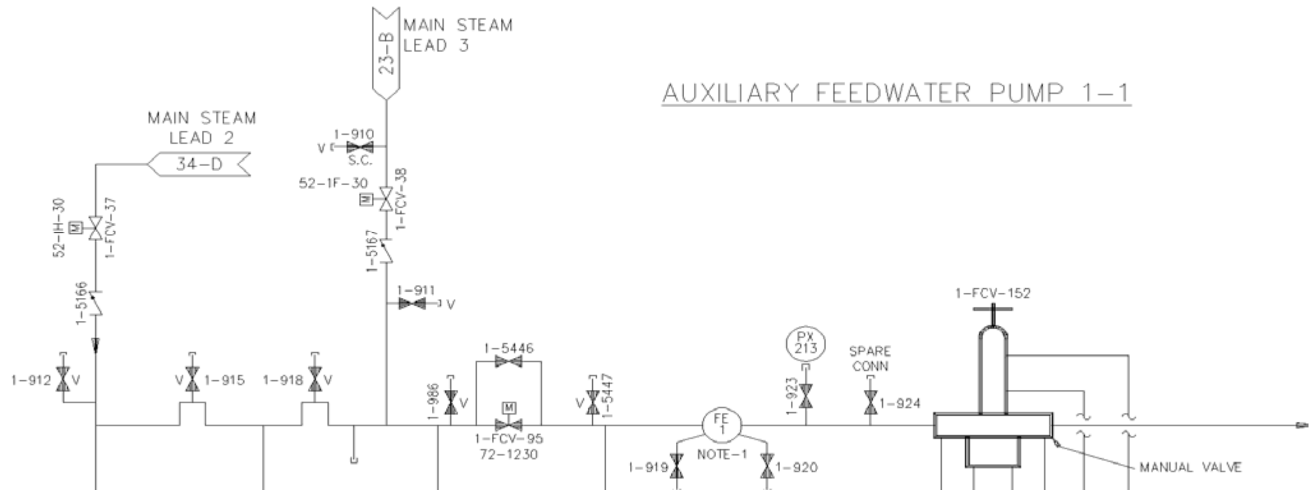
Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments:

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

OVID 106704 sheet 4



RO Question 11

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 054 G2.2.22	
	Importance Rating:	3.4	4.1

Proposed Question:

Which of the following components in the Main Feedwater system are relied upon to isolate the safety portion of the system from the non-safety portion of the system?

A. Feedwater Isolation valves only.

B. Feedwater Main Feed Reg valves only.

C. Feedwater Main Feed Reg and Bypass Valves only.

D. Feedwater Isolation valves, Main Feed Reg and Bypass valves.

Proposed Answer:

A. Feedwater Isolation valves only.

Explanation:

A correct. TS Bases B3.7.3 - The MFIVs isolate the non-safety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops….

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

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

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

Technical Reference(s): TS B3.7.3 and C8A, Main Feedwater

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Design Information, Continued

Importance to safety
Obj 4

The MFW system provides a number of safety-related functions as described below.

- The safety-related portion of MFW system consists of the four MFW Isolation Valves, their associated check valves, and the piping downstream to the S/Gs. In addition, four Design Class I Control Valves and Bypass Valves are located upstream in non-safety-related piping.
- The non safety related portion that contains the control and bypass valves has been seismically analyzed and supported to assure the integrity and operability of the Control and Bypass Valves.

Function	The MFW system must be able to …
Feedwater Isolation	stop feed flow in the event of a feedline or steam line rupture, to ensure that the minimum amount of high energy fluid is dumped into the Containment or to the environment. • The control valves and bypass valves, backed up by the main feedwater isolation valves (MFIVs), are all closed.
AFW flowpath	provide a *flow path for AFW* system from the MFW check valves to the S/Gs, to minimize the number of lines running into containment.

B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), MFRV Bypass Valves, and Main Feedwater Pump (MFWP) Turbine Stop Valves

BASES

BACKGROUND

The safety related function of the MFRVs and the MFRV bypass valves is to provide the initial isolation of main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Since the MFRVs and MFRV bypass valves are located in non-safety related piping, the MFIVs also provide safety related isolation of the MFW flow to the secondary side of the steam generators a short time later. Closure of the MFRVs and MFRV bypass valves or tripping of the MFWPs and closure of the MFIVs a short time later terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFRVs and MFRV bypass valves, or tripping of the MFWPs and closure of the MFIVs a short time later effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

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

One MFIV and one MFRV and MFRV bypass valve, are located on each MFW line, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

(continued)

BACKGROUND (continued)	The MFIVs and MFRVs and MFRV bypass valves, close on receipt of any safety injection (SI) signal, or steam generator (S/G) water level - high high signal. They may also be actuated manually. The MFWP turbine is also tripped upon receipt of an SI or S/G water level - high high signal (as well as other pump related trips), however, these are Class II trips and are only credited as a backup to the single failure of a MFRV and MFRV bypass valve trip. The MFRVs and MFRV bypass valves also close on receipt of a T_{avg} - Low coincident with reactor trip (P-4). In addition to the MFIVs and the MFRVs and MFRV bypass valves, a check valve located upstream of the MFIV is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the intact steam generators do not continue to feed the feedwater line break in the non-safety related piping upstream of the feedwater isolation check valves and that the AFW flow will be to the steam generators. A description of the MFIVs, MFRVs, and MFRV bypass valves is found in the FSAR, Section 10.4.7 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the MFIVs, MFRVs, and MFRV bypass valves is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFRVs and MFRV bypass valves, or tripping of the MFWPs and closure of the MFIVs a short time later, is relied on to terminate an SLB for core and containment response analysis and excess feedwater event upon the receipt of a feedwater isolation signal on high-high steam generator level. Failure of an MFIV, MFRV, or the MFRV bypass valves to close, or failure of the MFWPs to trip, following an SLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event. The MFIVs, MFRVs, MFRV bypass valves, and MFWP trip satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).
LCO	This LCO ensures that the MFIVs, MFRVs and MFRV bypass valves, and tripping of the MFWPs, will isolate MFW flow to the steam generators, following an FWLB or main steam line break, or an excessive feedwater event. The MFIVs will also isolate the non-safety related portions from the safety related portions of the system.

(continued)

LCO (continued)	This LCO requires that four MFIVs, four MFRVs and four MFRV bypass valves be OPERABLE. The MFIVs and MFRVs and MFRV bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal. This LCO also requires that the MFWP turbine stop valves be OPERABLE. The MFWP turbine stop valves are considered OPERABLE when their closure times are within limit and they close on a feedwater isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event and failure to meet the LCO may result in the introduction of water into the main steam lines.
APPLICABILITY	The MFIVs, MFRVs, MFRV bypass valves, and the MFWP turbine stop valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs, MFRVs, MFRV bypass valves, and the MFWP turbine stop valves are required to be OPERABLE to limit the amount of available fluid that could be added to the steam generators in the case of a secondary system pipe break inside containment or an excessive feedwater event. They are not required to be OPERABLE when the MFIVs, MFRVs, and MFRV bypass valves are closed and deactivated or isolated by a closed manual valve, or when the MFWP turbine stop valves are closed and the steam supplies to the MFWP turbine stop valves are isolated, or the MFWP discharge to the steam generators is isolated by a closed manual valve. When the MFIVs, MFRVs, and MFRV bypass valves are closed and deactivated or isolated by a closed manual valve, they are already performing their safety function. A single MFWP is operated at low power levels. It is placed in service and taken out of service at approximately 2 percent power. Before a MFWP is placed in operation, the MFWP turbine stop valves are closed and the high pressure and low pressure steam supplies to the MFWP turbine are isolated. When the MFWP turbine stop valves are closed and the steam supplies to the MFWP turbine stop valves are isolated, or the MFWP discharge to the steam generators is isolated by a closed manual valve, the safety function of the MFWP turbine stop valves is being performed.

(continued)

APPLICABILITY (continued)	In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFRVs, and MFRV bypass valves are normally closed and the MFWPs are tripped since MFW is not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

RO Question 12

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	EPE 055 EK1.02	
	Importance Rating:	4.1	4.4

Proposed Question:

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

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

A. Stopping when RCS temperature is 270°F

B. Stopping when secondary pressure is 240 psig.

C. Depressurizing the secondary at the maximum rate.

D. Maintaining intact Steam Generator levels between 20 and 50%.

Proposed Answer:

B. Stopping secondary depressurization at 240 psig.

Explanation:

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

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

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:

New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

During SG depressurization, AFW flow may have to be increased to maintain the required SG narrow range level. Control of AFW flow will have to be performed from the control room or locally depending on plant specific design. Full AFW flow should be established to any SG in which level drops out of the narrow range.

RCS cold leg temperatures should be monitored during SG depressurization to ensure that the depressurization does not impose a challenge to the Integrity Critical Safety Function. This check is included in Step 16c since guideline ECA-0.0 has priority over the Function Restoration Guidelines and the operator is instructed to not implement a Function Restoration Guideline even if a Critical Safety Function challenge is detected by the Critical Safety Function Status Trees. Consequently, Step 16c implicitly protects the Integrity Critical Safety Function. The SG depressurization should not result in a challenge to the Integrity Critical Safety Function since the resultant RCS cold leg temperatures should not approach the temperature limit (i.e., T2 temperature) at which a challenge will exist.

Once the target SG pressure is reached, the SG PORVs and AFW flow should be controlled to maintain SG pressure at the target value until ac power is restored.

The target SG pressure for Step 16 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS. The target SG pressure should be based on the nominal SG pressure to preclude nitrogen addition, plus margin for controllability (e.g., 100 psi). To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tanks pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume.

The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1 V_1^{\gamma} = P_2 V_2^{\gamma}$$

where γ = 1.25 for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta p from P_2 and adding the margin to controllability. The RCS to SG delta p should be calculated as described in the RCP TRIP/RESTART section in the Generic Issues of the Executive Volume. Instrument uncertainties are not included in the determination of the steam generator pressure limit to preclude a bias toward either having more accumulator water injected into the RCS or having less nitrogen injected into the

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. CHECK CST Level - GREATER THAN 10%	IMPLEMENT OP D-1:V, ALTERNATE AUXILIARY FEEDWATER SUPPLIES
a. Continue to monitor CST Level - GREATER THAN 10%	- -

17. DEPRESSURIZE Intact Steam Generators To Reduce RCS Pressure To Inject Accumulators:

****

CAUTION 1: Accumulator Nitrogen injection into the RCS may occur if S/Gs are Depressurized to LESS THAN 140 PSIG.

CAUTION 2: S/G NR Level should be maintained GREATER THAN 6% [16%] in AT LEAST ONE intact S/G. IF S/G Level CAN NOT be maintained, THEN S/G depressurization should be stopped until S/G Level is restored in AT LEAST ONE intact S/G.

****

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

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check S/G NR Levels - GREATER THAN 6% [16%] IN AT LEAST ONE INTACT S/G	a. Perform the following: 1) Maintain Maximum AFW flow until S/G NR Level GREATER THAN 6% [16%] in at least one intact S/G. 2) WHEN S/G NR Level is GREATER THAN 6% [16%] in at least one intact S/G, THEN Perform Steps 17b (Next Page), 17c, 17d and 17e. 3) Continue with Step 18 (Page 16) -

THIS STEP CONTINUED ON NEXT PAGE

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. <u>DEPRESSURIZE Intact Steam Generators To Reduce RCS Pressure To Inject Accumulators:</u> (Continued)	

<u>NOTE 1</u>: PZR Level may be lost and Reactor Vessel Upper Head VOIDING may occur due to depressurization of S/Gs. Depressurization SHOULD <u>NOT</u> be stopped to prevent these occurrences.

<u>NOTE 2</u>: DO <u>NOT</u> start or continue S/G depressurization per step 17b unless at least one intact S/G Level is GREATER THAN 6% [16%].

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Manually Dump Steam at MAXIMUM rate using S/G 10% steam dumps	b. Locally dump steam using 10% steam dumps.
c. Check RCS Cold Leg temperatures - GREATER THAN 270°F	c. Perform the following: 1) Control S/G 10% Steam Dumps to Stop S/G Depressurization. 2) GO TO Step 18 (Next Page).
d. Check S/G Pressures - LESS THAN 240 PSIG	d. Perform the following: 1) <u>WHEN</u> S/G Pressures decrease to LESS THAN 240 PSIG, <u>THEN</u> Perform Step 17e. 2) GO TO Step 18 (Next Page)
e. Control S/G Pressure at 240 PSIG using 10% Steam Dumps	e. Locally control S/G 10% steam dumps to maintain S/G Pressures at 240 PSIG.

RO Question 13

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 057 AA2.19	
	Importance Rating:	4.0	4.3

Proposed Question:

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

How are the indications and operation of the controller affected?

A. Lights on the controller go out; the controller fails to Manual.

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

C. Lights on the controller remain lit; the controller fails to Manual.

D. Lights on the controller remain lit; the controller fails to Auto-Hold.

Proposed Answer:

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

Explanation:

A incorrect. Controller fails to Auto-Hold.

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

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

D incorrect. Lights go out.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
DCPP Bank P-73313

Question History: Last NRC Exam: N/A

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

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

Comments:

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

NOTE 1: Manual power supplies power to the Hagan controller lights. Therefore, if manual power is lost, the lights on the controller will go out and the controller will go to the AUTO-HOLD setpoint, indicated by the controller demand meter indicating a value above zero. The controller will remain in AUTO-HOLD until manual power is restored. Once manual power is restored, the Hagan controller will transfer to Manual in 5-30 seconds. During the transfer, the controller Auto light will be lit with the controller in AUTO-HOLD. It should be noted that a "bump" could occur as the auto amplifier attempts to return the parameter to the setpoint if the controlled variable deviates from the Auto setpoint by a large amount.

NOTE 2: When both the Manual and Auto power are lost, the controller output will go to zero. One way to quickly recognize this is that the demand meter reads zero and the lights on the controller are out.

The following is a summary of Hagan controller response to a temporary loss of power (with the exception of LCV-8 and 12):

INITIAL CONDITION	TYPE OF POWER FAILURE	RESULTS	
		*INTERIM	FINAL
Auto	Loss of Auto Power momentarily	Manual	Manual
Auto	Loss of Manual Power momentarily	Auto Hold	Manual
Manual	Loss of Manual Power momentarily	Auto Hold	Manual
Manual	Loss of Auto Power momentarily	Manual	Manual

* The Hagan controller will be in this condition until power is restored.

7970 Identify failure modes associated with Hagan controllers

P-73313	Points:	1.00	Multiple Choice

The Pressurizer Pressure controller is operating in AUTOMATIC.

A loss of PY-11 occurs which causes a loss of MANUAL power to the controller.

How are the indications and operation of the controller affected?

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

B. Lights on the controller remain lit; the controller still controls in automatic.

C. Lights on the controller go out; the controller fails to Manual.

D. Lights on the controller remain lit; the controller fails to Auto Hold.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7970	Identify failure modes associated with Hagan controllers

Reference Id: P-73313
Must appear: No
Status: Active
User Text: 4303.030853
User Number 1:
User Number 2:
Difficulty: 3.00
Time to complete: 2
Topic: LPA-4, loss of manual power to pzr press controller
Cross Reference: LPA4,OBJ3(4303) PG 6&7
Comment: loss of manual power to a Hagan controller causes all lights to go out. in addition, the controller shifts to auto-hold. when power is restored, the lights will re-energize and the controller will control in manual.
reference: OP O-2, attachment 9.1

RO Question 14

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 062 AA2.02	
	Importance Rating:	2.9	3.6

Proposed Question:

GIVEN:

- Unit 1 is in a normal full power lineup.
- STP M-9A, Diesel Engine Generator Routine Surveillance Test is being performed on DEG 12.
- The diesel has been paralleled with offsite and loaded to 2.5 MW.

The running ASW pump, 11 trips.

Which of the following describes what could have tripped ASW pump 11 and response of ASW pump 12?

A. Pump 11 trips on overcurrent. Pump 12 will not start.

B. Pump 11 trips on overcurrent. Pump 12 starts on low system pressure.

C. Pump 11 trips on low system pressure. Pump 12 will not start.

D. Pump 11 trips on low system pressure. Pump 12 starts when the breaker for 11 opens.

Proposed Answer:

B. Pump 11 trips on overcurrent. Pump 12 starts on low system pressure.

Explanation:

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

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 5365 - Analyze ASW pump control logic.

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

ASW Pumps

Purpose
Obj 2,3, 21

The purpose of the ASW pumps is to provide the driving head necessary to deliver saltwater to the CCW heat exchangers at a sufficient flow rate to remove the design heat load.

Sensitive Information

Each pump is in a separate room with a watertight door.

Power supplies
Obj 4

The power supplies for the pump motors are:

ASW Pump	4 kV vital bus
1-1	F
1-2	G
2-1	F
2-2	G

Continued on next page

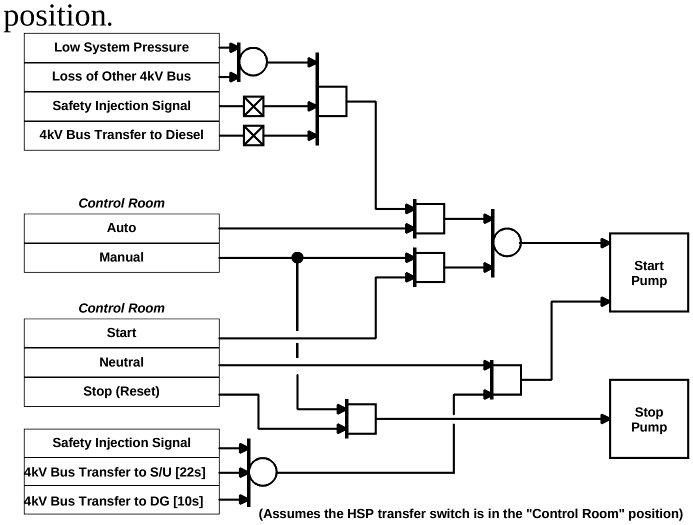
ASW Pumps, Continued

Logic (continued)

When controlling ASW pumps from the HSDP,
- all automatic trips continue to function normally.
- the manual trip at the HSDP and the switchgear still operate normally.
- all ASW pump lights in the Control Room are extinguished.

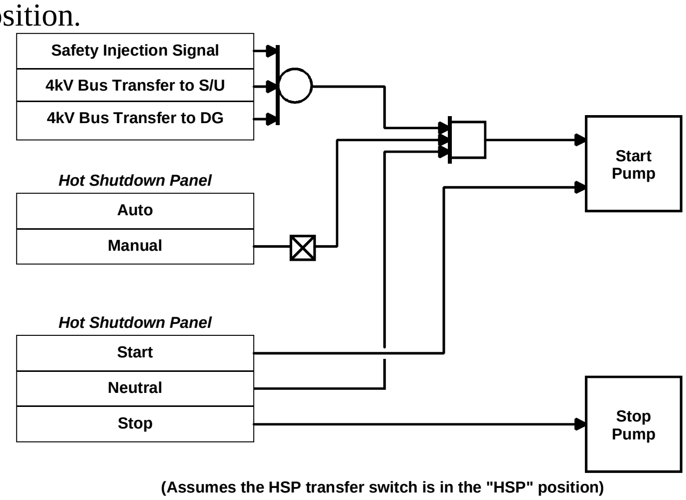
Logic diagrams for pump starts and stops from the control schemes
Obj 9

The following logic diagram assumes the transfer switch at the breaker in the CONT RM position.



ASW-05

The following logic diagram assumes the transfer switch at the breaker in the HSD PNL position.



ASW-06

Continued on next page

ASW Pumps, Continued

Trips
Obj 9, 27

The ASW pumps have these automatic trips:
- overcurrent
- 4 kV bus undervoltage (load shed)

Indications
Obj 7, 8, 22

The following indications are available for the ASW pumps:

VB1 START/STOP switch		
Indicating Light	**Meaning**	**Normal Status**
Red	Pump motor breaker is closed.	ON/OFF
Green	Pump motor breaker is open.	OFF/ON
White	Pump available for remote start from • control room • Safety Injection with time delay	ON
Amber	Pump in AUTO, control at VB1.	ON
Blue	Pump tripped on overcurrent.	OFF

ASW Pump #1		
Indication	**Can be read at ...**	**Normal reading**
PI-452, Discharge Press	Pump room	49 psig
Pump amps	VB1	56 A

ASW Pump #2		
Indication	**Can be read at ...**	**Normal reading**
PI-454 Discharge Press	Pump room	49 psig
Pump amps	VB1	56 A

ASW Pump #1 and #2 Monitor Light Box C on VB1		
Indicating Light	**Meaning**	**Normal Status**
White	Pump NOT RUNNING	OFF*

*The Monitor Light Box is normally de-energized and is activated 20 seconds after an actuation signal to monitor the status of affected equipment.

Continued on next page

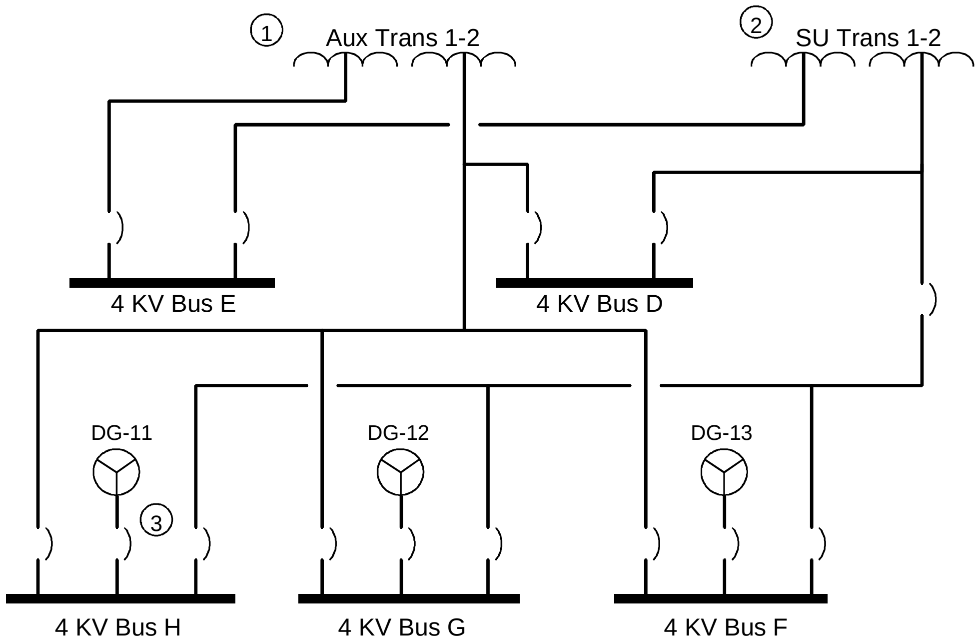
Basic Description

Purpose of the 4 kV system
Obj 1

The purpose of the 4 kV system is to supply electrical power to:
- Various motors (<3000 hp).
- 480 v motor control centers (MCC).
- the 230 kV switchyard service power transformer.
- the 500 kV switchyard service power transformer.

Description 4 kV system

The 4 kV system is shown here.



4KV-01

Each unit has five 4 kV buses divided into two groups designated as:
- Nonvital
  - Buses D and E.
- Vital
  - Buses F, G, and H.

The nonvital group supplies power to the nonsafety-related and secondary plant 4 kV loads.

The vital group supplies power to the reactor protection and engineered safeguard loads.

Continued on next page

RO Question 15

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	APE 065 AA1.02	
	Importance Rating:	2.6	2.8

Proposed Question:

Unit 1 is at full power.

An air leak develops upstream of instrument air valve 221.

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

A. Both headers will rapidly depressurize until FCV-584 closes to isolate both headers.

B. Both headers will rapidly depressurize until FCV-584 closes to isolate containment. The penetration area will continue to depressurize.

C. A check valve between the break and the Backup Air Receiver will minimize the effect on both headers.

D. A check valve between the break and the Backup Air Receiver will minimize the effect on the containment header. The penetration area header will depressurize.

Proposed Answer:

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

Explanation:

A incorrect. Check valve will isolate the break.

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

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

D incorrect. The check valve is before the Receiver.

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

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

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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


255
256
257
258
259
1-585
S.C.
1-1021
1-227
C-1132-2
1-2496
1-FCV-584
A-16
1-226
C-2067-2 EL. 110'
AINMENT INST AIR
A-12
A-11
A-10A
A-10
A-9A
1-1015
1-1014
1-1013
1-1012
1-1011
1-RV-241
1-231
1-223
1-302
1-221
C-1
EL. 109'-6"
269-A
A-9
E
SV 325
1-1273
1-1272
1-1272A
1-3983
1-1271
1-RV-360
T.C.
1-4159
TRP 17
1-222
1-2138
1-3357
1-2136
1-2139
P I 366
C-1431 EL. 107'-9 3/4"
-9354B
ZR STM
E (OC)
SV 326
1-1274
1-1271A
B.U. A/S
RECEIVER
FOR RCS
SAMPLE
VLV 9356B
INSTRUMENT AIR
BACKUP RECEIVER 1-1
EL. 100'
D
-9355B
ZR LIQ
E (OC)
1-1277
1-1275
1-2501
P-100G-02
344-E
1-4158
T.C.
1-1266
RESIDUAL HEAT
EXCHANGER
1-2133
1-RV-417
1-1261
1-2134
1-1267
1-5073
1-5074
1-1267A
1-1261A
1-2133A
C
SV 328
1-1276
1-1278
PO 637
FM 637
1-1262
R 637
P I
1-5075
ATC/FO
1-HCV-637
TO COLD LEG
LOOP 3 & 4
SV 1428
-9357B
SAMPLE
R (OC)
1-1280
1-1279
1-1279A
1-1268
1-5079
1-5080
SV 513
1-1281
1-1268A
1-FCV-1428
POST LOCA
SAMPLE RETURN
TO CONTAINMENT
PO 670
FM 670
1-1263
V-254
N PPS
(OC)
PANEL 184 EL. 100'
R 670
B
1-5081
SV 515
1-1282
1-1285
1-1286
SV 355
1-1285A
ATO/FC
ACCUM
TEST (OC)
1-8961
ATC/FO
1-HCV-670
BYPASS
1-3355
V-256
DRAIN
(OC)
1-1288
1-1287
SV 357
ATO/FC
1-1270
ACCUM N2
SUPPLY HDR
1-8880
1-1269
1-5076
1-5077
1-1269A
SV 532
1-1283
1-1284
PO 638
FM 638
1-1264
1-1289
SV 360
1-1289A
R 638
1-1265
ATO/FC
SAFETY INJ
PP TEST LINE
1-8883
1-5078
A
V-257
IN TK
(OC)
PANEL 76
EL. 100'
ATC/FO
1-HCV-638
TO COLD LEG
LOOP 1 & 2
1-1290
1-1291
SV 314
1-1290A
ATO/FC
1-FCV-681
CONTAINMENT
AIR SAMPLE (OC)
UNIT ONE
PG & E CO.
SHEET 25
106725
REV. 122

RO Question 16

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	EPE E04 EK3.2	
	Importance Rating:	3.4	4.0

Proposed Question:

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

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

A. Stop the RHR pumps to conserve RWST inventory.

B. Stop one train of injection to conserve RWST inventory.

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

D. Commence makeup to the RWST while attempting to locate the break because there is no inventory in the Containment sump.

Proposed Answer:

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

Explanation:

A and B incorrect. These types of actions are taken by other EOPs, such as ECA-1.1 but not ECA-1.2

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

D incorrect. RWST makeup is not performed in ECA-1.2

Technical Reference(s): ECA-1.2 background

Proposed references to be provided to applicants during examination: ECA-1.2

Learning Objective: 5433 - Identify exit conditions for the EOPs

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

STEP: Check If Break Is Isolated

PURPOSE: To determine if the LOCA outside containment has been isolated from previous actions

BASIS:

This step instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions. If the break is isolated in Step 2, a significant RCS pressure increase will occur due to the SI flow filling up the RCS with break flow stopped.

The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump.

ACTIONS:

o Determine if RCS pressure is increasing
o Transfer to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1
o Transfer to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1

INSTRUMENTATION:

RCS pressure indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

It should be noted that for some breaks SI flow may cause an RCS pressure increase independent of break isolation. It should also be noted that for larger breaks, RCS repressurization may be delayed following break isolation. Additionally, if the RCS is saturated or a cooldown is in progress, RCS repressurization will proceed more slowly. Other means of verifying break isolation should be checked. For example, increasing RVLIS trend due to injection flow, decreasing trends in local abnormal conditions and local observation (if practical) may be useful.

RO Question 17

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	E05 G2.1.27	
	Importance Rating:	2.8	2.9

Proposed Question:

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

A. Prevent lifting the Pressurizer safeties.

B. Delay the time to steam generator dryout.

C. Depressurize the RCS to allow automatic SI actuation.

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

Proposed Answer:

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

Explanation:

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

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

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Step description:
STEP DESCRIPTION TABLE FOR FR-H.1Step 15

STEP: Establish RCS Bleed Path

PURPOSE: To open all PRZR PORVs to establish an RCS bleed path

BASIS:

The operator ensures that the pressurizer block valves are open and opens both pressurizer PORVs to establish an RCS bleed path. These valves must be maintained in the open position until secondary heat sink is restored.

Once the pressurizer PORVs are open, the RCS will depressurize and the charging/SI pumps and/or high-head SI pumps will deliver subcooled flow to the RCS. This will provide adequate RCS heat removal until flow can be established to the steam generators to restore secondary heat sink.

ACTIONS:

- o Determine if power is available to PRZR PORV block valves
- o Determine if PRZR PORV block valves are open
- o Restore power to block valves
- o Open block valves
- o Open all PRZR PORVs

INSTRUMENTATION:

- o PRZR PORV and block valve position indication
- o Indication of power available to PRZR PORV block valves

CONTROL/EQUIPMENT:

- o PRZR PORV and block valve switches
- o Block valve power supply controls

RO Question 18

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_1___	_____
	K/A:	EPE E11 EA1.1	
	Importance Rating:	3.9	4.0

Proposed Question:

GIVEN:

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

One train of ECCS is to be stopped.

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

A. Locally reset each 4 kV Vital Bus Auto Transfer Trip Relay, only.

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

C. Depress Vital 4KV Auto Transfer pushbutton for each bus on VB4 and locally reset each 4 kV Vital Bus Auto Transfer Trip Relay.

D. Depress Vital 4KV Auto Transfer pushbutton for each bus on VB4 and place each Bus Auto Xfer to S/U Cutout Switch to CUTOUT.

Proposed Answer:

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

Explanation:

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

B correct. This is the only action required.

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
11. Check if ECCS Is In Service	GO TO Step 22 (Page 11)
• SI Pps - ANY RUNNING OR • Charging Injection - NOT ISOLATED OR • RHR Pps - ANY RUNNING	- -

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

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
12. ESTABLISH One Train Of SI Flow:	
a. Depress Vital 4KV Auto Transfer Relay Resets: Blue Light - OFF	
b. Verify CCPs - ONLY ONE RUNNING	
c. Verify SI Pps - ONLY ONE RUNNING	
d. RCS Pressure - LESS THAN 300 PSIG	d. Stop Both RHR Pps AND GO TO Step 13 (Next Page). -
e. Verify RHR Pps - ONLY ONE RUNNING	

Bus Auto Xfer to S/U Cutout Switch

Purpose
Obj 4, 5

The purpose of the Bus Auto Xfer to S/U Cutout Switch is to allow the operators to block the actuation of the transfer to S/U relay. The reasons this may be done are:

- Allow for the controlled restoration from a loss of power.
- Allow for the disabling of one 12kv bus auto transfer during degraded 230 kv grid conditions.

Location

All of the Bus Auto Xfer to S/U Cutout Switches are located in the Control Room on VB4/VB5 next to the bus that they control.

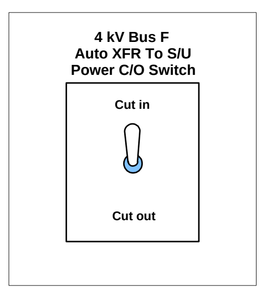
Description

A Bus Auto Xfer to S/U Cutout Switch is available for the following buses.

- 12 kV Bus D
- 12 kV Bus E
- 4 kV Bus D
- 4 kV Bus E
- 4 kV Bus F
- 4 kV Bus G
- 4 kV Bus H

Controls

A Bus Auto Xfer to S/U Cutout Switch shown below is available in the Control Room.



EPT-02

Control	Operation
CUT IN/CUT OUT	2 position maintained toggle

Pages 2.1-5 and 2.1-6

Bus Auto Xfer Reset Pushbutton

Purpose
Obj 4, 5

The purpose of the Bus Auto Xfer Reset Pushbutton is to reset the associated Auto Transfer Relays and remove the associated control functions.
Note: Refer to Section 2.2 for a description of the functions associated with each of the relays.

Location

All of the Bus Auto Xfer Reset Pushbuttons are located in the Control Room on VB4/VB5 next to the bus that they control.

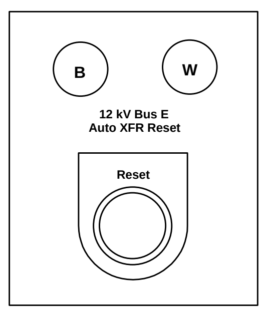
Description

A Bus Auto Xfer Reset Pushbutton is available for the following buses.

- 12 kV Bus D
- 12 kV Bus E
- 4 kV Bus D
- 4 kV Bus E
- 4 kV Bus F
- 4 kV Bus G
- 4 kV Bus H

Controls

A Bus Auto Xfer Reset Pushbutton shown below is available in the Control Room.



EPT-03

Control	Operation
RESET	pushbutton

Continued on next page

Logic
Obj 3

The logic associated with the Bus Auto Xfer Reset Pushbutton is described in the table below.

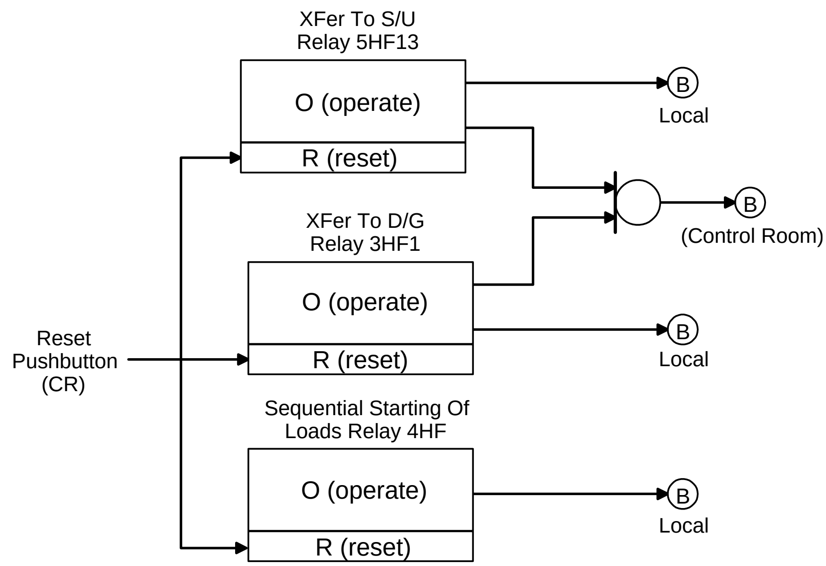
If the pushbutton is ...	And ...	Then the following relays and associated actuating signals will be reset
Depressed	The initiating logic to the relay has been cleared.	• 12 kV fast auto Xfer • 12 kV slow auto Xfer • 4 kV Xfer to S/U • 4 kV Xfer to D/G (vital only) • 4 kV sequential starting of loads (vital only)

Note: The initiating logic is automatically cleared when the S/U feeder or D/G output breaker supplying the bus is closed.

Note: Refer to Section 2.2 for a description of the initiating logic and associated functions for each of the relays.

Logic diagram
Obj 3

The logic diagram for the Bus Auto Xfer Reset Pushbutton is shown below.



Bus Auto Xfer Reset Pushbutton, Continued

Logic
Obj 3

The logic associated with the 4 kV Vital Bus Auto Transfer Trip Relay Reset Device is described in the table below:

If the auto transfer to S/U lock-out relay reset switch is taken to	And ...	Then...
RESET	The initiating logic to the relay has been cleared.	The lockout relay will be returned to its RESET position.
	The relay still has its initiating logic made-up.	The lockout relay will spring return to the TRIP position.

Note: The reset device is procedurally never manually taken to the TRIP position. Doing so could cause damage to the lock-out relay.

Note: The initiating logic is automatically cleared when the S/U feeder or D/G output breaker supplying the bus is closed.

Note: Refer to Section 2.2 for a description of the initiating logic and associated functions for each of the relays.

RO Question 19

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	APE 001 G2.4.31	
	Importance Rating:	3.3	3.4

Proposed Question:

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

Rods begin to step out at 8 steps per minute.

The following alarms are received:
- PK04-03, Tavg Deviation From Tref
- PK05-16, Pzr Pressure High
- PK05-22, Pzr Level Hi/Lo

The operator reports Tref is unchanged.

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

A. Trip the reactor due to a vapor space leak.

B. Place rods in Manual due to a dropped rod.

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

D. Place rods in Manual due to a power range instrument failure.

Proposed Answer:

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

Explanation:

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

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

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

TITLE: Continuous Withdrawal or Insertion of a Control Rod Bank

1 AND 2

07/30/02
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

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).

RO Question 20

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	APE 028 AK2.02	
	Importance Rating:	2.6	2.7

Proposed Question:

GIVEN:

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

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

A. LT-459 fails low.

B. LT-461 fails low.

C. LT-459 fails high.

D. LT-461 fails high.

Proposed Answer:

A. LT-459 fails low.

Explanation:

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

B incorrect, backup channel does not affect charging flow.

C incorrect, failing high causes charging to decrease.

D incorrect, charging not affected by backup channel.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
DCPP P-1509

Question History: Last NRC Exam: N/A

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

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

Comments:

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

#	1.00

Given the following:

Unit 1 is at 100% power
All control systems are in their normal alignment
Pressurizer Level Control is selected to 459/461

Which of the following Pressurizer level failures would result in the fastest increase in reactor coolant inventory?

A. LT-459 fails LOW.

B. LT-461 fails HIGH.

C. LT-461 fails LOW.

D. LT-459 fails HIGH.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

36926	Explain the effects of Pressurizer system failures, including: ? Transmitters ? Valves o PORVs and Safeties (including leakage or water relief) o Spray Valves (Normal or Aux)
4621	Explain the effects of Pressurizer Level Transmitter failures

Reference Id: P-1509
Must appear: No
Status: Active
User Text: 4621.020045
User Number 1: 0000003.10
User Number 2: 0000003.90
Difficulty: 4.00

Pressurizer Level Channel Failures

Controlling Channel Failure	Indications	Alarms	Plant Response
Controlling PZR level fails HI	PZR level meter goes Hi PZR level recorder (if selected) goes High	PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06, PZR LEVEL HI/LO PK05-21 PZR LEVEL HI/LO CONTROL PK05-01 RCP Seal Alarms PK05-01	Backup heaters will initially energize (+5% deviation) Charging flow will decrease, causing actual level to drop, which eventually will cause letdown to isolate at 17%, and heaters to turn off. After letdown is isolated, actual Pressurizer level will start to increase and eventually cause a high level reactor trip, if the PDP is in service. If the CCP is in service FCV-128 can fully close stopping the increase in PZR level. This response of FCV-128 would also cut off flow to the RCP seals.
Controlling PZR level fails Low (or loses power)	PZR level meter goes Low PZR level recorder (if selected) goes Low	PZR LEVEL HI/LO PK05-21 PZR LEVEL HI/LO CONTROL PK05-22	Charging flow will increase. Letdown will isolate and the heaters will turn off. Actual level will increase, which will eventually cause a high level reactor trip.
Backup PZR level fails low	PZR level recorder (if selected) goes Low	PZR LEVEL HI/LO CONTROL PK05-22	Letdown will isolate and the heaters will turn off
Backup or Non-selected PZR level fails high	PZR level recorder (if selected) goes High	PROTECTION CHANNEL ACTIVATED (High Level trip) PK04-06	None

Note: No effects other than indication if LT-462 fails or loses power.

RO Question 21

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	APE 033 AK3.01	
	Importance Rating:	3.2	3.6

Proposed Question:

GIVEN:
- Plant startup in progress
- N35 is reading 2 x 10^{-11} amps due to compensation problems
- N36 is steady at 1 x 10^{-8} amps while taking critical data
- A repair plan is being prepared for N35

N36 loses detector voltage and fails low.

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

A. Verify the reactor trip, Source Ranges are energized.

B. Reduce power to less than P-6 within the next 2 hours, Source Ranges will automatically energize.

C. Verify the reactor trip and manually energize Source Ranges.

D. Reduce power to less than P-6 within the next 2 hours and manually energize Source Ranges.

Proposed Answer:

A. Verify the reactor trip, Source Ranges are energized.

Explanation:

A correct, with 2 Intermediate Ranges below P-6, the Source Ranges will enegize and the unit will trip on high Source Range flux.

B incorrect, this is the action for 2 inoperable IR's.

C incorrect, Source Ranges will be energized.

D incorrect, Tech Spec action for 2 failed IR's.

Technical Reference(s): Technical Specification 3.3.1 Action G, OIM B-2.6a

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

Learning Objective: 36973 - Explain the effect of Excore NIS channel failures, including:
- Individual Channel Failures
- Loss of power supplies/fuses

Question Source:
New

Question History: Last NRC Exam: N/A

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

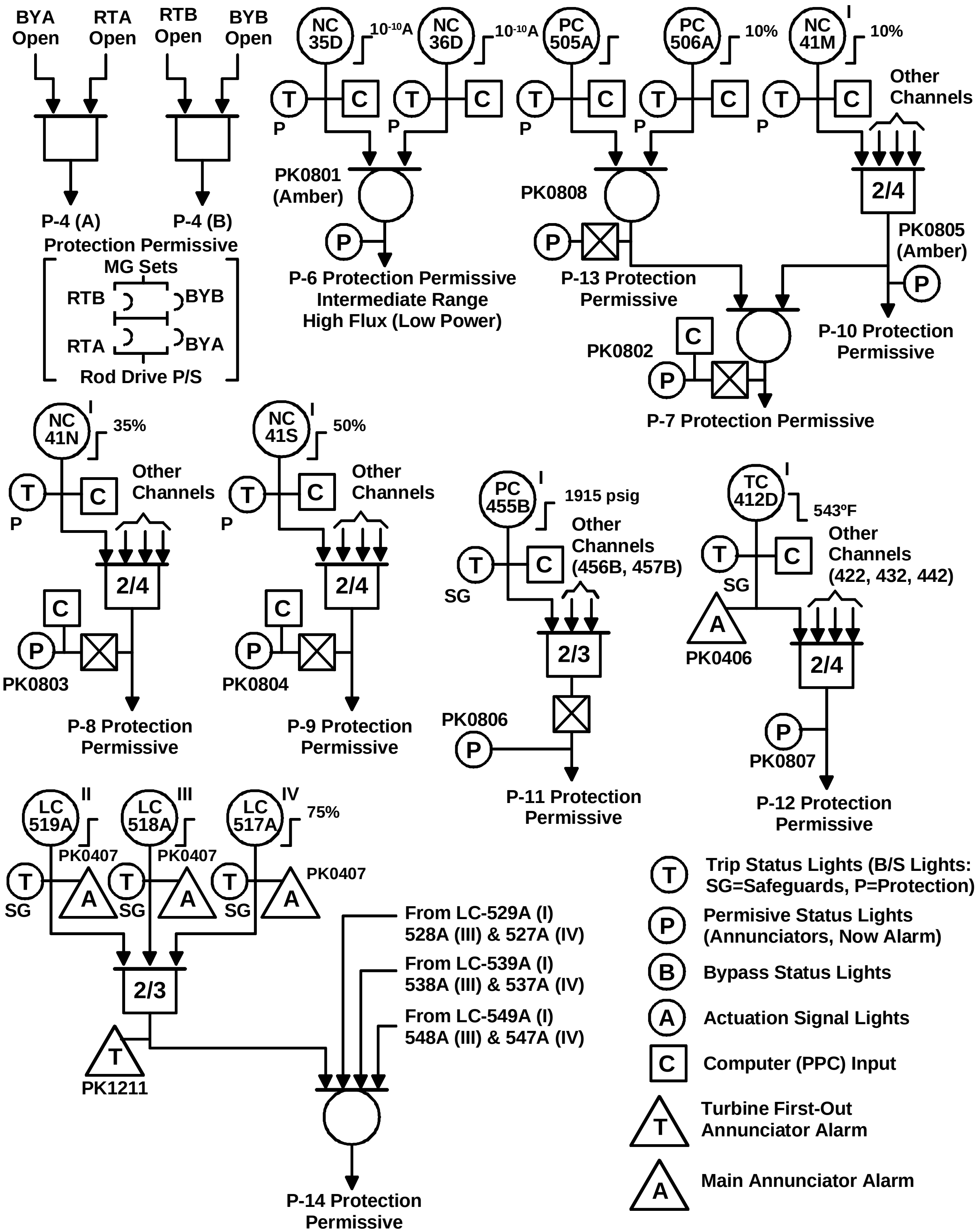
10 CFR Part 55 Content: 55.41.5 - Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 -----------NOTE------------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. ------------------------------- Suspend operations involving positive reactivity additions.	Immediately
	AND	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. Not used		
I. One Source Range Neutron Flux channel inoperable.	I.1 -----------NOTE------------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. ------------------------------- Suspend operations involving positive reactivity additions.	Immediately
J. Two Source Range Neutron Flux channels inoperable.	J.1 Open reactor trip breakers (RTBs).	Immediately

Protection Permissives Logics



RO Question 22

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	APE 036 AA2.03	
	Importance Rating:	3.1	4.2

Proposed Question:

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

A. 1750

B. 2000

C. 2250

D. 2500

Proposed Answer:

D. 2500

Explanation:

D correct. ECG 42.4 requires the operator to immediately place the crane in a safe location if load over the Spent Fuel Pool exceeds 2500 pounds. In the basis, the reason is to limit a potential release to a single fuel assembly and prevent possibly causing a critical array.

Technical Reference(s): ECG 42.4.

Proposed references to be provided to applicants during examination: None

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

Question Source:
INPO

Question History: Last NRC Exam: Braidwood, 2001

Question Cognitive Level:

Memory or Fundamental Knowledge X
Comprehension or Analysis ____

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

Comments:

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

..036.AA2.03 10/29/2001

Braidwood Unit 1 R

Mark Question | Print Recor | New Sear

Limiting the magnitude of a potential release during a fuel handling accident is accomplished by limiting the maximum load traveling over the fuel assemblies in the Spent fuel Pool to LESS THAN OR EQUAL TO _______.

2000 lbs.

1000 lbs.

1500 lbs.

2500 lbs.

42.0 FUEL HANDLING SYSTEM

42.4 Refueling Operations - Crane Travel - Fuel Handling Building

ECG 42.4 No load > 2500 pounds shall travel over fuel assemblies in the spent fuel pool.

-----NOTE-----
The movable fuel handling building walls may travel over fuel assemblies in the spent fuel pool.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTIONS

-----NOTE-----
Prior to exceeding the Completion Time of any Required Action, a 10 CFR 50.59 evaluation must be approved by the PSRC justifying the acceptability of exceeding the Completion Time.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Load > 2500 pounds traveling over fuel assemblies in spent fuel pool.	A.1 Place crane load in a safe condition.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 42.4.1	Verify loads $\leq$ 2500 pounds.	Prior to movement over fuel assemblies in the spent fuel pool.

SURVEILLANCE	FREQUENCY

BASES

BACKGROUND	Fuel handling operations within the fuel handling area of the fuel handling building are accomplished with overhead cranes, specially designed fuel grapples, and miscellaneous other equipment. To effect the transfer of the fuel between the fuel handling building and containment, an underwater penetration called the transfer tube is provided through the adjoining walls. A conveyor cart is used to transport the fuel from the fuel handling building to the containment through this penetration (Reference 1). The restriction on movement of loads in excess of the nominal weight of a fuel assembly, rod cluster control assembly and associated handling tool (approximately 2500 pounds), except the movable fuel handling walls, over other fuel assemblies in the spent fuel pool (SFP) ensures in the event this load is dropped that: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of the fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analysis. The movable fuel handling building walls travel on rollers over the SFP and have been designed to remain in place during postulated seismic events.
APPLICABLE SAFETY ANALYSES	Dropping a fuel assembly (or the equivalent weight) on another fuel assembly is consistent with the assumptions and limitations of a fuel handling accident in the fuel handling building. The consequences of a fuel handling accident in the fuel handling building are described in Reference 2. Loads in excess of 2,500 pounds with the exception of the movable fuel handling building walls, are prohibited from travel over fuel assemblies in the SFP whenever fuel assemblies are in the pool, to preclude mechanical damage to the spent fuel from a postulated heavy load drop event. A heavy load is defined as any load, carried in a given area after a plant becomes operational, that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool (Reference 3). Crane travel is restricted by redundant electrical interlocks installed on the fuel handling building crane based on the guidelines in Reference 4.
LCO	Limiting loads < 2500 pounds, except the movable fuel handling walls, over other fuel assemblies in the SFP ensures in the event this load is dropped that the assumptions of the accident are maintained. 2500 pounds is the approximate weight of a fuel assembly and control assembly and associated handling tool. This is the assumed load in the fuel handling accident. The movable fuel handling building walls travel on rollers over the SFP and have been designed to remain in place during postulated seismic events.

(continued)

BASES (continued)

APPLICABILITY	Loads in excess of 2500 pounds are prohibited from travel over fuel assemblies in the SFP, whenever fuel assemblies are in the SFP, to preclude mechanical damage to the spent fuel from the drop of a postulated heavy load.
ACTIONS	A.1 With load > 2500 pounds immediately place crane load in a safe condition. This will prevent the load from being dropped, which could result in radiological consequences different from those previously analyzed in the fuel handling accident.
SURVEILLANCE REQUIREMENTS	SR 42.4.1 Loads shall be verified $\leq$ 2500 pounds prior to movement over fuel assemblies in the SFP. This will ensure that if the load were dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of the fuel in the storage racks will not result in a critical array.
REFERENCES	1. DCM T-11, "Control of Heavy Loads." 2. FSAR Update Section 15.22. 3. NUREG 0612 (July 1980), "Control of Heavy Loads at Nuclear Power Plants," Section 5.1.1. 4. FSAR Update Section 9.1.2.3.

06/27/00
Effective Date

RO Question 23

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	APE 059 AA2.05	
	Importance Rating:	3.6	3.9

Proposed Question:

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

Which of the following automatic action(s) occur?

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

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

C. RCV-18 closes and all liquid radwaste pumps that are running receive a trip signal.

D. RCV-18 (liquid radwaste overboard) remains open and FCV-477 (recirc to the EDR) opens.

Proposed Answer:

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

Explanation:

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

B incorrect, the EDR recirc opens.

C incorrect, pumps do not trip.

D incorrect, RCV-18 closes.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank - DCPP P-0179

Question History: Last NRC Exam: N/A

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

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

Comments:

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

1.00

A HIGH RADIATION alarm is received on RE-18, the liquid radwaste rad monitor. Which of the following automatic valve action(s) occur?

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

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

C. RCV-18 (liquid radwaste overboard) remains open, alarm is used as a warning only.

D. RCV-18 (liquid radwaste overboard) remains open and FCV-477 (recirc to the EDR) opens.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8436	Analyze Liquid Rad Waste system control logic ? Containment Liquid Radwaste Isolation ? LRW Discharge High Radiation
69251	Explain the automatic actions associated with the Liquid Radwaste system.

Reference Id: P-0179
Must appear: No
Status: Active
User Text: 8442.130494
User Number 1: 0000002.90
User Number 2: 0000003.40
Difficulty: 0.00
Time to complete: 2

Radiation Monitors

Number	Use	Detec for Type	Location	Auto Actions	C/R Ind.
0-RM-1	Control Room	GM	By Rad Mon Rack	None	Y
RM-2 *	Containment	GM	By Pers. Hatch	None	Y
0-RM-3	Oily Wtr Sep	Scin.	85' OWS Room	None	Y
RM-4	Recip Charging	GM	73' Pump Room	None	Y
RM-6	NSSS Sample	GM	100' Aux Bldg	None	Y
RM-7	Incore Seal Tbl	GM	119' Containment	None	Y
0-RM-10	Aux Cont Bd	GM	85' Aux Bldg	None	Y
RM-11	Cont Air Part	Scin.	100' GE	None	Y
RM-12	Cont Rad Gas	GM	100' GE	None	Y
RM-13	RHR Exh Duct	Scin.	113' Aux Bldg Hall	None	Y
RM-14, 14R, 87	Pint Vnt Rad (Gas)	Beta Scint	85' FHB White Room	RM-14 auto starts RM-87 when RM-14 reaches high and of range	Y
RM-15, 15R	Air Ejectors	Scin.	104' Turb Bldg	None	Y
RM-17A/B	CCW	Scin.	73' Aux Bldg	Shuts Vent (RCV-16)	Y
0-RM-18	Liquid Radwaste	Scin.	55' Pipe Tunnel	Shuts OVBD (RCV-18) Opens EDR (FCV-477)	Y
RM-19	S/G B/D Sample	Scin	100' GE (Dual Setpoint)	Shuts B/D OC Valves Shifts FCV-498/499 to EDR vice OVBD	Y
RM-22	Waste Gas Vent	GM	55' Pipe Tunnel	Shuts OVBD (RCV-17)	Y
RM-23 *	S/G B/D Effluent	Scin.	100' GW	Same Actions as RM-19 above	N
RM-24/24R	Plant Vent Iodine	Scin.	85' FHB White Room	None	N
RM-25, 26	CR Vent Intake	Scin.	150' by 10% Dumps	CR Vent to Mode 4	Y
RM-28/28R	Plant Vent Part.	Beta Scin.	85' FHB White Room	None	Y
RM-29	Plant Vent Gross	Ion Ch.	140' by Escape Hatch	Alarms at OES	PAM2
RM-30, 31	Cont Hi Range	Ion Ch.	140' Containment	None	PAM2
RM-34	Area Monitor	Ion Ch.	85' White Room	None	PAM2
RM-41, 42, 43	Gas Decay Tanks	Ion Ch.	64' by Decay Tanks	None	N
RM-44 A&B	Containment Exhaust	Beta	100' GE	Containment Vent Isolation (CVI)	Y
RM-48	Post-LOCA Sample	Ion Ch.	85' Penetration Area	None	N
RM-51, 52, 53, 54	CR Press Intake	GM	140' Turb (Both Ends)	Transfers to other Unit	Y
RM-58, 59	Spent/New Fuel Area	GM	104' Fuel Hdlg Bldg	FHB Vent to Iodine Removal Mode	PAM2
0-RM-60, 61, 62, 63, 64, 65	TSC Area Monitors	GM	100' TSC Areas	None	N
0-RM-66	TSC Air Partic.	Scin.	TSC HVAC Room	None	N
0-RM-67	TSC Noble Gas	Scin.	TSC HVAC Room	None	N
0-RM-68	TSC Lab Air Partic.	Scin.	TSC HVAC Room	None	N
0-RM-69	TSC Lab Noble Gas	Scin.	TSC HVAC Room	None	N
RM-71, 72, 73, 74	Main Stream Line	GM	OC, line before MSIV	None	Y
0-RM-82, 83	TSC Iodine	Scin.	TSC HVAC Room	None	N
0-RM-90, 92	Radwaste Bldg Area	GM	Truck Bay, Laundry Rm	None	N

* Dual setpoint selectable via toggle switch at local indication panel

RO Question 24

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	APE 069 AK3.01	
	Importance Rating:	3.8	4.2

Proposed Question:

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

Why are the CFCUs operated in LOW instead of HIGH?

A. CCW cooling is isolated to the units.

B. To prevent overwhelming the drain system.

C. To prevent the units from tripping on overcurrent.

D. Design flow through the annular ring would be exceeded.

Proposed Answer:

C. To prevent the units from tripping on overcurrent.

Explanation:

A incorrect, cooling is not isolated to the units.

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

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

D incorrect, not a design concern.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Emergency Operations

Alignments
Obj 2

The CFCU system no longer has a normal or emergency damper alignment since the filters have been discarded.

- Normal damper alignment is now the same for emergency damper alignment.
- <u>Emergency operations call for CFCU fans to be in LOW speed to prevent the fans tripping on overcurrent due to the extremely dense atmosphere after a LOCA.</u>

Related EOPs
Obj 21

The emergency operating procedures with actions related to the CFCU system include:

EOP Procedure	Title	This EOP provides direction to address ...
E-1	Loss of Reactor or Secondary Coolant	• Appendix A-Blackout Emergency Loading of Vital Buses : Ensuring that the vital busses re-loading does not challenge the capability of restored power.
E-1.3	Transfer to Cold Leg Recirculation	• verifies a max of 3 CFCUs running if < 2 ASW Pps flowing thru 2 CCW HXs
ECA-1.1	Loss of Emergency Coolant Recirculation	• ensures that containment cooling is established. • ensures that if containment cooling is not adequate the CS pumps are started.
FR-Z.1	Response to High Containment Pressure	• ensures that containment cooling is established.
FR-Z.3	Response to Containment High Radiation Level	• ensures that containment cooling is established.

Actions outside the Control Room may require breaker operations or cooling alignments be made to ensure CFCU operations during emergencies.

STEP: Verify Containment Fan Coolers - RUNNING IN EMERGENCY MODE

PURPOSE: To ensure that the containment fan coolers are running in the emergency mode

BASIS:

This step instructs the operator to verify that all containment fan coolers are operating in the emergency mode. If the fan coolers are not running, the operator should manually start the fan coolers. The intent of this step is to provide containment heat removal and therefore reduce containment pressure. In the reference plant design, the containment fan coolers running in the emergency mode provide an alternate heat removal means to the containment spray pumps. The fan coolers operate at low speed under post accident conditions (emergency mode) to prevent the fan from stalling under the higher density accident atmospheric conditions.

The operator should also ensure that proper cooling is being provided to the fan coolers. Typically, cooling water is provided to the fan coolers from either the component cooling water or service water systems.

ACTIONS:

- o Determine if containment fan coolers are running in the emergency mode
- o Start fan coolers in emergency mode

INSTRUMENTATION:

Containment fan cooler indicator lights

CONTROL/EQUIPMENT:

Containment fan cooler switches

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

RO Question 25

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	APE 076 AK2.01	
	Importance Rating:	2.6	3.0

Proposed Question:

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

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

A. RM-15 counts.

B. RCS Argon-40.

C. RM-73 counts.

D. Steam Generator 1-2 tube leak flow.

Proposed Answer:

A. RM-15 counts.

Explanation:

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

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

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

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

Technical Reference(s): STG G4A, O-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 33808 - Interpret indications of core damage

Question Source:
Bank - INPO

Question History: Last NRC Exam: DCPP 10/02

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

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

Comments:

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	076.AK2.01	
	Importance Rating	2.6	3.0

APE: 076 High Reactor Coolant Activity
AK2 Knowledge of the interrelations between the High Reactor Coolant Activity and the following:
AK2.01 Process radiation monitors.

Proposed Question # 58 :

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

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

A RM-15 counts will increase.

B S/G 1-2 tube leak flow will increase.

C RCS Argon-40 will increase.

D RM-72 counts will increase.

Proposed Answer: A

Explanation:

Technical Reference(s): OP O-4 page 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8480 Demonstrate the ability to determine plant implications of each RMS radiation monitor indication.

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

2.3 The Air Ejector Off-gas monitors, RM-15 and 15R, are the most sensitive indicators of small changes in Steam Generator tube leakage. The Plant Process Computer (PPC) receives an input from RM-15 and 15R. The Plot or Archive function of the PPC can be used to detect small changes in the Air Ejector Off-gas count rate.

2.3.1 The High Alarm set point of RM-15 and 15R is set to the activity level that should be seen by the detector if the previously calculated RCS activity still exists and a total of 30 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the High Alarm setpoint may correspond to a leak rate greater than 30 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

2.3.2 The Alert alarm set point of RM-15 and 15R is set to the activity level that would be seen by the detector if the previously calculated RCS activity still exists and a total of 20 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the Alert Alarm setpoint may correspond to a leak rate greater than 20 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

2.3.3 Because the established monitor High Alarm and Alert Alarm setpoints are fixed values, they may correspond to a leak rate value that is different than 30 gpd or 20 gpd, depending upon changes in assumptions used to calculate the setpoints. The table below describes how some of these changes affect the setpoint:

Parameter	Relative Parameter Change	Alarm Setpoint Affect
RCS Noble Gas Concentration	↑ ↓	More Conservative Less Conservative
SJAE Exhaust Flowrate	↑ ↓	Less Conservative More Conservative

2.3.4 Primary to secondary leak rates can be inferred by comparing existing RM-15, 15R readings to the Alarm and Alert set points. The relationship of leak rate to activity level is linear as indicated activity increases.

2.3.5 Chemistry projects what the RM-15(R) count rate will be for a Steam Generator Tube leak of 30 gpd, 75 gpd and 150 gpd. This information is provided for reference in accordance with CY1.DC3. This projected value can be used by the Shift Foreman to roughly estimate Steam Generator leak rates.

2.3.6 Frequent analysis of RM-15(R) count rate and evaluation of the change in count rate with time is the best indicator of changes in the Steam Generator leak rate.

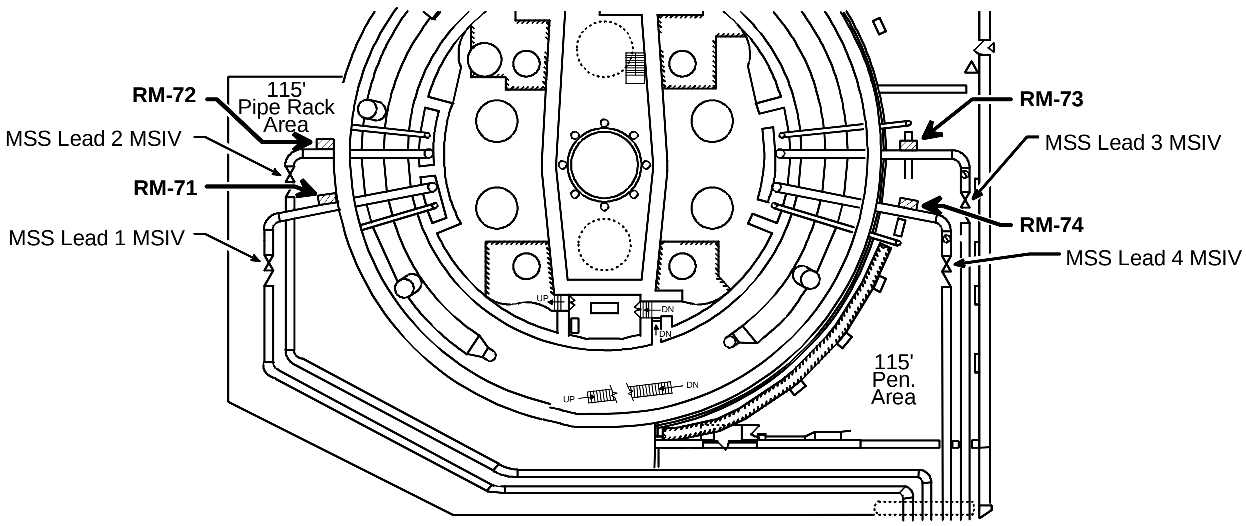
RM-71, 72, 73, and 74 Main Steam Line Monitors

Purpose
Obj 5

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

Location of detectors
Obj 6

The location of the detectors for the Main Steam Line monitors for Unit 1 is shown below. Unit 2 is a mirror image.



RMS-51

Power supplies

The power supplies for RM-71, 72, 73, and 74 are shown below.

Monitor	Panel
RM-71	PY14
RM-72	
RM-73	
RM-74	

Continued on next page

RO Question 26

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	EPE E02 EK1.1	
	Importance Rating:	3.2	3.8

Proposed Question:

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

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

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

A. Increase

B. Decrease

C. No change

D. Initial sharp decrease, then slowly stabilize

Proposed Answer:

C. No change

Explanation:

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

Technical Reference(s): STG B3

Proposed references to be provided to applicants during examination: None

Learning Objective: 6743 Explain PZR response during ECCS reduction sequence

Question Source:
Bank P-6157

Question History: Last NRC Exam

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

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

Comments:

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

1.00

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

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

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

A. No change

B. Initial increase, then slow continuous decrease

C. Increase

D. Decrease

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):
6743 Explain PZR response during ECCS reduction sequence

Reference Id: P-6157
Must appear: No
Status: Active
User Text: 6743.020304
User Number 1: 0000003.10
User Number 2: 0000004.00
Difficulty: 3.00
Time to complete: 2
Topic: E-1.1 - Pzr level response during ECCS flow reduction

Safety Injection Pumps, Continued

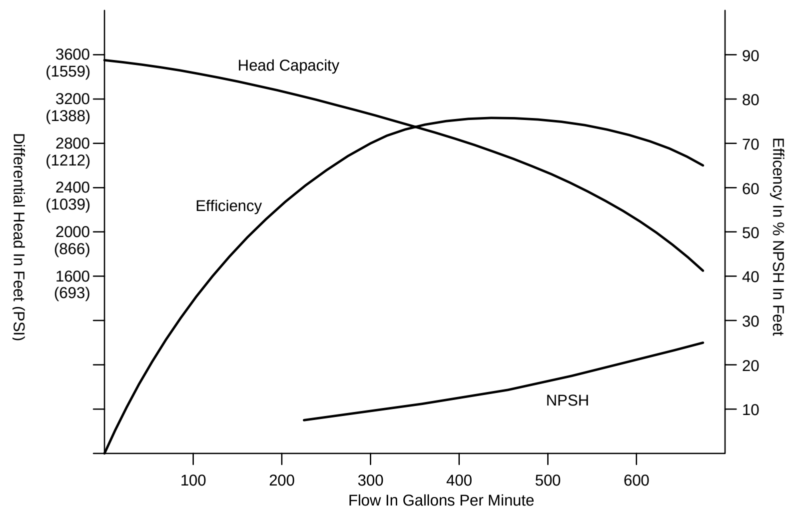
Physical description
Obj 10

Each unit has two 100% capacity pumps with the following characteristics.

Characteristic	Details
Type	10 stage, horizontal centrifugal pump.
Pump drive	4 kV 400 hp induction motor
Design Pressure	1700 psig
Design Temperature	300°F
Shutoff head	1559 psig
Design flow/head	425 gpm @ 2500 ft/1150 psid
Runout flow/head	675 gpm @ 1500 ft/675 psid[GRH35]

SIP operating characteristics
Obj 10

The diagram below illustrates the flow characteristics of the SIPs.



ECC-08

Continued on next page

RO Question 27

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	EPE E07 EA1.3	
	Importance Rating:	3.5	3.9

Proposed Question:

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

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

A. Subcooling restored.

B. ECCS pumps secured.

C. RCS temperature less than 200°F.

D. Core exit thermocouples stable or decreasing.

Proposed Answer:

A. Subcooling restored.

Explanation:

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

B incorrect. Procedure verifies pump injecting not secured.

C incorrect. No cooldown in the procedure.

D incorrect. No guidance regarding core exit thermocouples.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

*** *ISSUED FOR USE BY:* ________________ *DATE:* __________ *EXPIRES:* _________ ***

PACIFIC GAS AND ELECTRIC COMPANYNUMBER **EOP FR-C.3**
NUCLEAR POWER GENERATION **REVISION 9**
DIABLO CANYON POWER PLANT **PAGE 1 OF 7**
EMERGENCY OPERATING PROCEDURE **UNIT**

1

TITLE: **Response to Saturated Core Conditions**
EFFECTIVE DATE **10/26/01**

PROCEDURE CLASSIFICATION: QUALITY RELATED

1.0 SCOPE

1.1 This procedure provides actions to restore subcooled core cooling.

1.2 The major actions in EOP FR-C.3 are:

- o Establish ECCS flow to maintain a minimum RCS subcooling,
- o Check for open RCS vent paths.

2.0 VERIFY ENTRY CONDITION FOR EOP FR-C.3

2.1 EOP F-0.2, YELLOW Condition

RO Question 28

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	003 K5.02	
	Importance Rating:	2.8	3.2

Proposed Question:

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

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

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

A. Negative MTC

B. Trip with rods above the RIL

C. Natural circulation flow

D. Continued forced flow

Proposed Answer:

D. Continued forced flow

Explanation:

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

Technical Reference(s): STG A6

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank A-0157

Question History: Last NRC Exam: N/A

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

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

Comments:

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

ID: A-0157 Points: 1.00

With the plant initially at 100% power, a loss of all offsite power results in a reactor trip.

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

A. Continued forced flow

B. Natural circulation flow

C. ECCS flow

D. Sufficently low core thermal power

Answer: A

Associated objective(s):

State the purpose of RCP system and equipment

Question 1 Details

Question Type: Multiple Choice
Topic: Purpose of RCP flywheel
System ID: 6635
User ID: A-0157
Status: Active
Always select on test: No
Authorized for practice: No
Difficulty: 4.00
Time to Complete: 2
Point Value: 1.00
Cross Reference Number: 03

Motor, Continued

Part/Function table
Obj 6

The table below lists the functions of the parts of the RCP motor.

Part	Function
Flywheel	Increase pump coast down time, to extend the period of forced flow following a pump trip.
Upper/lower radial bearings	Babbitt-on-steel, pivoted-pad bearing assemblies providing maximum shaft stability by minimizing radial movement.
Thrust bearing	Double-acting Kingsbury thrust bearing minimizing axial movement. See Figure RCP-15 on page 2-26.
Upper brg oil cooler	Cools the oil in the upper oil reservoir. A lower bearing oil cooler cools the oil in the lower oil reservoir (not shown).
Oil lift pump and motor	Creates an initial film of oil between the thrust bearing shoes and thrust bearing runner on startup.
Anti-Reverse rotation device	Prevents reverse rotation when the pump is shut down and other RCPs are running, thus preventing excessive starting currents from starting the pump while it is rotating backwards
Space heaters (not-shown)	Prevents moisture condensation in the motor windings. A breaker B contact automatically energizes them when the pump is shut down.
Shaft coupling and spool piece	Connects the motor to the pump • An 18″ spool piece (not shown) is used below the coupling to allow access to and removal of the seal package without removing the motor.
Oil level indicators Upper (shown) Lower (not shown)	Provide local indication of the oil levels in the upper and lower oil reservoirs.
Stator	Insulated, rectangular, copper wire, impregnated with epoxy resin for insulation. Provides for the production of a rotating magnetic field.
Rotor	Copper alloy bars fitted into rotor slots, brazed to a copper connection ring at both ends. The rotating magnetic field in the stator induces an electrical charge in the rotor, developing a torque capable of driving the pump impeller.

Continued on next page

RO Question 29

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	004 A2.13	
	Importance Rating:	3.6	3.9

Proposed Question:

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

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

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

A. Manually initiate Safety Injection.

B. Stop the SI pump and realign charging suction back to the VCT.

C. Align the SI system for Cold Leg Recirculation using EOP E-1.3.

D. Immediately initiate makeup to the RWST.

Proposed Answer:

C. Align the SI system for Cold Leg Recirculation using EOP E-1.3.

Explanation:

C correct.
Per OP AP-24, caution:
CAUTION 1: IF RWST level decreases to less than 33%, the SI system should be aligned for cold leg recirculation using EOP E-1.3.

Technical Reference(s): OP AP-24, step 4

Proposed references to be provided to applicants during examination: None

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

Question Source:

Bank P-65994

Question History: Last NRC Exam

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

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

Comments:

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

1.00

During the performance of OP AP-24, while preparing to re-align one charging pump through charging injection, RWST level decreases to 33%. Neither Safety Injection nor Containment Isolation Phase A has actuated.

What action should be taken?

A. Align the SI system for cold leg recirculation using EOP E-1.3, Transfer to Cold Leg Recirculation.

B. Manually stop all ECCS pumps and go to EOP ECA-1.2, LOCA Outside Containment.

C. Manually initiate SI and go to EOP E-1.3, Transfer to Cold Leg Recirculation.

D. Manually stop all ECCS pumps and go to EOP ECA-1.1, Loss of Emergency Coolant Circulation.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3477	Describe the major actions of abnormal operating procedures

Reference Id: P-65994
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 3.00
Time to complete: 2
Topic: AP-24 - Action to take if RWST level decreases to 33%

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

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

CAUTION 2: If RCS leakage is isolated, steps 32 (Page 17) through 36 should be immediately performed to terminate SI flow.

****

4. REALIGN One Charging Pump Through Charging Injection

a. Charging Pps – ONLY One Running

RNO: a. Stop charging pump to establish only one pump running.

b. OPEN 8805A AND 8805B

c. OPEN Charging Injection Valves:
- 8803A AND 8803B
- 8801A AND 8801B

d. CLOSE 8107 AND 8108

e. Verify FCV-128 – Fully OPEN

5. EVACUATE Non-essential Personnel In Containment:

a. Activate the containment evacuation alarm

b. Notify Plant Personnel of LOCA via the Public Address system

6. ACTUATE Containment Isolation Phase A:

a. Actuate Phase A

b. Check Phase A portion of Monitor Light Box B:
- Red Activated Light – ON
- White Status Lights – OFF

RNO: b. Manually Close the Phase A Isol vlvs with White Status Lights - ON

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. CHECK If RCPs Must Be Stopped: • RCP No. 1 Seal differential pressure – GREATER THAN 255 PSID AND • RCP No. 1 Seal Leakoff flow – INSIDE APPENDIX B guidance	Stop affected RCP(s).
8. DISPATCH Personnel To Locally Restore Power To Locked Out ECCS Equipment:	
a. Install UC fuses on SI Pump with its breaker racked in	
b. Make all other Charging/SI pumps available	
9. CHECK If One Safety Injection Pump Should Be Started:	
a. Check the following • PZR Level – LESS THAN 12% [36%] OR • RCS Subcooling based on core exit T/Cs – LESS THAN 20ºF	GO TO step 13 (Page 6)
b. Establish flow from one Safety Injection Pump: 1) Verify SI system aligned for Cold Leg injection 2) Start one SI Pp	

RO Question 30

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	005 K5.05	
	Importance Rating:	2.7	3.1

Proposed Question:

GIVEN:
- The plant is in MODE 5, solid
- One train of RHR in service
- Letdown Pressure Control Valve, PCV-135, is in AUTO
- RHR Letdown Flow is being maintained via CVCS HCV-133.
- Charging pump 12 is in service.

The running RHR pump trips.

Which of the following describes the RCS pressure response?

A. RCS pressure INCREASES due to HCV-133 closing and charging is in service.

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

C. RCS pressure DECREASES due to PCV-135 opening.

D. RCS pressure DECREASES due to the loss of RHR pump discharge pressure.

Proposed Answer:

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

Explanation:

A incorrect. HCV-133 will not move.

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

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

C incorrect. Operation is opposite of what occurs.

D incorrect. Pressure increases when the pump trips.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

9.2.12 The plant is particularly vulnerable to exceeding the PZR Heatup and Cooldown rates while in Mode 5, with the RCS at cooler temp., and a bubble in the PZR. The Heatup limit is 100°F per hour, and the cooldown rates are 200°F per ECG 7.5.

a. The cooldown is self explanatory, however the Heatup can catch you off guard. If there is plant transient which causes the plant to cool greater than 100°F, caution should be used when trying to restore temp.

b. Cooldown RATE shall be assured by limiting the cooldown rate to 200°F per hour. The cooldown should be uniform over time thus avoiding large step changes.

c. If T_{COLD} is 200°F less than the PZR liquid space temperature, do not exceed a PZR fill rate (net charging rate) of 35 GPM. Energize all PZR heater banks during the fill process. Limit any PZR fill rate so as not to exceed a 200°F per hour cooldown.[T31586, T32450]

d. Pressurizer cooldown and heatup rates should be monitored closely, during any evolution planned or unplanned affecting makeup or rejection.

9.2.13 <u>When operating in a solid condition with RCS pressure being maintained by CVCS-1-PCV-135, changes to the flow rate in the RHR loops or starting and stopping the RHR pumps will result in changes in RCS pressure. CVCS-1-PCV-135 should be operated in MANUAL and adjust as necessary to minimize RCS pressure transient</u>.

9.2.14 If the RCS is to be depressurized with the head on for an extended period, the RCS is susceptible to gas accumulation in the vessel head. Actions which may be taken to minimize/detect gas accumulation include:[T35812]

a. Verify at least one RVLIS train in service and monitor for decreasing vessel level.

b. Monitor Pressurizer level for increases that appear larger than can be explained by a charging/letdown mismatch.

c. Reduce VCT Nitrogen pressure, or vent to atmosphere.

d. If gas accumulation is suspected, contact system engineering and maintenance to setup a temporary vent path to remove any accumulated gas.

9.2.15 Use of N/A shall be in accordance with AD2.ID1, "Procedure Use and Adherence," with the following additions. Any N/A for non-conditional steps must be explained sufficiently in the Remarks Section that this procedure will stand alone, and must be initialed by the SFM to indicate concurrence with the action. As stated in AD2.ID1, conditional steps that are N/A'd do not require explanations or specific SFM approval.

DATE / TIME / INITIALS

9.3.4 Place the RCS in a SOLID condition as follows:[T31830]

NOTE: PCV-135 can be operated in manual or automatic.

a. Exercise CVCS-1-PCV-135 and locally verify that it strokes smoothly through its entire valve travel. (A partial STP V-2I2 in conjunction with stroke of PCV-135 including TCV-149 and HCV-133 is recommended) ____/____/____

b. Energize all pressurizer heater banks while adding water to the pressurizer. ____/____/____

c. Increase net charging to fill the PZR at a rate less than 35 gpm. [T31586, T32450] ____/____/____

d. Use sprays as necessary, to control RCS pressure as PZR level increases. ____/____/____

e. As the bubble collapses, as indicated by inability to control pressurizer pressure with spray, reduce charging and adjust the position of CVCS-1-PCV-135 as necessary to control RCS pressure. ____/____/____

f. When PZR is solid, turn OFF all PZR heaters and CBC Tag to SFM using Attachment 12.3, Section V. ____/____/____

g. Close the PZR steam space purge to VCT by closing sample line valves NSS-1-9354A and B. ____/____/____

h. With CVCS-1-PCV-135 in MANUAL, isolate normal letdown one orifice valve at a time, (CVCS-1-8149C, 8149B, and 8149A), and minimize RCS pressure transients. ____/____/____

NOTE: RHR Letdown Flow is maintained via CVCS-1-HCV-133.[T35757]

i. Close LCV-459, LCV-460 and CVCS-1-8152 to prevent crud entrapment during forced oxygenation. ____/____/____

RO Question 31

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	005 A4.03	
	Importance Rating:	2.8	2.7

Proposed Question:

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

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

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

NOTE:
- HCV-670 – Bypass Flow Control Valve
- HCV-637 – Flow Control Valve for Heat Exchanger 12
- HCV-638 - Flow Control Valve for Heat Exchanger 11

A. Close down on HCV-670. HCV-637 and HCV-638 will open to maintain a constant flow.

B. Open both HCV-637 and HCV-638. HCV-670 will close down to maintain a constant flow.

C. Open either HCV-637 or HCV-638 while opening HCV-670.

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

Proposed Answer:

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

Explanation:

A incorrect. Valves must be manually operated.

B incorrect. HCV-670 does not respond automatically.

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
INPO

Question History: Last NRC Exam: Seabrook 05/03

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

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

Comments:

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

ID: 24643 Cog level: 2

Plant conditions are as follows:

$RCS temperature is 300 F
$RCS pressure is 340 psig
$Train A Residual Heat Removal (RHR) is in service with flow set at 3500 gpm; Train B RHR is in ECCS standby mode
$The current cooldown rate is 12 F/hr.
$The US orders the cooldown rate increased from 12 F/hr to 40 F/hr over 10 minutes while maintaining RHR flow relatively constant.

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

A. Throttles the RHR heat exchanger bypass valve, RHR-FCV-618, in the closed direction. This causes less water to flow through the RHR heat exchanger bypass line. Flow is automatically increased through the RHR heat exchanger to maintain the combined flow rate constant at 3500 gpm.

B. Throttle open RHR-FCV-610 RHR pump A mini-flow. The RHR heat exchanger outlet and bypass valves require no throttling since RHR system flow rate remains constant at 3500 gpm.

C. Throttles the RHR heat exchanger outlet valve, RHR-HCV-606, in the closed direction such that RHR system water will spend more time in the RHR heat exchanger to be cooled further by PCCW. Flow is automatically increased through RHR-FCV-618 to maintain the combined flow rate constant at 3500 gpm.

D. Throttles the RHR heat exchanger outlet valve, RHR-HCV-606, in the open direction. This causes more RHR system water to flow through the RHR heat exchanger. Flow is automatically decreased by RHR-FCV-618 to maintain the combined flow rate constant at 3500 gpm.

D - correct - the operator manipulates the outlet valve to send more or less water through the Hx to be cooled. The bypass valve modulates to maintain 3500 gpm total coming from the Hx and bypass line.

A - incorrect - will reduce CD rate
B - incorrect - PCCW is not manipulated by procedure;
C - incorrect - bypass valve is modulated, the Hx outlet is not.

6.2.32 Crack open the RHR heat exchanger bypass flow control valve HCV-670 while observing temperature recorder TR-636/TR-648.

a. Verify that RHR temperature is slowly increasing to equal the RCS temperature.

6.2.33 When the RHR and RCS temperatures are approximately equal, slowly open HCV-670 until the flow to the RCS cold legs is approximately 3000 gpm.

NOTE: The Plant Cooldown may be performed using one RHR Pump with two RHR Heat Exchangers OR two RHR Pumps with two RHR Heat Exchangers. (See P&L 5.3 for flow limitations)

6.2.34 Start the second RHR Pump, if desired.

a. Verify FCV-641A (FCV-641B), minimum flow recirculation valve, OPENS and then CLOSES.

6.2.35 When the system temperatures have stabilized, open HCV-670 until the total flow to the RCS cold legs is approximately 3000 to 5000 gpm as indicated on FI-970 and FI-971.

6.2.36 Commence the primary system cool down process per OP L-5.

NOTE: CCW temperatures are available for trending on the PPC as points T0611A and T0612A.

a. During the cooldown, monitor CCW return line temperature from the RHR Heat Exchanger. Temperature shall not exceed the maximum qualified piping temperature of 145°F.

b. Gradually open the RHR heat exchanger flow control valves HCV-637 and 638 while closing off the bypass flow control valve HCV-670 to maintain a cool down rate consistent with OP L-5.

NOTE: It may be necessary to place PCV-135 in MANUAL to achieve an adequate letdown flow rate due to the low RCS pressure.

6.2.37 Initiate letdown flow to the CVCS by slowly opening letdown flow control valve HCV-133.

6.2.38 During the cool down process, proportion the flow through the RHR heat exchangers to ensure compliance within the following limits:

a. Total RHR flow to the RCS cold legs should be limited to approximately 5000 gpm.

b. The temperature of the component cooling water leaving the CCW heat exchangers should not be allowed to exceed 120°F.

6.2.39 The RHR system can now be used to control the RCS temperature and letdown for operation in Mode 4 and 5.

RO Question 32

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	006 K6.02	
	Importance Rating:	3.4	3.9

Proposed Question:

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

A. Discharge of the accumulators occurs at a later time and will interfere with natural circulation core cooling.

B. Discharge of the accumulators occurs at a later time and could initiate a rapid RCS cooldown and re-initiate break flow.

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

D. Insufficient water will be available in the recirculation sump to provide long-term core cooling ability during a post-LOCA recovery.

Proposed Answer:

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

Explanation:

Only C correct.
B3.5.1
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. <u>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.</u>

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

Technical Reference(s): Tech Spec Bases B3.5.1, LMCDFRC

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-0545

Question History: Last NRC Exam

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

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

Comments:

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

Difficulty taken from bank question.

1.00

A large-break LOCA has occurred. Two of the ECCS accumulators fail to discharge, as indicated by constant accumulator level and pressure readings.   The MAJOR safety concern is that

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

B. discharge of the accumulators at a later time will interfere with natural circulation core cooling.

C. discharge of the accumulators at a later time could initiate a rapid RCS cooldown and re-initiate break flow.

D. insufficient water will be available in the recirculation sump to provide long-term core cooling ability during a post-LOCA recovery.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

5459	Explain why ECCS Accumulators are isolated for various plant conditions.

Reference Id: P-0545
Must appear: No
Status: Active
User Text: 5459.020065
User Number 1: 0000003.30
User Number 2: 0000004.00
Difficulty: 3.00
Time to complete: 2
Topic: ECCS - Safety concern with accumulators not inj per design.

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	<u>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.</u> 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)

Overview, Continued

ICC, continued
Obj 15,51,52

Smaller LOCAs, which slowly depressurize the RCS, rely on the HPSI system to refill the RCS. Without this system, a small LOCA would continue to remove RCS coolant, eventually uncovering the core.

The core remains uncovered until RCS pressure drops below the Accumulator setpoint. An ICC condition may develop during this time. Larger LOCAs, which depressurize the RCS quickly, rely on the Accumulators and low head SI to refill the RCS. Without these systems, the core would quickly uncover and remain uncovered until the HPSI system could refill the RCS. An ICC condition may develop during this refilling period. The probability of a larger LOCA is lower than the probability of a small LOCA. When combined with a probability of losing both Accumulators and the low head SI, the probability of a large LOCA inadequate core cooling event is very small indeed.

The time required for the symptoms of ICC to develop is dependent upon the RCS volume, break size and location, core heat generation rate and the type and amount of SI available. A small LOCA without high-head SI at a large 1000 MWe four loop plant near the beginning of life will take a considerably longer period of time to reach an ICC condition than a similar size break at a smaller two loop plant near the end of life.

In any case, either the failure of, or the intentional operator action to turn off the SI system is required to obtain an ICC condition. The SI system, designed to refill the RCS following a LOCA, provides both high pressure and low pressure pumped flow to the RCS. A total failure of this system is considered unrealistic since multiple failures would be required. Manual action in the Control Room to prematurely terminate SI is a more probable cause for loss of SI. This is what happened at TMI.

Continued on next page

RO Question 33

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	007 K4.01	
	Importance Rating:	2.6	2.9

Proposed Question:

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

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

A. Automatic opening of the Primary Water system supply valve.

B. Manually opening the Primary Water system supply valve.

C. Venting the PRT to the Waste Gas Header.

D. Draining the PRT to the RCDT.

Proposed Answer:

B. Manually opening the Primary Water system supply valve.

Explanation:

A incorrect. The valve does not open automatically.

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

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

Technical Reference(s): AR PK05-25, PK 05-25, PRT Press/Lvl/Temp

Proposed references to be provided to applicants during examination: none

Learning Objective: 4950 - Explain the operation of PRT system

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

PRT
PRESS/LVL
TEMP

02/14/06
Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
318	Pzr Relief Tk Temp Hi	TC471	> 130°F	2.1
367	Pzr Relief Tk Lvl Hi	LC470A	> 89%	2.1
545	Pzr Relief Tk Press Hi and Vent Hdr Isol	PC472X	> 10 psig	2.1
1394	Pzr Relief Tk Lvl Lo	LC470B	< 56%	2.1

2. OPERATOR ACTIONS

2.1 General Actions **(All Inputs)**

2.1.1 Check the following indications to determine alarm cause (VB2):

- PI-472, Pzr Relief Tk Press [ ]
- LI-470, Pzr Relief Tk Level [ ]
- TI-471, Pzr Relief Tk Temp [ ]

2.1.2 IF low PRT level exists,
THEN GO TO Section 2.2. [ ]

2.1.3 IMPLEMENT OP AP-1, "Excessive RCS Leakage", to aid in identifying source and magnitude of leakage. [ ]

2.1.4 IF source of leakage is a safety valve or PORV,
THEN REFER TO the following for operability concerns:

- TS 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)" [ ]
- TS 3.4.13, "RCS Operational LEAKAGE" [ ]

2.1.5 IF high PRT level exists,
THEN perform the following:

a. Verify CLOSED RCS-1-8030, PRT Pri Wtr Sply (VB2). [ ]

> **CAUTION**: RCS-1-8031, PRT Drain to RCDT, is a slow-closing valve. If closing is not initiated at 60% RCDT level, the RCDT may overflow before the valve is fully closed. Overfilling the RCDT will fill the LI reference legs, which will then require draining by maintenance.

b. Lower PRT level to approximately 84% by draining to the RCDT as follows:

1. Establish communications with the Aux Senior watch. [ ]

2. Direct the Aux Senior watch to perform the following:

   a) IF RCDT Level reaches 60% as read on LI-188, Reactor Clnt Drain Tank (Aux Control Board),
   THEN immediately inform the Control Room. [ ]

   b) Run BOTH RCDT pumps to keep up with the drain rate by holding BOTH pump start switches in the "ON" position. [ ]

3. IF at any time RCDT level reaches 60%,
THEN perform the following:

   a) Immediately CLOSE RCS-1-8031, PRT Drain to RCDT (VB2). [ ]

   b) Allow the RCDT level to lower to approximately to 34% as read on LI-188, Reactor Clnt Drain Tank (Aux Control Board). [ ]

   c) IF desired to continue to lower PRT Level,
   THEN OPEN RCS-1-8031. [ ]

4. OPEN RCS-1-8031 to begin draining. [ ]

5. WHEN PRT level reaches 84%,
THEN CLOSE RCS-1-8031. [ ]

2.1.6 <u>IF</u> high PRT temperature exists,
<u>THEN</u> reduce PRT temperature to 120°F by performing the following:

a. OPEN RCS-1-8030, PRT Pri Wtr Sply (VB2). [ ]

b. <u>WHEN</u> TI-471, Pzr Relief Tk Temp, is 120°F or lower,
<u>THEN</u> CLOSE RCS-1-8030 (VB2). [ ]

c. <u>IF</u> lowering PRT level is desired,
<u>THEN</u> GO TO step 2.1.5 to drain the PRT to the RCDT. [ ]

2.1.7 <u>IF</u> high PRT pressure exists,
<u>THEN</u> perform the following:

a. Verify CLOSED RCS-1-8045, PRT N_2 Supply Isol (VB2). [ ]

<u>NOTE</u>:	RCS-1-PCV-472, PRT Vent to Vent Hdr, is NOT available when PRT pressure is 10 psig or above.

b. <u>IF</u> PRT pressure is 10 psig or above,
<u>THEN</u> lower PRT pressure to slightly less than 10 psig by performing ONE of the following:

- Lower PRT level PER step 2.1.5. [ ]
- Direct an Operator to remove blank flange from RCS-1-8048, PRT Vent Vlv To Contmt Atmosphere, and bleed pressure off slowly (Ctmt 91 ft el, PRT). [ ]

c. <u>WHEN</u> PRT pressure is less than 10 psig,
<u>THEN</u> perform the following:

1. Inform the Aux Building watch of the intention to vent the PRT to the Waste Gas Header. [ ]

2. Verify Waste Gas Header available. [ ]

3. OPEN PCV-472, PRT Vent to Vent Hdr, to continue to lower pressure to approximately 3 psig or until pressure otherwise stabilizes (VB2). [ ]

4. CLOSE PCV-472. [ ]

<u>NOTE</u>:	A PRT pressure increase without increases in temperature or level indicates a possible failure of RCS-1-PCV-8035, PRT N_2 Supply.

d. Direct maintenance to troubleshoot. [ ]

e. IF there is a slow increase in PRT pressure, THEN maintain PRT pressure between 3 and 10 psig by performing the following:

1. Verify CLOSED RCS-1-8045, PRT N_2 Supply Isol (VB2). [ ]

2. OPEN PCV-472 to lower pressure to approximately 3 psig or until pressure otherwise stabilizes. [ ]

3. CLOSE PCV-472. [ ]

f. IF the nitrogen regulator is working properly, THEN OPEN RCS-1-8045. [ ]

2.1.8 Probable Causes

a. High PRT pressure, temperature or level:
- RCS-1-PCV-474, 455C or 456, Pressurizer power relief valves, leaking or lifting
- RCS-1-8010A, B or C, Pressurizer safety valves, leaking or lifting
- Malfunction of Pzr pressure control system

b. High PRT pressure or level:

1. Relief valve (from outside containment) lifting or leaking:

• CS-1-RV-9007A, Containment Spray Hdr	260 psig
• SI-1-RV-8858, SI Pump Suct Hdr	220 psig
• SI-1-RV-8856B, RHR Ht Exch 1-2	600 psig
• CVCS-1-RV-8125, Charging Pps Suct Hdr	220 psig
• SI-1-RV-8853B, SI Pp 1-2 Disch Hdr	1750 psig
• SI-1-RV-8853A, SI Pp 1-1 Disch Hdr	1750 psig
• SI-1-RV-8851, SI Pps Disch Hdr	1750 psig
• SI-1-RV-8856A, RHR Ht Exchg 1-1	600 psig
• CS-1-RV-9007B, Contmt Spray Hdr	260 psig

2. Relief Valve (from inside containment) lifting or leaking:

• CVCS-1-RV-8121, Seal Wtr Ret Hdr	150 psig
• CVCS-1-RV-8117, CVCS Letdown Hdr	600 psig
• RHR-1-RV-8707, RHR Pp Suct Hdr	450 psig
• SI-1-RV-8856A, B, RHR Pp Inj Loops 1 & 2	600 psig
• RHR-1-RV-8708, RHR to RCS Hot Legs Hdr	600 psig

c. High PRT temperature or level due to valve stem leak-off from:
- RCS-1-8000A, B, C, Pzr PORV Iso
- RCS-1-PCV-455A, B, Pzr Spray
- RHR-1-8701 and 8702, Loop 4 to RHR system
- RCS-1-8033A, B, C, D, Pzr Spray Iso
- RCS-1-8076, CVCS Ltdn Iso
- CVCS-1-8143, Excess Ltdn Divert
- CVCS-1-HCV-123, Excess Ltdn Flow Control

d. High PRT level:
- Pressurizer safety valve loop seal drain header valve(s) leaking
- RCS-1-8030, PRT Pri Wtr Sply, leaking

e. High PRT pressure due to RCS-1-PCV-8035, PRT N2 Supply, malfunction

2.2 Low PRT Level Alarm

CAUTION: Primary Water fill rate to the PRT is quicker than the venting rate. Do NOT exceed 10 psig in the PRT. RCS-1-PCV-472, PRT Vent to Vent Hdr, is not available to depressurize the RCDT when PRT pressure is 10 psig or above.

2.2.1 Monitor PI-472, PRT Press, during filling (VB2). [ ]

2.2.2 IF at any time PRT pressure approaches 10 psig,
THEN perform the following:

a. CLOSE RCS-1-8030, PRT Pri Wtr Sply. [ ]

b. WHEN PRT pressure lowers sufficiently,
THEN OPEN RCS-1-8030. [ ]

2.2.3 OPEN RCS-1-8030, PRT Pri Wtr Sply. [ ]

2.2.4 WHEN PRT level is 84%,
THEN CLOSE RCS-1-8030. [ ]

2.2.5 Verify CLOSED RCS-1-8031, PRT Drain to RCDT (VB2). [ ]

2.2.6 IF PRT is still draining to the RCDT,
THEN cycle RCS-1-8031 OPEN and CLOSED. [ ]

2.2.7 IF PRT level continues to drop,
THEN direct an Operator to check PRT and instrument taps for leaks. [ ]

2.2.8 Probable Causes

- RCS-1-8031, PRT Drain to RCDT, leaking by
- PRT or level tap leaking below the water line

3. AUTOMATIC ACTIONS

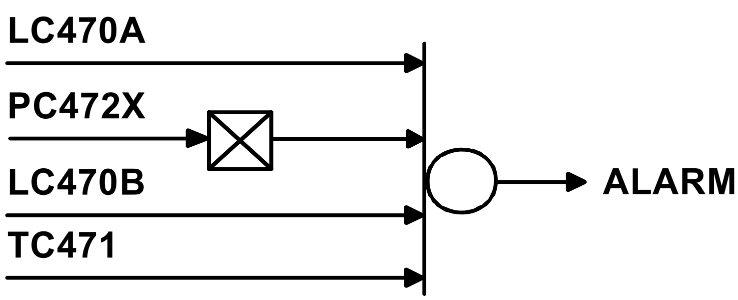
3.1 Vent header isolation at 10 psig (RCS-1-PCV-472, PRT Vent to Vent Hdr, closes)

4. REFERENCES

4.1 501127, "Electrical Schematic Diagram - Main Annunciator" (Electrical Drawing Section 8)

4.2 OP A-4B:IV, "Pressurizer Relief Tank - Normal Operation"

5. LOGIC DIAGRAM



RO Question 34

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	008 A1.02	
	Importance Rating:	2.9	3.1

Proposed Question:

The plant is at 35% power.

Which of the following conditions requires the operator to trip the reactor?

A. CCW temperature of 125°F with no ASW pumps running.

B. PORV-455 fails open and the block valve is open.

C. Turbine vacuum of 8 inches absolute.

D. Intermediate Range N35 fails high.

Proposed Answer:

A. CCW temperature of 125°F with no ASW pumps running.

Explanation:

A correct. If CCW temperature exceeds 120F, a reactor trip is required, per AP-11.
B incorrect. Action is to close the block valve.
C incorrect. A turbine trip would be required, but a reactor trip below P-9 is not required.
D incorrect. IR High Flux trip is blocked above 10%.

Technical Reference(s): AP-11.

Proposed references to be provided to applicants during examination: None

Learning Objective: 7927 - Explain the conditions requiring reactor trip

Question Source:
New

Question History: Last NRC Exam: N /A

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

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

Comments:

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

SECTION A: LOSS OF A CCW PUMP/HIGH CCW SYSTEM TEMP (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. VERIFY That ASW Is Supplying The In Service CCW Hx	IMPLEMENT OP AP-10, LOSS OF AUXILIARY SALTWATER, unless it has been determined that ASW can not be recovered due to equipment damage.
• At least one ASW Pump in service • ASW Pp Amps – Normal	IF ASW can not be recovered or CCW temperature is > 120°F, THEN: a. TRIP the reactor b. TRIP all RCPs c. GO TO EOP E-O, REACTOR TRIP or SAFETY INJECTION while IMPLEMENTING the following: • Appendix B to secure CCW loads • Appendix C to provide backup cooling to Charging Pumps. • Appendix D for Loss of Ultimate Heat Sink Instructions.
5. VERIFY That The CCW Hx Is Not Clogged PK01-01 NOT in alarm	IMPLEMENT OP AP-10, LOSS OF AUXILIARY SALTWATER.
6. **PLACE A Second Train of ASW/CCW In Service Per OP E-5:II Section 6.1**	Place two CCW Hx in parallel on one ASW pump per OP E-5:II Section 6.2.
7. CHECK CCW TEMPERATURES • DECREASING AND • WITHIN TABLE 1 VALUES	Reduce Heat Loads per Appendix B as required.
8. RETURN To Procedure and Step In Effect	

- END -

RO Question 35

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_1___	_____
	Group:	_2___	_____
	K/A:	008 A3.10	
	Importance Rating:	2.9	3.0

Proposed Question:

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

A reactor trip and Safety Injection occurs.

Which CCW pumps will be running?

A. All 3 CCW pumps will be running.

B. Only 1-1 and 1-2.

C. Only 1-1 and 1-3.

D. Only 1-1.

Proposed Answer:

A. All 3 CCW pumps will be running.

Explanation:

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

B incorrect. The 1-3 pump will start.

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

CCW Pumps, Continued

Logic
Obj 8, 25, 26

The logic associated with CCW pump operation is described in the following tables:

With the breaker transfer switch selected to CONT RM:

If the VB1 Auto/Manual switch is in ...	Then the pump will ...
AUTO	Start on: • low vital header pressure 69.6 psig (with no SIS or Transfer to Diesel) NOTE: The start will be delayed until lube oil pressure reaches 6 psig.
MANUAL	Manual start and stop (overcurrent reset also permitted) NOTE: The start will be delayed until lube oil pressure reaches 6 psig.

Pump starts independent of Auto/Manual selection	
When the following occurs...	**the pump will start...**
4 kV Auto Transfer to Diesel (assumes no SI signal)	after a time delay of 5 sec.
SI signal	after a time delay of ≅ 18 seconds (14 sec for 1-3). (If bus voltage is lost, the time delay will start upon voltage restoration)

With the breaker transfer switch selected to HSD PNL:

If the HSDP Auto/Manual switch is in ...	Then the pump will ...
AUTO	Start on: • 4 kV Auto Transfer to Diesel (no SI signal) • SI signal • Manual NOTE: The LOW header pressure auto-starts are disabled.
MANUAL	Manual start and stop. NOTE: All auto-starts are disabled. The start will be delayed until lube oil pressure reaches 6 psig.

Continued on next page

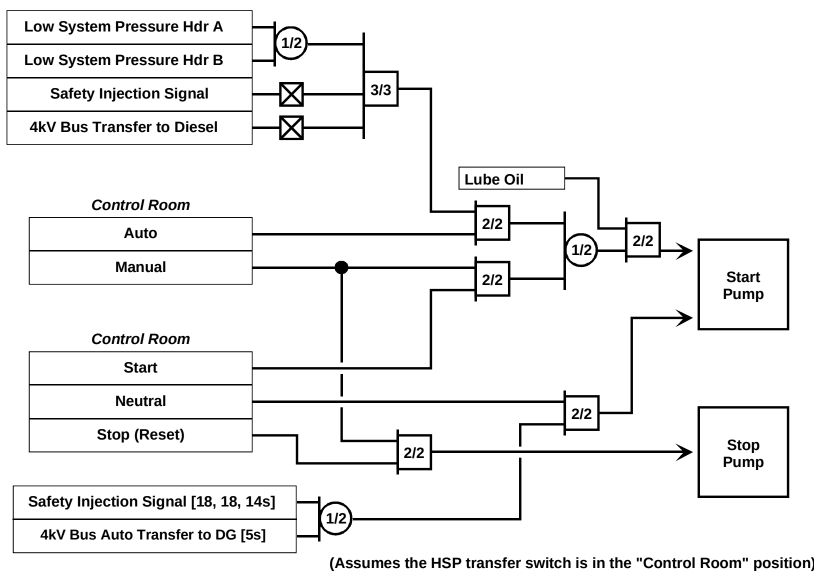
CCW Pumps, Continued

Logic (continued)

When controlling CCW pumps from the HSDP:
- all automatic trips continue to function normally,
- the manual trip at the HSDP and the switchgear still operate normally, and
- all CCW pump lights in the Control Room, with the exception of the white light, are extinguished.

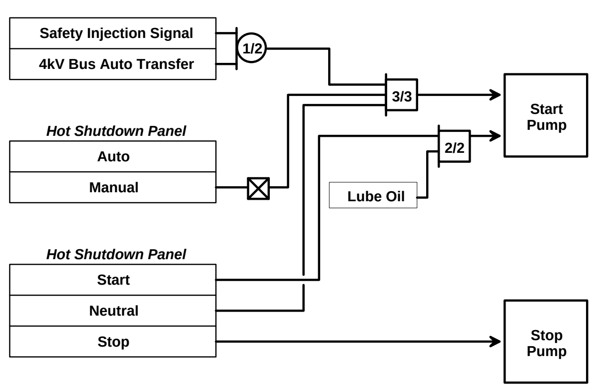
Logic diagrams for pump starts and stops from the control schemes
Obj 8, 9, 26

The following logic diagram assumes the transfer switch at the breaker in the CONT RM position:



CCW-03

The following logic diagram assumes the transfer switch at the breaker in the HSD PNL position:



CCW-04

Continued on next page

RO Question 36

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	010 A2.01	
	Importance Rating:	3.3	3.6

Proposed Question:

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

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

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

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

A. None, the heaters energized when SI was reset.

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

C. Dispatch a Nuclear Operator to locally reset the breakers on Bus F and G.

D. Dispatch a Nuclear Operator to locally reset the breakers on Bus G and H.

Proposed Answer:

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

Explanation:

A incorrect. SI opens the breakers.

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

C incorrect. The bus power supplies for the heaters is G and H.

D incorrect. SI opens the breakers, but they are reclosed by cycling the control switch.

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

Proposed references to be provided to applicants during examination: None

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

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

Question Source:
New

Question History: Last NRC Exam

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

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

Comments:

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

ACTION/EXPECTED RESPONSE **RESPONSE NOT OBTAINED**

35. CONTROL RCS Pressure And Charging Flow To Minimize RCS-To-Secondary Leakage: (Continued)

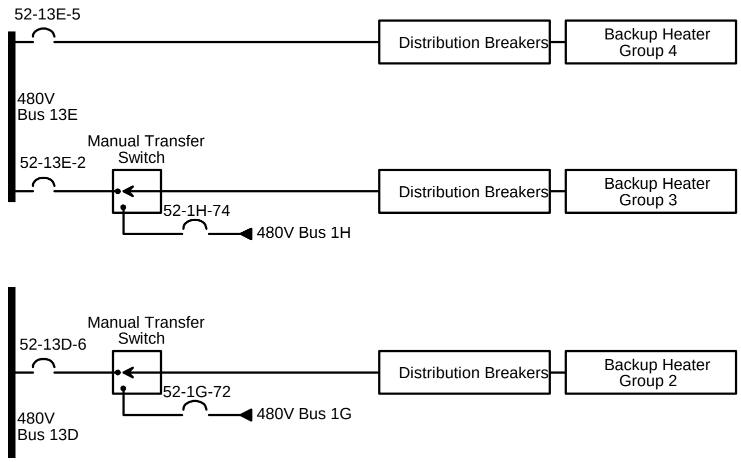
c. Decision table:

	RUPTURED S/G NR LVL INCREASING	**RUPTURED S/G NR LVL DECREASING**	**RUPTURED S/G NR LVL OFF SCALE HIGH**
PZR LEVEL LESS THAN 17%[47%]	• Increase Charging Flow AND • Depressurize RCS	Increase Charging Flow	• Increase Charging Flow AND • Maintain RCS And Ruptured S/Gs Pressures Equal
PZR LEVEL BETWEEN 17%[47%] AND 50%	Depressurize RCS	Verify PZR Heaters - ON	Maintain RCS And Ruptured S/Gs Pressures Equal
PZR LEVEL BETWEEN 50% AND 74%[66%]	• Depressurize RCS AND • Decrease Charging Flow	Verify PZR Heaters - ON	Maintain RCS And Ruptured S/Gs Pressures Equal
PZR LEVEL GREATER THAN 74%[66%]	Decrease Charging Flow	Verify PZR Heaters - ON	Maintain RCS And Ruptured S/Gs Pressures Equal

Backup Heaters, Continued

Power distribution diagram

The diagram below illustrates the electrical power distribution to Backup Heater groups.



PZR-24

Each distribution breaker supplies three heaters, one per phase.

Power supplies
Obj 14

The table below lists the power supplies to the Backup Heaters.

Heater group	Power Supply	
Group 2	**Normal**	480V bus D
	Emergency	480V bus G
Group 3	**Normal**	480V bus E
	Emergency	480V bus H
Group 4	**Normal**	480V bus E

Continued on next page

Backup Heater HSDP Controls, Continued

Backup heater automatic actions
Obj 12

The table below describes the conditions under which the Backup Heaters are energized.

Switch Positions	Condition
• Transfer Switch in CONT RM • Normal Power Aligned	• Backup Heater Control switch on CC-1 is taken to ON and released to AUTO • Pressure deviation-low from setpoint, MPC output less than or equal to 9.4% (-25 psi), when the Backup Heater Control switch is in AUTO-after-OFF • Level deviation-high from setpoint, +5% of span (LC-459E), when the Backup Heater Control switch is in AUTO-after-OFF
Group 2 • Transfer switch in CONT RM position • Vital power aligned	Backup Heater Control switch on CC-1 to ON
Group 3 Vital power aligned	Backup Heater Control switch on CC-1 to ON
Group 2 & 4 • Control Transfer Relay Cutout Switch in CUTIN • Transfer switch in LOCAL (HSDP)	Backup Heater Local Control switch (HSDP) to ON (note: for group 2 only, normal power must be aligned)

Continued on next page

Backup Heater HSDP Controls, Continued

Backup heater automatic actions (continued)

The table below describes the conditions under which the Backup Heaters are deenergized.

Switch Positions	Condition
• Transfer switch in CONT RM • Normal power aligned	• Backup Heater Control switch on CC-1 is placed in OFF. • MPC output ≥ 15.6% when the Backup Heater Control switch is in AUTO-after-OFF. • Level deviation-high reduced to less than +5% of span (LC-459E) when the Backup Heater Control switch is in AUTO-after-OFF. • Low Pressurizer level, 17% of span (LC-459C or 460C), when the Backup Heater Control switch is in AUTO-after-OFF or AUTO after-ON. • Overcurrent.
• Transfer switch in CONT RM • Vital power aligned	• Backup Heater Control switch on CC-1 to OFF. • Safety Injection signal trips the breaker to prevent overloading of the vital bus; the breaker cannot be reclosed until the SI signal is reset.
Groups 2 and 4 • Control Transfer Relay Cutout Switch is CUTIN • Transfer switch in LOCAL (HSDP) • Normal power aligned (Gp 2)	• Backup Heater Local Control switch on HSDP to OFF • Low Pressurizer level, 17% of span (LC-459C or 460C)

Continued on next page

RO Question 37

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	010 G2.4.6	
	Importance Rating:	3.1	4.0

Proposed Question:

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

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

A. PORV

B. Aux Spray

C. Normal Spray

D. Operating ECCS pumps as necessary

Proposed Answer:

B. Aux Spray

Explanation:

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

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

C incorrect. No RCPs running to use normal sprary.

D incorrect. Not a method of pressure control.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

ACTION EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **CONTINUE RCS Cooldown And Initiate Depressurization Toward RHR Entry Conditions (Continued):**	
	11) IF Letdown CANNOT be established, THEN (a) Use one PZR PORV to Depressurize the RCS. (b) GO TO Step 4 (Next Page).
f. Use Auxiliary Spray and Depressurize as follows: 1) Open both PCV-455A and B 2) Open 8145 or 8148, Auxiliary Spray Vlvs 3) Close 8146 and 8147, Normal and Alt Chg to LP4 and LP3 Cold Leg Vlvs 4) Control RCS Pressure by: o Adjusting Charging flow o Bypassing Auxiliary Spray with PCV-455A or B o Controlling PZR Heaters as necessary	f. IF Auxiliary Spray is NOT desired, THEN Use one PZR PORV AND GO TO Step 4 (Next Page)
g. Maintain Delta-T between Aux Spray flow and PZR Steam Space LESS THAN 320°F (TI-454 and TI-126, or PPC address U0498)	g. Perform the following: 1) Stop Auxiliary Spray flow. 2) Continue Depressurization using one PZR PORV. 3) GO TO Step 4 (Next Page).

RO Question 38

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	012 K1.03	
	Importance Rating:	3.7	3.8

Proposed Question:

The plant is at 100% power.

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

A. N44 fails high.

B. PT-505 fails low.

C. Loop 1 Thot fails high.

D. Controlling Pressurizer pressure channel fails low.

Proposed Answer:

A. N44 fails high.

Explanation:

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

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

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

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

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

Proposed references to be provided to applicants during examination: None

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

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Control Interlocks and Permissives

Signal	Setpoint	Logic	Interlock	Function
C-1 Intermediate Range Rod Stop (PK03-08)	> 20% power (current equiv.)	1/2	Blocked when trip is blocked (P10)	Stops outward rod motion.
C-2 Power Range Rod Stop (PK03-09)	> 103% nuclear power	1/4	Can be bypassed at NI cabinets	Stops outward rod motion.
C-3 OTdT Rod Stop & Turbine Runback (PK04-04/08-09)	1% below calculated trip setpoint	2/4		Stops outward rod motion. Runs back turbine at 10%/min net rate
C-4 OPdT Rod Stop & Turbine Runback (PK04-05/08-10)	1% below calculated trip setpoint	2/4		Stops outward rod motion. Runs back turbine at 10%/min net rate
C-5 Turbine Low Power (PK-8-11)	< 15% turbine load (PT-505)	1/1		Stops auto outward rod motion.
C-7A Turbine Load Rejection (PK08-12)	10% loss of load within 140 seconds (PT-506)	1/1	Manually reset	Arms 40% steam dump valves in Tavg mode.
C-7B Turbine Load Rejection (PK08-13)	50% loss of load within 140 seconds (PT-506)	1/1	Manually reset	Arms 35% & 10% steam dump valves in Tavg mode.
C-9 Condenser Available (PK08-14) (A)	> 20" Hg Vacuum -and- Circ water pump running (breaker closed)	2/2		Arming signal for 40% steam dump valves.
C-11 Bank D Rod Stop (PK03-15) (A)	Control Bank D > 220 steps			Stops auto outward rod motion.
C-20 AMSAC Armed (PK08-15) (A)	PT-505/506 equiv. to > 40% Core Rated Thermal Power (PT-505/506)	2/2		Arms AMSAC

(A) Indicates AMBER lens (a permissive which is normally lit during high power operation)

A "permissive" is a condition which permits certain actions to occur.
An "interlock" is a condition which prevents certain actions from occurring.

RO Question 39

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	013 K3.01	
	Importance Rating:	4.4	4.7

Proposed Question:

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

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

A. Core uncovery and overheating.

B. RCS pressure exceeding design pressure.

C. Return to criticality due to core boil off causing a loss of boron inventory.

D. Steam Generator dryout and tube damage from excessive steam generator pressure.

Proposed Answer:

A. Core uncovery and overheating.

Explanation:

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank modified M-0068

Question History: N/A

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

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

Comments:

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

1.00

What is the main concern during a loss of secondary heat sink accident, assuming no operator-initiated recovery actions?

A. Core uncovery and overheating.

B. S/G dryout and tube damage from overheating.

C. Excessive differential pressure across the S/Gs tubes.

D. Uncontrolled RCS cooldown and reactor restart.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

11317	Discuss the analyzed events associated with a Loss of Heat Sink and be able to describe the identified operator actions used to mitigate this event.
11319	Describe the loss of feedwater event leading to core damage.

Reference Id: M-0068
Must appear: No
Status: Active
User Text: 11317.
User Number 1:
User Number 2:
Difficulty: 1.00

1. INTRODUCTION

Guideline FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, provides guidance to address an extreme challenge (i.e., RED priority) to the Heat Sink Critical Safety Function that results if total feed flow is below a minimum value and level is below the narrow range in all SGs at any time. Less severe challenges to the Heat Sink Critical Safety Function that result from secondary inventory concerns on individual steam generators are discussed in the background documents for guidelines FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL, and FR-H.5, RESPONSE TO STEAM GENERATOR LOW LEVEL.

An early indication that secondary heat transfer capability may be challenged is that auxiliary feedwater (AFW) flow is not available to any steam generator. Following a reactor trip and safety injection (SI), main feedwater isolation is automatically initiated. Auxiliary feedwater flow to the steam generators must be automatically or manually initiated in order to maintain adequate secondary inventory for decay heat removal. Consequently, a failure of the AFW system results in a challenge to the Heat Sink Critical Safety Function. If reactor trip and SI occur, the operation of the AFW system is verified in guideline E-0, REACTOR TRIP OR SAFETY INJECTION, prior to Status Tree monitoring. If minimum AFW flow is not being provided, the operator is directed to implement guideline FR-H.1.

The objective of guideline FR-H.1 is to maintain reactor coolant system (RCS) heat removal capability by establishing feed flow to an SG or through establishing RCS bleed and feed heat removal. <u>Guideline FR-H.1 is entered at the first indication that secondary heat removal capability may be challenged.</u>
<u>This permits maximum time for operator action to restore feedwater flow to at least one steam generator before secondary inventory is depleted and secondary heat removal capability is lost. Once secondary heat removal capability is lost, RCS bleed and feed must be established to minimize core uncovery and prevent an inadequate core cooling condition.</u>

During Period 5, flow out of the pressurizer PORVs is a subcooled liquid. Even the minimum capacity pressurizer PORVs are expected to be large enough to maintain RCS pressure at the PORV set pressure. Thus, during Period 5, the RCS pressure remains constant at the PORV setpoint. If operator action is not taken, the system will continue in this mode until decay heat addition causes the reactor fluid to increase to the saturation temperature for water at the pressure of the PORV setpoint (e.g., T_{SAT} at 2335 psig is 659°F). Therefore, Period 5 (without operator action) is a time period where core decay heat energy is absorbed by the RCS fluid and the subcooling of the RCS liquid inventory is reduced. Furthermore, a great deal of RCS inventory is lost out of the PORVs in order to maintain pressure at the PORV setpoint.

<u>Period 6</u>

When the RCS fluid temperature reaches saturation, boiling will begin. The generation of steam within the RCS results in a large volumetric increase in the RCS inventory and the pressurizer PORVs may not be able to compensate for this. The result is that system pressure will rise until either the PORVs can handle the volumetric increase or the pressurizer code safety valves open relieving the excessive volumetric buildup in the RCS. In the example case (Figure 1) the PORVs are able to compensate for the steam generation at a system pressure of around 2405 psig and, thus, safety valves did not open. However, the PORVs are now being held open continuously by the steam generation rate in the core, and this condition will persist until the system boils off all the liquid inventory above the core and partial core uncovery occurs. <u>Significant core uncovery and potential core damage will follow unless operator action is taken to initiate bleed and feed or feedwater capability is restored.</u>

During the time interval covered by Periods 1 through 6, an automatic SI signal will not have been generated. If SI were manually actuated, it would not be effective in preventing core uncovery due to the limited capability of the SI pumps to inject when RCS pressure is at or above the pressurizer PORV setpoint. The mass flow rate out of the pressurizer PORVs is anywhere from 50 to 100 lbm/sec. The charging/SI pump system on the reference plant can inject

RO Question 40

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	013 A4.02	
	Importance Rating:	4.3	4.4

Proposed Question:

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

Current plant conditions:

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

The operator resets both trains of Safety Injection.

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

A. Only PK02-02 will be cleared.

B. Only PK08-21 will be cleared.

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

D. Neither PK08-21 or PK02-02 will be cleared.

Proposed Answer:

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

Explanation:

A incorrect. PK08-21 will also be cleared.

B incorrect. PK02-02 will also be cleared.

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

D incorrect. Both will be cleared.

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

Proposed references to be provided to applicants during examination: None

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

- Annunciator Windows
- Status Lights
- Monitor Light Boxes

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

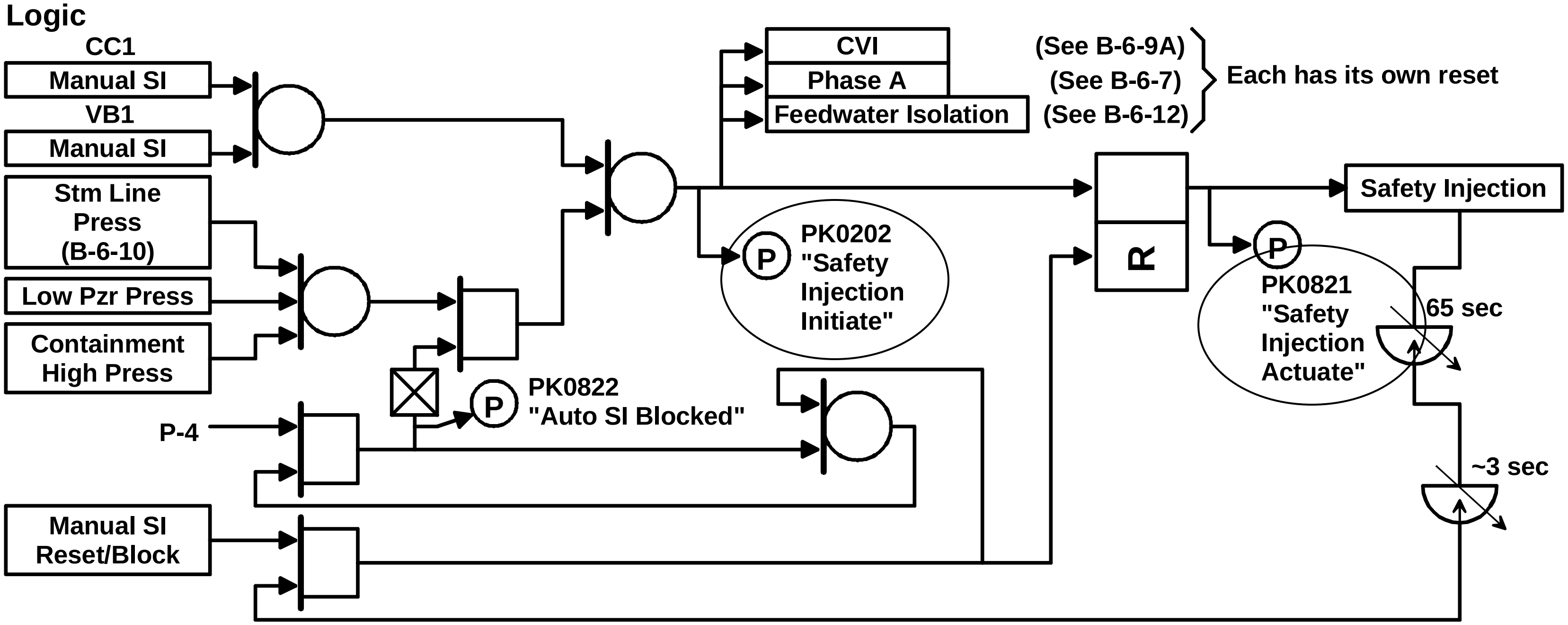
Safety Injection Signals

Signals

("S" Signal)

Signal	Setpoint	Coincidence	Interlocks	Protection Afforded
Manual	(NA)	1/2	None	Operator judgement
Containment High Pressure	3 psig	2/3	None	High energy line break inside containment
Pressurizer Low Pressure	1850 psig	2/4	P-11	LOCA
Low Steam Line Pressure	600 psig lead-lag	(See B-6-10)	P-11	Steam line break

Logic



Reference Drawing 495848

RO Question 41

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	022 A2.06	
	Importance Rating:	2.8	3.2

Proposed Question:

The plant is at full power. Three CFCUs are running in HIGH speed.

One of the running CFCUs trips on overcurrent.

The operator is directed to place a standby CFCU in service in HIGH speed.

Which of the following is the sequence of actions taken to place the unit in service?

A. Start the CFCU with the Speed Select switch in HIGH. The Speed Select switch remains in HIGH.

B. Start the CFCU with the Speed Select switch in LOW, then place the Speed Select switch in HIGH. After current stabilizes, return the Speed Select switch to LOW.

C. Start the CFCU with the Speed Select switch in HIGH. After current stabilizes, place the Speed Select switch to LOW.

D. Start the CFCU with the Speed Select switch in LOW, turn the unit OFF, then quickly restart with the Speed Select switch in HIGH. After current stabilizes, return the Speed Select switch to LOW.

Proposed Answer:

D. Start the CFCU with the Speed Select switch in LOW, turn the unit OFF, then quickly restart with the Speed Select switch in HIGH. After current stabilizes, return the Speed Select switch to LOW.

Explanation:

A incorrect. Initially started in LOW.

B incorrect. Unit turned off between switching from LOW to HIGH.

C incorrect. Started in LOW.

D correct. The unit is started in LOW. Then it is turned OFF and quickly started in HIGH. Following the start, the switch is taken back to LOW.

Technical Reference(s): OP H2:I

Proposed references to be provided to applicants during examination: None

Learning Objective: 68453 - Explain the basic principles of operation for the CFCUs.

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

6.1.7 IF the transfer switch at the Hot Shutdown Panel (HSDP) is in LOCAL, THEN position it to CONTROL ROOM and reset the transfer relay at the 480V switchgear.

6.1.8 Perform STP M-51 to verify CFCU operability.

6.1.9 For the initial start following maintenance, monitor CFCU performance from PPC group display "PK01-21."

NOTE: The CFCU is now available for use.

6.1.10 If desired, then start CFCUs using Section 6.2.

6.2 Starting a CFCU

6.2.1 Review Attachment 9.3 for limitations on running CFCUs.

NOTE 1: A CFCU vibration alarm can be expected when starting a CFCU or shifting a running CFCU to high speed.

NOTE 2: If an RCP oil level is near an alarm setpoint level, starting, stopping, or changing configuration of CFCUs may bring in an RCP alarm due to changes in local air temperature.

6.2.2 Place the Speed Select Switch in LOW speed and press it to start the CFCU.

6.2.3 Verify that current stabilizes.

6.2.4 IF high speed operation is desired, THEN perform the following:

a. Press the STOP pushbutton for the CFCU to stop it.

b. Quickly place the Speed Select Switch in HIGH speed and press it, to restart the CFCU.

c. Verify that current stabilizes.

d. Return the CFCU Speed Selector Switch to LOW speed.

6.2.5 IF annunciator PK01-21, "Contmt Fan Clrs", alarms, THEN perform the following:

a. Check the annunciator printout to confirm the cause as high vibration on the CFCU just started.

b. Press the Reset button on VB1 to reset the alarm.

c. IF alarm returns, THEN consult AR PK01-21 for further actions.

6.2.6 Repeat for each CFCU to be started.

RO Question 42

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	026 K4.09	
	Importance Rating:	3.7	4.1

Proposed Question:

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

A. 8809A, RHR cold leg injection isolation valve.

B. 9001A, Containment spray pump discharge valve.

C. 8982A, Containment recirculation sump suction valve.

D. 9003B, RHR to containment spray rings isolation valve.

Proposed Answer:

C. 8982A, Containment recirculation sump suction valve.

Explanation:

Only C correct.
9003A will open if both
- RHR-8982A, RHR pump 1 suction from Containment sump is open AND
- RHR-8701 OR -8702, RHR suction from RCS loop 4 is closed

Technical Reference(s): STG I2, Containment Spray

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-0308

Question History: Last NRC Exam: N/A

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

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

Comments:

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

1.00

Which valve from the list below is interlocked with the residual heat removal (RHR) to containment spray rings isolation valve (9003A)?

A. Containment recirculation sump suction valve (8982A).

B. RHR cold leg injection isolation valve (8809A).

C. Containment spray pump discharge valve (9001A).

D. RHR to containment spray rings isolation valve (9003B).

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

68489	Describe Containment Spray System automatic features.
37578	Analyze the control logic for Containment Spray system equipment, including: ? Pumps ? Valves

Reference Id: P-0308
Must appear: No
Status: Active
User Text: 6043.060264
User Number 1: 0000003.60
User Number 2: 0000003.40
Difficulty: 1.00

RHR to Containment Spray Valves (CS-9003A/B), Continued

Controls

Valves CS-9003A/B have control capability from the
- Control Room on VB1 and
- local handwheel.

Control at VB1	Operation
CLOSE/STOP/OPEN	3 position, maintain position.

Logic
Obj 9

The logic associated with valves CS-9003A/B is described in the following tables

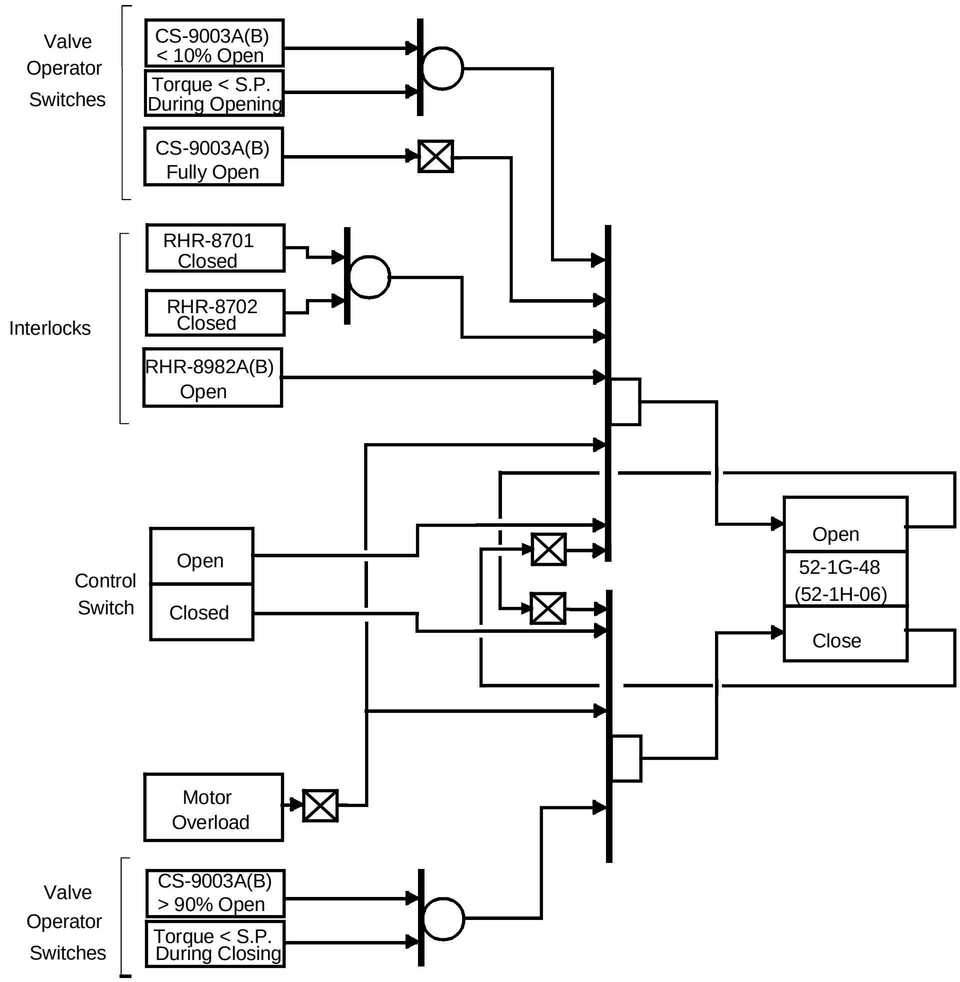
If the VB1 control switch is in ...	Then valve CS-9003A will ...
CLOSE	Close
OPEN	Open IF both • RHR-8982A, RHR pump 1 suction from Containment sump is open AND • RHR-8701 OR -8702, RHR suction from RCS loop 4 is closed
If the VB1 control switch is in ...	**Then valve CS-9003B will ...**
CLOSE	Close
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

RHR to Containment Spray Valves (CS-9003A/B), Continued

Logic Diagram
Obj 9

This is a logic diagram of the controls for CS-9003A, CS-9003B is similar.



CSS-13

Continued on next page

RO Question 43

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	026 A4.01	
	Importance Rating:	4.5	4.3

Proposed Question:

GIVEN:

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

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

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

A. No effect. To manually actuate, take the Phase B switch of the non-running train to Actuate.

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

C. Manual actuation is been blocked until Containment pressure is less than 3 psig.

D. Manual actuation is been blocked until Containment pressure is less than 22 psig.

Proposed Answer:

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

Explanation:

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

Technical Reference(s): STG B6A, RPS

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments:

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

Phase B CI and Containment Spray Actuation, Continued

Logic (continued)

IF	AND	Then

Phase B CI Train A (B) signal is RESET		the Train A (B) Phase B CI signal is reset.
		• the manual actuation signal goes through a retentive memory unit and will seal in until reset, once reset, manual actuation of Phase B CI is still available.
		• the auto actuation signal goes through a retentive memory with manual reset unit and will also seal in until reset.
		  • the signal can be reset even though the initiating signal is still present.
		  • automatic actuation is blocked until the Containment High High Pressure signal is cleared.
		• PK02-01, CI Phase A/B will clear.
		• Red light for Train A (B) above Monitor Light Box D goes out.

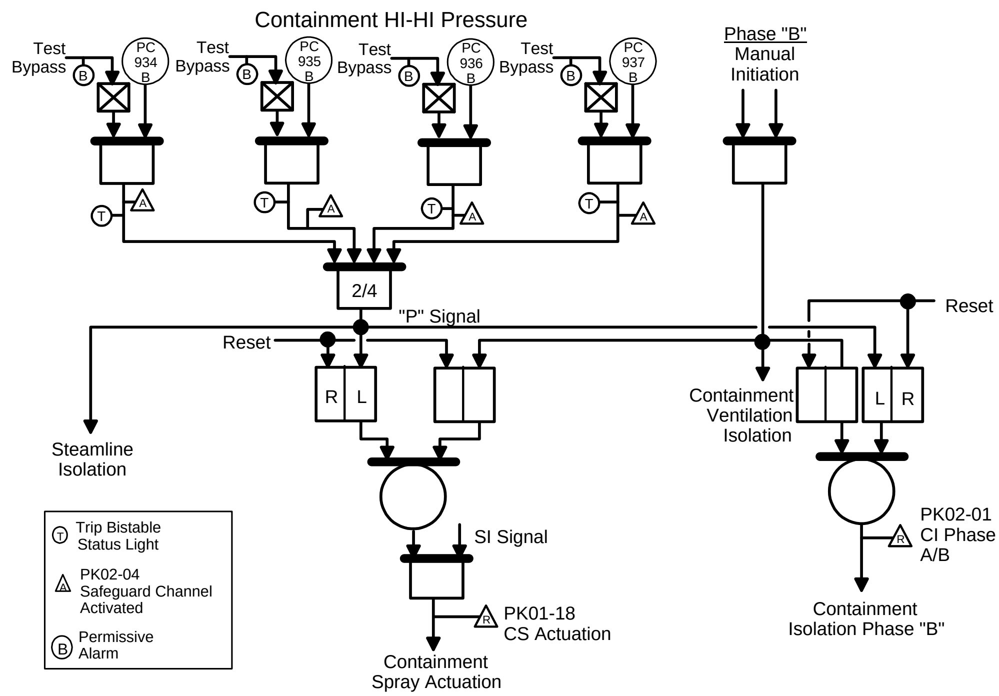
CS Train A (B) signal is RESET		the Train A (B) CS signal is reset.
		• the manual actuation signal goes through a retentive memory unit and will seal in until reset, once reset, manual actuation of CS is still available.
		• the auto actuation signal goes through a retentive memory with manual reset unit and will also seal in until reset.
		  • the signal can be reset even though the initiating signal is still present.
		  • automatic actuation is blocked until the Containment High High Pressure signal is cleared.
		• PK01-18, CS Actuation will clear.

Continued on next page

Phase B CI and Containment Spray Actuation, Continued

Logic diagram
Obj 11

The logic diagram for the Phase B CI and CS Actuation signal is shown below. Only one SSPS Train is shown, the other is identical.



RPS-33

Indications
Obj 5

The following indications are available for the Phase B CI CS Actuation signal.

VB1 Bistable Status Panel		
Indicating Light	**Meaning**	**Normal Status**
Red	Hi Cntnmt Press bistable PC934B tripped	OFF
White	Hi Cntnmt Press bistable PC935B tripped	
Blue	Hi Cntnmt Press bistable PC936B tripped	
Yellow	Hi Cntnmt Press bistable PC937B tripped	

Continued on next page

RO Question 44

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	039 A1.10	
	Importance Rating:	2.9	3.0

Proposed Question:

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

A. INCREASING SJAE Exhaust Flowrate or INCREASING RCS Noble Gas Concentration.

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

C. DECREASING SJAE Exhaust Flowrate or DECREASING RCS Noble Gas Concentration

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

Proposed Answer:

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

Explanation:

A incorrect. Decreasing SJAE flow not increasing.

B incorrect. Decreasing SJAE flow and Increasing Noble gas.

C incorrect. Increasing noble gas.

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

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

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

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

TITLE: Primary to Secondary Steam Generator Tube Leak Detection

12/07/06
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This Operating Order sets forth the policies and procedures that are to be followed to provide early detection of a leaking Steam Generator tube.

1.2 This Operating Order provides direction for the on-shift crew on the required actions to be taken if the indicated Steam Generator tube leakage increases.

2. DISCUSSION

2.1 The response to Steam Generator Tube Leakage within this procedure is divided into four major categories. The section used is dependent upon the amount of primary to secondary leakage per Steam Generator, estimated from radiation monitor readings. Each category gives specific guidance on operator actions to mitigate the effects of the primary to secondary leakage. The six major categories are:

2.1.1 Normal Operation **-** Less than detectable OR < 5 gallons per day (gpd).

2.1.2 Increased Monitoring **-** ≥ 5 gpd and < 30 gpd.

2.1.3 Action Level 1 **-** ≥ 30 gpd and < 75 gpd (0.05 gpm).

2.1.4 Action Level 2 - ≥ 75 gpd - sustained ≥ 1 hr. **WITH** a leak rate increase < 30 gpd/hr.

2.1.5 Action Level 3a - ≥ 75 gpd **- WITH** a leak rate increase ≥ 30 gpd/hr.

2.1.6 Action Level 3b - ≥ 150 gpd. (0.1 gpm)

2.2 The rate of increase associated with Action Levels 2 and 3a (see step 2.1) can be evaluated as leak rate changes over time intervals not exceeding 30 minutes (e.g., leak rate has not increased ≥ 15 gpd in a 30 minute period). Alternately, the PPC rate of increase calculation may be used

2.3 The Air Ejector Off-gas monitors, RM-15 and 15R, are the most sensitive indicators of small changes in Steam Generator tube leakage. The Plant Process Computer (PPC) receives an input from RM-15 and 15R. The Plot or Archive function of the PPC can be used to detect small changes in the Air Ejector Off-gas count rate.

2.3.1 The High Alarm set point of RM-15 and 15R is set to the activity level that should be seen by the detector if the previously calculated RCS activity still exists and a total of 30 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the High Alarm setpoint may correspond to a leak rate greater than 30 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

2.3.2 The Alert alarm set point of RM-15 and 15R is set to the activity level that would be seen by the detector if the previously calculated RCS activity still exists and a total of 20 gallons per day is leaking from the RCS to the Steam Generators. When RCS activity is very low, the Alert Alarm setpoint may correspond to a leak rate greater than 20 gallons per day. Refer to the I&C RMS Data Book for actual High Alarm set point.

2.3.3 Because the established monitor High Alarm and Alert Alarm setpoints are fixed values, they may correspond to a leak rate value that is different than 30 gpd or 20 gpd, depending upon changes in assumptions used to calculate the setpoints. The table below describes how some of these changes affect the setpoint:

Parameter	Relative Parameter Change	Alarm Setpoint Affect
RCS Noble Gas Concentration	↑ ↓	More Conservative Less Conservative
SJAE Exhaust Flowrate	↑ ↓	Less Conservative More Conservative

2.3.4 Primary to secondary leak rates can be inferred by comparing existing RM-15, 15R readings to the Alarm and Alert set points. The relationship of leak rate to activity level is linear as indicated activity increases.

2.3.5 Chemistry projects what the RM-15(R) count rate will be for a Steam Generator Tube leak of 30 gpd, 75 gpd and 150 gpd. This information is provided for reference in accordance with CY1.DC3. This projected value can be used by the Shift Foreman to roughly estimate Steam Generator leak rates.

2.3.6 Frequent analysis of RM-15(R) count rate and evaluation of the change in count rate with time is the best indicator of changes in the Steam Generator leak rate.

2.4 The PPC monitoring of RM-15(R) also provides an alarm and several warnings if Air Ejector Off-gas count rate increases. These warnings/alarms are:

2.4.1 HALM - Provides an alarm at the same time the Main Annunciator PK11-06, alarm input 423 does. Refer to I&C Rad Monitor Data Book for set point.

2.4.2 HWRN - Provides the first warning if RM count rate increases from the normal. Refer to I&C Rad Monitor Data Book for set point.

2.4.3 Significant Change Alarming - The HWRN alarm will reflash each time the value increases by a third of the difference between the HWRN and HALM alarm set points. (i.e., if the HWRN and HALM set points are 250 cpm and 1000 cpm, the HWRN will reflash at 500 cpm and again at 750 cpm.)

2.4.4 PPC points available for monitoring (GRPDIS OP O-4):

a.	RM-15	R0015A
b.	RM-15R	R0115A
c.	RM-19	R0019A
d.	FIT-81	F0704A or U5515
e.	Leak rate based on RM-15	U5503
f.	Leak rate based on RM-15R	U5504
g.	Rate of increase based on RM-15	U5507
h.	Rate of increase based on RM-15R	U5508

2.4.5 Wide-range SJAE FIT-81 live data is used in the PPC S/G tube leakage calculation when FIT-81 is OPERABLE (KPSLK is set to zero, and F0704A is used). However, PPC monitoring may still be credited when FIT-81 is out-of-service. In this case, KPSLK is set to 1, and the PPC substitutes an existing SJAE off gas flow constant (KSJAEFLW, based on previous chemistry data) into the calculation (U5515). This is the "pre-planned alternate method of determining SJAE flow rate" described in ECG 2.1, action A.1.1.2 bases.

2.5 Confirmed spiking on RM-15 and 15R that is increasing in frequency and/or amplitude is a precursor to SG tube failure. These symptoms were observed at Comanche Peak Unit One prior to their tube leak/failure (LER 445-02002).

3. RESPONSIBILITIES

3.1 Shift Foreman (SFM) is responsible for evaluating the RM-15(R) count rate increases and grab sample results for Steam Generator leakage changes.

3.2 Chemistry is responsible to provide RCS activity levels to the SFM so that RM-15(R) readings can be equated to leak rates between the primary and secondary systems. Immediate notification of the Control Room should be made if indications by grab samples indicate a leak rate ≥ 5 gpd.

3.3 Normally, the PPC will monitor and alarm steam generator tube leakage. If PPC monitoring is out of service, the Control Operator or the BOP Control Operator shall compare the changes in RM-15(R) count rate at least every 4 hours per step 6.1.3. As manpower needs dictate, utilize the Shift Engineer to perform manual monitoring, especially when 15-minute intervals are required.

4. PREREQUISITES

4.1 Air Ejector Off-gas monitors, RM-15 and RM-15R shall have a High Warning alarm set to a value high enough above normal reading, but low enough to give an early warning of changes in Air Ejector Off-gas count rate.

4.2 PPC address U5503 and U5504, Primary-to-Secondary leak rate (based on RM-15(R) and Air Ejector Flowrate) should have a High Warning alarm set to a value that gives an early warning of a significant change in leak rate.

4.3 PPC address U5507 and U5508, Primary-to-Secondary leak rate of increase (based upon an average of recent leak rate calculations) should have a High Warning alarm set to a value that gives an early warning of a significant indication of leak rate change (gpd/hr).

4.4 The Significant Change Alarming function for Modes 1, 2 and 3 shall be enabled.

5. PRECAUTIONS

None

6. INSTRUCTIONS

NOTE: RM-15 and RM-15R are identical and redundant monitors. Either one can be used to meet the intent of this procedure. RM-15 is the preferred data point for consistent log data.

6.1 NORMAL OPERATION - Less than detectable OR < 5 gpd per S/G

6.1.1 PPC monitoring of S/G tube leakage:

a. Initiate PPC monitoring (GRPDIS OP O-4).

b. Verify RM-15 or RM-15(R), Condenser Air Ejector Radiation Monitors, and Plant Process Computer, PPC, are in service and OPERABLE, and PPC quality codes are NOT light blue (cyan).

c. Verify FIT-81, SJAE Off-gas Flow monitor status (ECG 2.1).

1. If FIT-81 is OPERABLE and in service, verify PPC constant KPSLK = 0.

2. If FIT-81 is NOT OPERABLE, verify PPC constant KPSLK = 1 in accordance with OP O-15, "Control of PPC Addressable Points".

d. Check R0015A (R0115A) HWRN or HALM not in alarm status.

e. Check leak rate U5503 (U5504) HWRN or HALM not in alarm status.

f. Check rate of increase, U5507 (U5508) HWRN or HALM not in alarm status.

g. Inform the SFM if any S/G tube leakage monitoring points are in alarm.

6.1.2 Perform a qualitative evaluation of Air Ejector radiation monitor count rate by viewing a continuous trend of the monitor output. Look for any upward trends. The evaluation of the count rate may be made using any one of the following methods:

a. Maintaining a continuous plot of R0015A or R0115A on any Control Room PPC CRT with a time scale of 270 minutes.

b. Maintaining a continuous trace of R0015A or R0115A on Control Room analog trend recorder.

c. Running an Archival Plot of R0015A or R0115A and another associated PPC radiation monitor input (R0019A or R0023R is suggested) every four hours.

6.1.3 Manual monitoring of S/G tube leakage (PPC monitoring out of service).

a. Record RM-15(R) count rate, Air Ejector Off-gas flow rate, and the results of the evaluation on Attachment 9.1 every four hours.

b. Evaluate the RM-15(R) count rate at least once every four hours for an increase in the instantaneous value and/or a slow increase of count or leak rates over the previous four hours.

c. Enter UNSAT for the evaluation if any of the following conditions exist:

1. RM-15(R) or Air Ejector grab sample count rates increase by a factor of 2 or greater over the previous 4 hour period.

2. RM-15(R) count rates indicate that primary to secondary leakrate has increased to 30 gpd, or has increased more than 20 gpd in the last 4 hours.

d. Inform the SFM if the results of the evaluation show an increasing count rate or rate of increase.

6.1.4 Normal Operation with RM-15 and RM-15R Out Of Service

a. Inform the shift C&RP technician to increase Air Ejector grab sampling per OM12.DC1.

b. <u>IF</u> the leak rate was unstable or on an increasing trend when RM-15 and RM-15R were both removed from service,
<u>THEN</u> perform the following:

1. Enter Action Level 2, paragraph 6.4

2. Inform the shift C&RP technician to increase Air Ejector grab sampling to every 4 hours.

c. <u>IF</u> in Action Level 2 when RM-15 and RM-15R are both removed from service,
<u>THEN</u> enter Action Level 3b, paragraph 6.6.

d. Elevate restoration of RM-15(R) to the highest priority.

6.2 Increased Monitoring - ≥ 5 gpd and < 30 gpd

6.2.1 Normally, this condition would be identified by chemistry's routine grab samples. Any indication of leakage ≥ 5 gpd should be immediately reported to the Control Room.

6.2.2 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to FIRST identify the source and THEN quantify and amount of primary to secondary leakage per chemistry procedure CAP D-15.

a. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.

6.2.3 If any Radiation Monitors for monitoring the secondary system per step 6.1 are out of service, elevate their restoration to the highest priority. Manual monitoring per steps 6.1.3 or 6.1.4 may be required.

NOTE: The following will provide a warning of additional increases in leakrate.

6.2.4 If any secondary system radiation monitors alarm, request that chemistry initiate a re-calculation and resetting of the setpoints above their current levels, but no higher than 30 gpd per CYC2.DC1, "Radiation Monitoring High Alarm Setpoint Control Program."

6.2.5 Tailboard OP AP-3.

6.3 Action Level 1 - ≥ 30 gpd and < 75 gpd.

6.3.1 If a HWRN alarm is received and evaluated as NOT having been caused by a transient noise spike on the radiation monitor, commence increased monitoring per step 6.3.3.

6.3.2 If RM-15(R) or Air Ejector grab sample indicate a substantial increase (greater than a factor of 2 over the previous 4 hour average) **OR** RM-15(R) is above the Alert set point, commence increased surveillance per step 6.3.3.

6.3.3 With indications of increasing primary to secondary leak rate as in step 6.3.1 or 6.3.2 above, begin increased surveillance as follows:

a. Increase monitoring of the RM-15(R) count rate to continuous by dedicating a Control Room PPC CRT to the trending of the RM-15 (R0015A) and/or RM-15R (R0115A) count rate(s), S/G leak rate calculations (U5503 or U5504) and rate of increase (U5507 or U5508).

1. If PPC is in service, visual monitoring of these points and associated alarms should be made approximately once every 15 minutes.

2. If PPC is out of service, manual monitoring, per step 6.1.3, should be completed every 15 minutes.

3. If the PPC, RM15 and RM-15R are all out of service, comply with step 6.1.4.

b. Direct the shift C&RP technician to attempt to FIRST identify the source and THEN quantify and amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.

1. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.

c. Direct the shift C&RP technician to increase SJAE sampling in accordance with OM12.DC1 until otherwise directed by the Shift Foreman.

6.3.4 Contact chemistry for an evaluation of the potential for contamination of the secondary system.

6.3.5 Once the leak rate is stable for 1 hour (≤ 10% increase during a one-hour period), monitoring of RMS responses can be reduced to once every 2 hours. Once leak rates are stable for 24 hours (≤ 10% increase during 24 hours), monitoring frequencies can return to those required by step 6.2.

a. Evaluate resetting RM-15(R) HWRN and HALM setpoints to above their existing baseline reading, but not over equivalent 75 gpd.

<u>NOTE</u>: The following will provide a warning of additional increases in leakrate.

6.3.6 Once leak rate is stable for 1 hour, request that chemistry initiate a re-calculation and reset RM-15(R) HWRN and HALM setpoints above their current levels, but no higher than 75 gpd per CYC2.DC1, "Radiation Monitoring High Alarm Setpoint Control Program."

6.3.7 Tailboard OP AP-3 if not done previously.

6.4 Action Level 2 **-** ≥ 75 gpd (0.05 gpm) sustained ≥ 1 hr. **WITH** a leak rate increase < 30 gpd/hr

6.4.1 Verify PPC is monitoring leak rate (U5503 or U5504) and rate of increase (U5507 or U5508) every 15 minutes. See step 6.1.1 for details.

a. If rate of increase is < 30 gpd/hr, continue monitoring.

b. If rate of increase is ≥ 30 gpd/hr, go to Action Level 3a.

6.4.2 If PPC monitoring is out of service and either RM-15 or RM-15R are available, determine leak rate and rate of increase every 15 minutes on Attachment 9.1

a. (Leak rate increase over a 30 minute period) x 2 = gpd/hr.

b. If < 30 gpd/hr, continue monitoring.

c. If ≥ 30 gpd/hr, go to Action Level 3a.

6.4.3 If the PPC, RM15 and RM-15R are all out of service, comply with step 6.1.4.

6.4.4 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to FIRST identify the source, and THEN quantify the amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.

a. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.

6.4.5 IF confirmed leak rate is ≥ 75 gpd,
AND rate of increase is < 30 gpd/hr,
THEN reduce power to take the unit off-line and be in Mode 3 within 24 hours.

a. A qualitative confirmation of the leak rate can be made by channel checking the RM-15(R) count rates against an upscale indication on either blowdown or steamline radiation monitors.

6.4.6 Contact chemistry and RP for an evaluation of the potential for contamination of the secondary system.

6.4.7 Tailboard OP AP-3 if not done previously.

NOTE: The 30 gpd/hr rate of increase should be sustained for a minimum of 30 minutes before taking action to initiate shutdown per Action Level 3a.

6.5 Action Level 3a - ≥ 75 gpd (0.05 gpm) **AND** ≥ 30 gpd/hr

6.5.1 IF confirmed leak rate is ≥ 75 gpd,
AND a continued rate of increase of ≥ 30 gpd/hr measured over a minimum of 30 minutes, (leak rate increase over a 30 minute period x 2 = gpd/hr),
THEN reduce to ≤ 50% power within 1 hour and be in Mode 3 within the next 2 hours.

a. A qualitative confirmation of the leak rate can be made by channel checking the RM-15(R) count rates against an upscale indication on either blowdown or steamline radiation monitors.

b. See step 6.1.1 for PPC monitoring.

c. See step 6.4.2 for manual monitoring.

6.5.2 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to FIRST identify the source, and THEN quantify the amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.

a. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.

6.5.3 Contact chemistry and RP for an evaluation of the potential for contamination of the secondary system.

6.6 Action Level 3b - ≥ 150 gpd (0.1 gpm). This limit includes leak rate spikes.

6.6.1 IF confirmed leak rate is ≥ 150 gpd,
THEN be in Mode 3 within 6 hours.

a. A qualitative confirmation of the leak rate can be made by channel checking the RM-15(R) count rates against an upscale indication on either blowdown or steamline radiation monitors.

b. If shutting down per Action Level 3a, do not relax the Mode 3 time limit if leak rate is ≥ 150 gpd.

c. See step 6.1.1 for PPC monitoring.

d. See step 6.4.2 for manual monitoring.

6.6.2 Direct the shift C&RP technician to increase surveillance of all secondary Radiochemistry and to attempt to FIRST identify the source, and THEN quantify the amount of primary to secondary leakage per chemistry procedures CAP AP-1 and CAP D-15, respectively.

a. Due to the complex nature of CAP D-15 it may be necessary to call in additional chemistry personnel.

6.6.3 Contact chemistry and RP for an evaluation of the potential for contamination of the secondary system.

6.6.4 Tailboard OP AP-3 if not done previously.

7. REFERENCES

7.1 Enclosure to PG&E Letter No. DCL-88-064

7.2 NRC Bulletin 88-02

7.3 NRC Information Notice 88-99

7.4 NRC Information Notice 91-43

7.5 EPRI Guideline 1008219 Final Report, December 2004, "PWR Primary-to-Secondary Leak Guidelines - Revision 3"

7.6 OM12.DC1, "Relieving the Watch"

8. RECORDS

8.1 Route the Data Sheet to the Shift Foreman for evaluation

8.2 Route completed Data Sheets to chemistry

9. ATTACHMENTS

9.1 "RM-15(R) Count Rate Evaluation Data Sheet," 05/08/04

DIABLO CANYON POWER PLANT
OP O-4
ATTACHMENT 9.1

1 AND 2

TITLE: RM-15(R) Count Rate Evaluation Data Sheet

Sample Date: ______________

UNIT [ ]1 [ ] 2

From Plant Chemistry Data: RM-15(R) Bkgnd (cpm) . ______________ RM-15(R) Net Response (cpm/gpd) ______________

Projected RM-15(R) cpm,
30 gpd leak rate ______________ cpm 75 gpd leak rate ______________ cpm 150 gpd leak rate ______________ cpm

DATE	TIME	RM-15 OR RM-15R (Circle One) CPM	SJAE O/G* Flow SCFM	LEAK RATE = RM-15(R) - RM15 Bkgnd / RM15(R) Net Response	30 MIN RATE OF INCREASE gpd/hr (N/A except per step 6.4.2)	Evaluation (See step 6.1.3c) (SAT/UNSAT)
		CPM	scfm	GPD	gpd/hr	

* If > 5 scfm variation contact chemistry for evaluation of flowrate.

Forward completed Data Sheets to chemistry

RO Question 45

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	039 G2.4.31	
	Importance Rating:	3.3	3.4

Proposed Question:

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

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

What is the significance of this alarm?

A. An MSI is about to occur.

B. A sudden increase in steam flow has occurred.

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

D. Delta P across the Steam Generator U-tubes is close to the design limit.

Proposed Answer:

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

Explanation:

C is correct. Alarm is at RCS pressure greater than 1850 psig and steam generators are less than 630 psig. If RCS pressure is allowed to increase above P-11, SI on low steam pressure would actuate.
Tsat for 485°F is about 590 psia (575 psig).

Technical Reference(s): AR PK06-06

Proposed references to be provided to applicants during examination: None

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

Question Source:

New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

RCS - S/G PRESS MISMATCH

02/01/06
Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
863	RCS S/G Pressure Mismatch Pre S.I. Warning	Y1000C	RCS pressure > 1850 psig <u>AND</u> SG pressure < 630 psig	2.1

2. OPERATOR ACTIONS

2.1 General Actions

NOTE: This alarm indicates that Low Steam Line Pressure Safety Injection setpoint is being approached.

2.1.1 IF pressurizing the RCS,
THEN perform the following:

a. STOP the RCS pressurization. [ ]

b. Heat up the RCS until SG pressures are above 630 psig PER OP L-1, "Plant Heatup From Cold Shutdown to Hot Standby". [ ]

c. RETURN TO procedure and step in affect. [ ]

CAUTION:	Prior to reducing pressure below P-11 (1915 psig) and blocking the Low Streamline Pressure SI signal, the RCS shall be borated per STP R-19 to cold shutdown conditions to avoid an unanalyzed condition during a main steam line rupture.[Ref 4.2]

2.1.2 IF cooling down the RCS,
THEN perform the following:

a. STOP the RCS cooldown. [ ]

b. Before continuing, verify RCS borated to cold shutdown conditions per STP R-19, "Shutdown Margin Determination".[Ref 4.2] [ ]

c. BLOCK Low Steam Line Pressure Safety Injection signal (CC2). [ ]

d. Before continuing, depressurize the RCS to less than 1850 psig PER OP L-5, "Plant Cooldown From Minimum Load to Cold Shutdown". [ ]

e. RETURN TO procedure and step in affect. [ ]

2.1.3 Probable Causes

- Pressurizing the RCS without heating up the RCS
- Cooling down the RCS without depressurizing the RCS

3. AUTOMATIC ACTIONS

3.1 Possible Safety Injection if main steam line pressure is 600 psig or below, and RCS pressure is 1915 psig or above

4. REFERENCES

4.1 501135, "Electrical Schematic Diagram - Main Annunciator" (Electrical Drawing Section 8)

4.2 A0564034, "Assessment Of Callaway Plant Event Report 39142"

5. LOGIC DIAGRAM

Y1000C ⟶ **ALARM**

RO Question 46

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	059 K1.05	
	Importance Rating:	3.1	3.2

Proposed Question:

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

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

A. A running MFP trips.

B. A 10% steam dump valve fails open.

C. A steam generator narrow range level transmitter fails high.

D. A steam generator main feedwater reg valve fails full open.

Proposed Answer:

A. A running MFP trips.

Explanation:

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

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

C incorrect. Control system will disregard a failed channel.

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

Technical Reference(s): LPA-15

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam

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

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

Comments:

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

Lesson

Main Feedwater System

Normal operation

During normal plant operation at 100% power, both Main Feedwater pumps are in operation.
- Each MFP is capable of approximately 60% flow.

Loss of MFP transient
Obj 1

The plant transient initially seen on a loss of a MFP is shown in the following table.

Component/ Parameter	Reaction
Reactor power	Reactor power decreases slightly due to decreased secondary heat transfer as seen by the following RCS indications: • T_{ave} increases • Pzr level increases • Loop ΔT decreases
Turbine load	Turbine load increases due to increased steam flow through LP turbines.
S/G	S/G inventory density decreases due to heatup. • The swell masks the inventory loss due the remaining feedwater pump not being able to supply enough flow for present steam flow.
Plant trip	With no operator action the plant will trip on low S/G level in approximate 45 seconds from 100% power.

Special considerations

While performing AP-15 to recover from the loss of a feedwater pump transient, some of the special considerations the operator should be aware of are:
- Rapidly reducing turbine load causes S/G level to shrink.
- Maintaining S/G pressure below the 10% atmospheric dump valve setpoint is desired.
- Very rapid operator action is required to prevent a reactor trip.

RO Question 47

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	059 G2.1.27	
	Importance Rating:	2.8	2.9

Proposed Question:

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

A. A constant Main Feed Pump speed for all power levels.

B. A constant mass in the steam generators for all power levels.

C. Inventory in the steam generators such that the U-tubes remain covered following a reactor trip.

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

Proposed Answer:

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

Explanation:

A incorrect, MFP speed changes as power level changes.

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

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank N-73157 modified

Question History: Last NRC Exam

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

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

Comments:

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

Basic Description

Purpose of the DFWCS
Obj 1

The purpose of the Digital Feedwater Control System (DFWCS) is to automatically maintain a programmed steam generator water level during steady-state operation and to restore and maintain the water level within acceptable levels during normal unit transients

Basic description

The DFWCS:

- uses a state of the art, microprocessor based system design,
- houses hardware in two cabinets, located behind the vertical boards in the control room,
  - circuitry dedicated to loops one and two and main feedwater pump speed control is housed in cabinet one,
  - circuitry dedicated to loops three and four is housed in cabinet two.

The DFWCS controls main feedwater flow by sending positioning signals to:

- main feed regulating (reg) valves,
- main feed reg bypass valves, and
- main feed pump speed control,
  - to maintain a programmed differential pressure between the feed and steam pressures, and
  - maintain the programmed level in the steam generators.

The control system derives an error signal for the controlling elements based on inputs from key parameters of the Steam Generator System, Main Steam Supply System, Main Turbine and Main Feedwater System.

Continued on next page

N-73157

The Digital Feedwater Control System controls feedwater flow to maintain:

A. programmed level in the Steam Generator.

B. constant mass in the Steam Generator.

C. constant level in the Steam Generator.

D. programmed pressure in the Steam Generator.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

70366	State the purpose of the Main Feedwater Control System

Reference Id: N-73157
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 1.00

RO Question 48

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	061 K5.03	
	Importance Rating:	2.6	2.9

Proposed Question:

Unit 1 trips from low power.

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

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

A. Pump head INCREASES and volumetric flowrate decreases to zero.

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

C. Pump head DECREASES and volumetric flowrate decreases to zero.

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

Proposed Answer:

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

Explanation:

A incorrect. Recirc maintains some flow.

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

C incorrect. Pump head increases.

D incorrect. Pump head increases.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

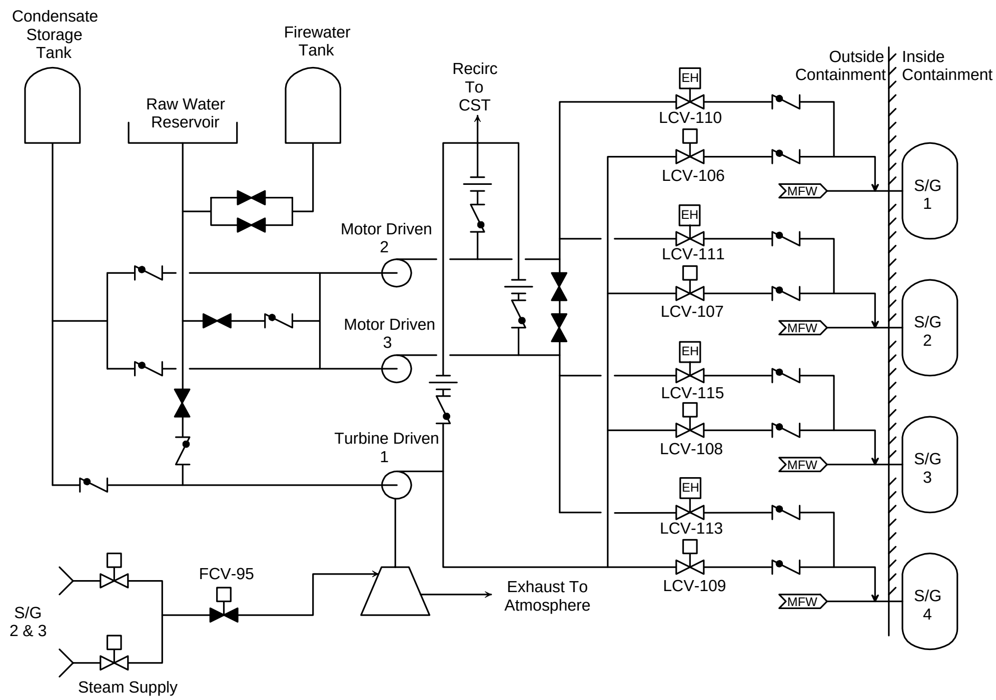
Comments:

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

Basic Description, Continued

Basic flowpath
Obj 2

The basic block and flow diagram of the AFW system is shown here.



D-1-1

System Coefficient Effects
(Open System)

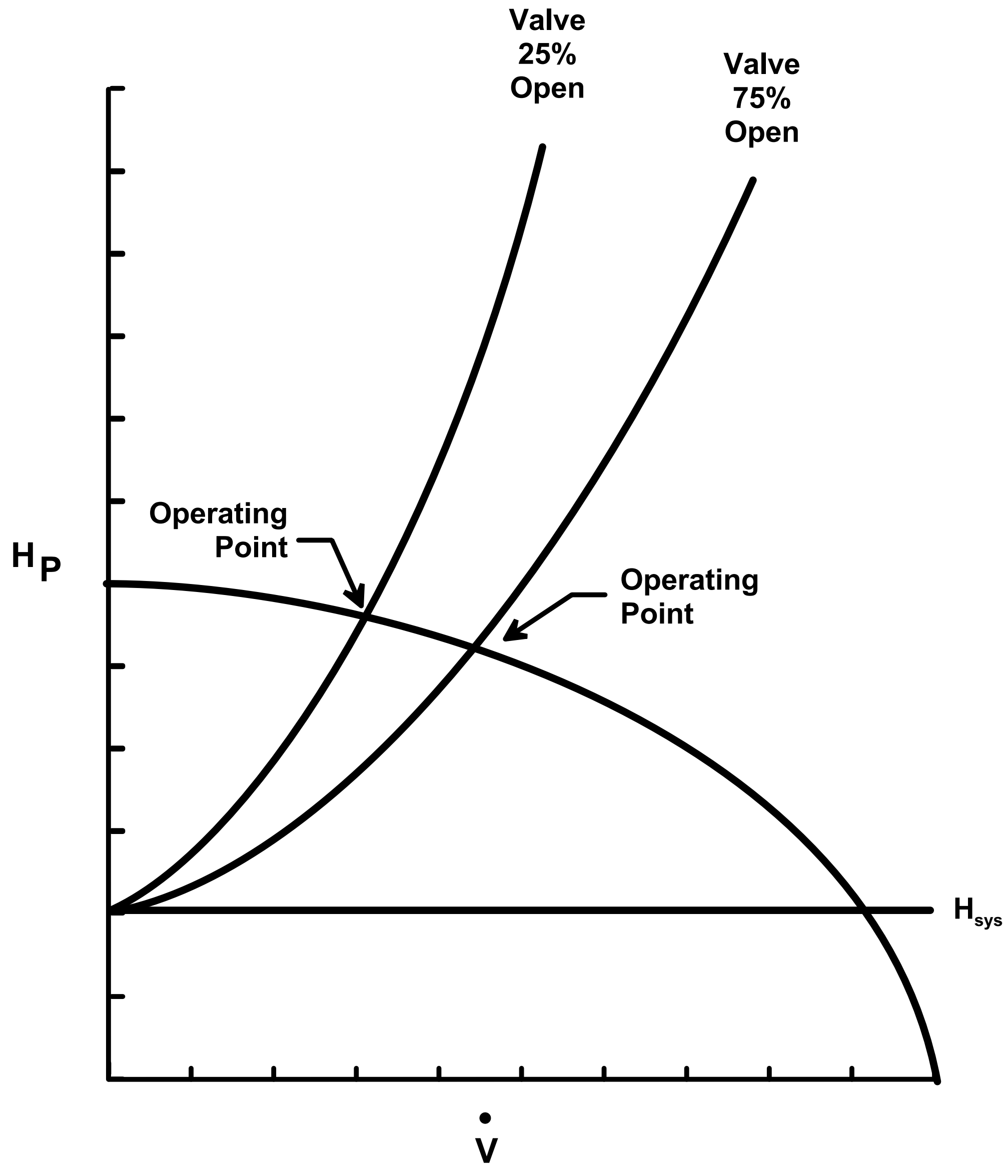


Figure TH11-18

RO Question 49

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	062 K2.01	
	Importance Rating:	3.3	3.4

Proposed Question:

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

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

A. Diesel 2-2 to Vital Bus G.

B. Diesel 2-1 to Vital Bus G.

C. Diesel 2-3 to Vital Bus F.

D. Diesel 2-1 to Vital Bus F.

Proposed Answer:

B. Diesel 2-1 to Vital Bus G.

Explanation:

Diesels supply power to the following vital buses:
1-1 Vital Bus H
1-2 Vital Bus G
1-3 Vital Bus F

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

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

A incorrect. Diesel 2-1 powers bus G.

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank – P-0648 modified

Question History: Last NRC Exam: N/A

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

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

Comments:

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

1.00

Which ONE of the following statements is correct concerning the POWER SUPPLY (F,G, or H), and ECCS TRAIN (A or B) for the 1-1 RHR pump?

A. Bus G, Train B

B. Bus G, Train A

C. Bus F, Train B

D. Bus F, Train A

Answer: A

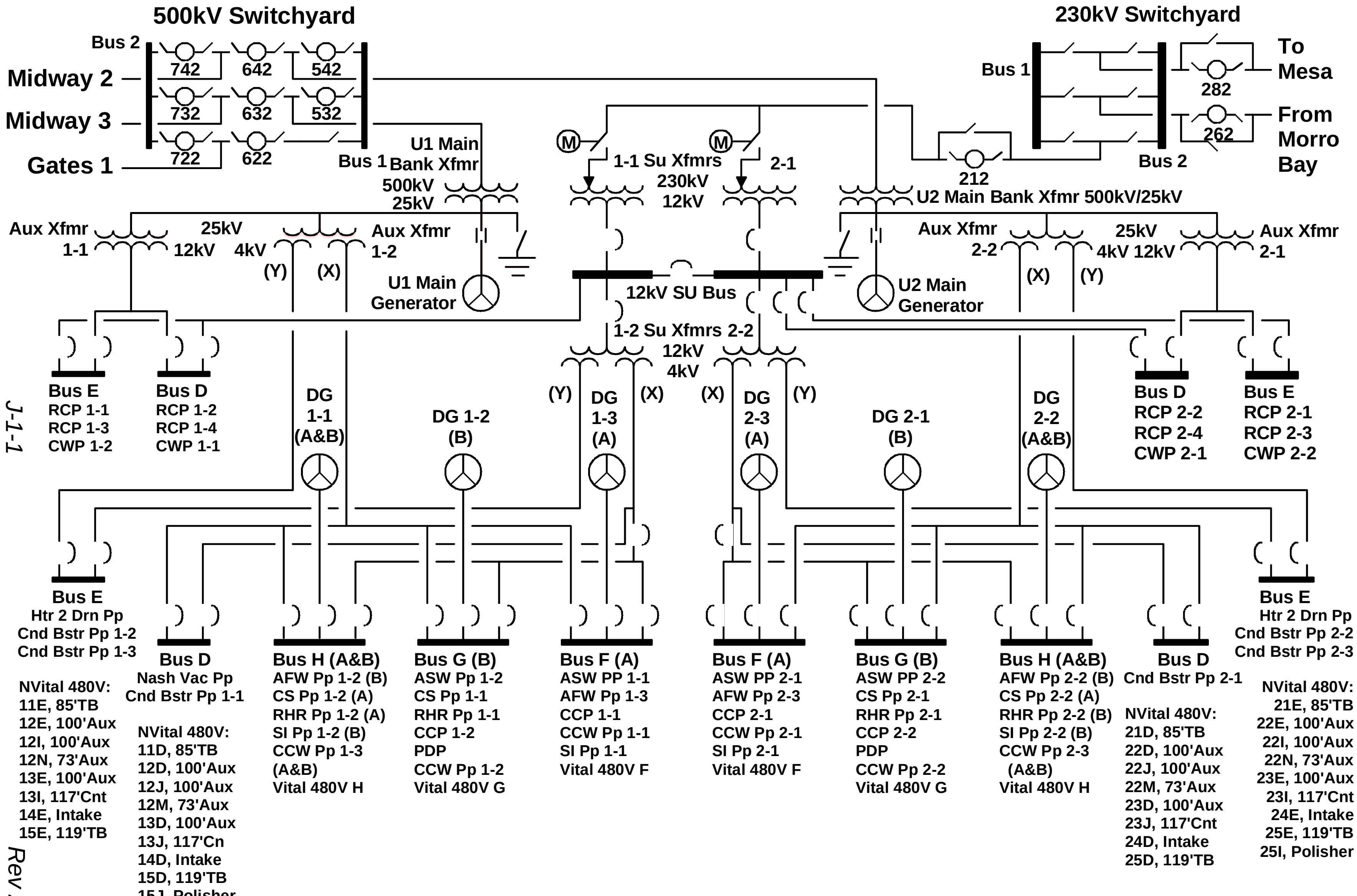
ASSOCIATED INFORMATION:

Associated objective(s):

7011	State the power supply to the RHR pumps

Reference Id: P-0648
Must appear: No
Status: Active
User Text: 7356.020122
User Number 1: 0000002.90
User Number 2: 0000003.30
Difficulty: 1.00
Time to complete: 3
Topic: Correct power supply for RHR pump.
Cross Reference: STG B-2

Electrical Distribution Overview



J-1-1

Rev 22

Note: OPI.DC38, ATT 8.2, Pg. 2 for information on ECCS/Safe Shutdown Trains

RO Question 50

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	063 K4.04	
	Importance Rating:	2.6	2.9

Proposed Question:

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

A. The breaker immediately trips open and cannot be reclosed until control power is restored.

B. The breaker can be tripped from the Control Room but automatic trip functions are not operable.

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

D. Automatic breaker trips are operable but tripping the breaker from the Control Room is not possible.

Proposed Answer:

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

Explanation:

A incorrect. The breaker will not open.

B incorrect. The breaker cannot be tripped remotely.

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

D incorrect. Automatic trips are defeated.

Technical Reference(s): STG J9, DC Power

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-1514 modified

Question History: Last NRC Exam: N/A

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

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

Comments:

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

#	1.00

WHICH ONE (1) of the following describes 12KV and 4KV breaker operation if DC control power is lost to these breakers?

A. Breakers will remain in their "as is" condition and operation would only be possible by manual means

B. Remote operation of the breakers would be available but automatic trip functions would not operate

C. Automatic breaker trips would remain operational but remote operation of breakers would not be possible

D. Breakers would trip open and operation would only be possible by manual means

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

34876	State the function of DC Control Power.

Reference Id: P-1514
Must appear: No
Status: Active
User Text: 4260.070623
User Number 1: 0000003.00
User Number 2: 0000003.40
Difficulty: 3.00
Time to complete: 3
Topic: J1 Loss of DC control power to 4KV & 12KV breakers

Abnormal Operations

Alternate system lineups
Obj 12

Refer to OP J-9:II Operating the Battery Chargers for the following operations.

Section	Title
6.1 to 6.5	Placing Vital Battery Charger ... In Service
6.6 to 6.10	Removing Vital Battery Charger ... From Service
6.11	Swapping Vital Battery Chargers
6.12 to 6.14	Placing Non-Vital Battery Charger ... In Service
6.15 to 6.18	Removing Non-Vital Battery Charger ... From Service

Malfunction effects
Obj 17

The following tables describe effects of malfunctions associated with DC system/component operation.

Effects of ...	Consequence
loss of AC power to the battery chargers on continued operation DC system.	• DC systems operation is limited to the capacity of the associated battery. • Batteries are sized to provide necessary DC to respond to a design basis accident. • Unnecessary loads should be secured to prolong battery operation.
loss of DC control power	Breakers used for 4 kV and higher voltage systems use 125 VDC for control power. The load is controlled by opening or closing the breaker using closing (CC) and trip coils (TC). • The control power is used in both control and protection schemes. It energizes the CC to CLOSE the breaker (control), and energizes the TC to OPEN the breaker (control and protection). • If the 125 VDC control power is lost, the breaker fails ‘as is’, that is, it cannot be opened or closed automatically or remotely.

Continued on next page

RO Question 51

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	064 K6.07	
	Importance Rating:	2.7	2.9

Proposed Question:

GIVEN:

- Unit 1 is at 100% power.
- A relief valve has failed open on Starting Air Receiver "A" for Diesel Generator 11.
- The leakage exceeds the capacity of the starting air compressor.
- The NO is responding to the associated alarm.

Which of the following describes the response of Diesel Generator 11 if a start signal occurs before any operator action?

A. It will start in the normal time from Starting Air Receiver "B" via all four starting air solenoids.

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

C. It will take longer to start or may NOT start because the starting air system will be depressurizing/depressurized.

D. It will take longer to start or may NOT start because the fuel rack fails to no fuel position and may not reset.

Proposed Answer:

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

Explanation:

A incorrect. Only two solenoids associated with B receiver.

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

C incorrect. Receivers are not cross tied.

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
INPO

Question History: Last NRC Exam: Salem 11/02

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

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

Comments:

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

#23185

064.k6.07 Cognitive Level: 1

Salem Unit 1 WEC 11/4/2002

Unit 1 is at 100% power. A relief valve has failed open on 1A Starting Air Receiver. The leakage exceeds the capacity of the starting air compressor. The Primary NEO is responding to the associated alarm.

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

It will start in the normal time via the two starting air solenoids associated with 1B Starting Air Receiver.

It will take longer to start or may NOT start because the starting air system will be depressurizing/depressurized.

It will take longer to start or may NOT start because the starting air system will be depressurizing/depressurized or the fuel rack may not reset

It will start in the normal time from 1B Starting Air Receiver via all four starting air solenoids.

(A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids

(A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids

(A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids

(A), (B) 1B independent of 1A; (D) each receiver feeds two solenoids

Starting Air Receivers

Purpose
Obj 11

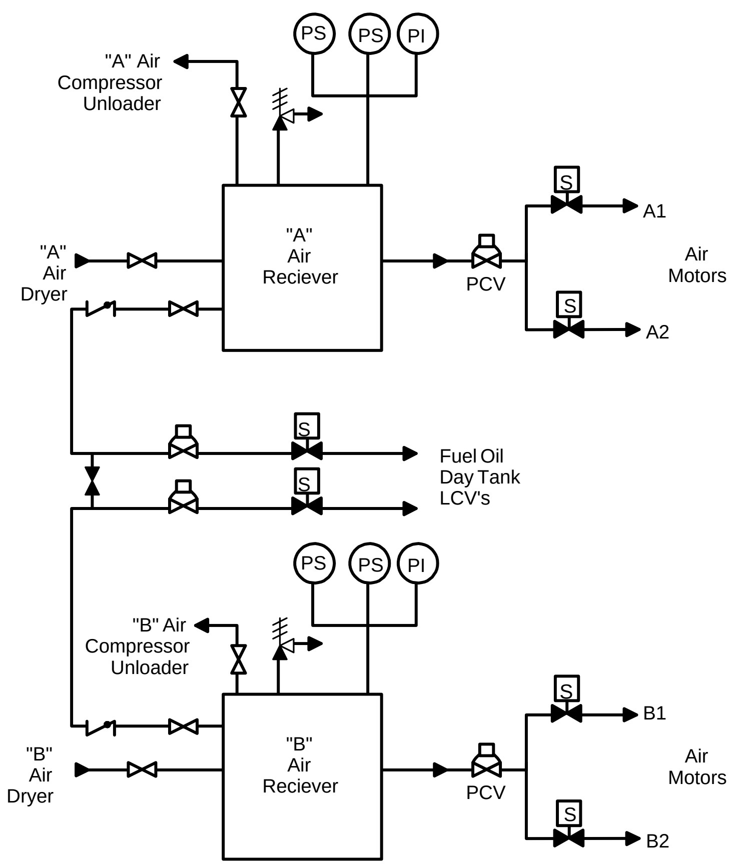
The purpose of the starting air receivers is to hold sufficient air for 45 seconds of continuous diesel cranking. Each receiver has the ability to perform 12 consecutive starts without recharging.
Each Starting Air Receiver also provides operating air for a Fuel Oil Day Tank Level control Valve.

Location
Obj 7

The starting air receivers are located on the north and south sides of the diesel engine room as illustrated on the starting air system location drawing

Physical description

Each starting air receiver has 53 cubic feet capacity. The flowpath through the starting air receivers is shown below:



DEG-49

Continued on next page

RO Question 52

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	073 K3.01	
	Importance Rating:	3.6	4.2

Proposed Question:

GIVEN:
- Reactor power is 45%.
- Circulating Water pump 11 is running. 12 is shutdown.
- A liquid radwaste discharge permit has been authorized.
- RE-18, Liquid Radwaste Monitor, is determined to be inoperable when adjusting the setpoint.

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

A. Start Circulating Water pump 12.

B. Install a temporary radiation monitor.

C. Station an operator at RCV-18 to close the valve if necessary.

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

Proposed Answer:

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

Explanation:

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-1245 modified

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments:

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

Given the following:

- A liquid radwaste discharge permit has been authorized.
- RE-18, Liquid Radwaste Monitor, was determined to be INOPERABLE when adjusting the setpoint.

WHICH ONE (1) of the following actions is correct with respect to liquid radwaste discharges?

A. Discharges are allowed if two independent samples and release rate calculations are done.

B. No discharges are allowed until RE-18 is OPERABLE.

C. A temporary monitor must be installed to perform discharges.

D. The SFM and the Chemistry and Rad Protection Foreman must jointly authorize the original discharge permit.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

8443	State the administrative requirements of Liquid Rad Waste system
69249	Explain the basic principles of operation for the Liquid Radwaste System and the major components and equipment.

Reference Id: P-1245
Must appear: No
Status: Active
User Text: 8443.130743
User Number 1: 0000001.80
User Number 2: 0000003.30
Difficulty: **2.00**
Time to complete: 2
Topic: Discharge actions when RE-18 det. INOP

6.5.2 If RE-18 is found to be inoperable proceed as follows:

****

CAUTION: Liquid Radwaste discharges may continue for up to 14 days after RE-18 is declared inoperable if the applicable ACTION statements in ECG 39.3 are followed.

****

a. Note the date and time that RE-18 is declared inoperable in the Shift Log.

b. Review the applicable portions of ECG 39.3.

c. Attach a Caution tag (Admin tagout) to the key switch for the radwaste discharge valve 0-FCV-647, made out to the SFM. Indicate on the tag the date and time of the expiration of the 14 day period when termination of batch releases are required.

d. Ensure chemistry analyzes a second independent liquid radwaste sample.

e. Ensure that the release rate calculations are verified by at least two qualified staff members.

f. If the tank activity is not below the limits stated in CAP A-5, process the liquid radwaste per chemistry instructions to reduce the tank activity.

g. Initiate an AR to initiate repair of RE-18.

h. Fill out the appropriate sections of the Discharge Permit.

6.5.3 If RE-18 has been inoperable for less than or equal to 14 days:

****

CAUTION: Liquid radwaste discharges may continue for up to 14 days after RE-18 is declared inoperable if the ACTION statements in ECG 39.3 are followed.

****

a. Review the applicable portions of ECG 39.3.

b. If RE-18 will remain inoperable longer than 14 days, go to step 6.5.4.

c. Ensure chemistry analyzes a second independent liquid radwaste sample.

d. Ensure that the release rate calculations are verified by at least two qualified staff members.

e. If the tank activity is not below the limits stated in CAP A-5, process the liquid radwaste, per chemistry instructions to reduce the tank activity.

f. Fill out the appropriate sections of the Discharge Permit.

RO Question 53

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	076 K2.01	
	Importance Rating:	2.7	2.7

Proposed Question:

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

	Pump 2-1	Pump 2-2
A.	Bus F	Bus G
B.	Bus G	Bus H
C.	Bus F	Bus H
D.	Bus G	Bus F

Proposed Answer:

A. Bus F Bus G

Explanation:

Pumps 1 and 2 are powered from Bus F and G respectively for both units.

Technical Reference(s): STG E5, ASW

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

ASW Pumps

Sensitive Information

Each pump is in a separate room with a watertight door.

Power supplies
Obj 4

The power supplies for the pump motors are:

ASW Pump	4 kV vital bus
1-1	F
1-2	G
2-1	F
2-2	G

Continued on next page

RO Question 54

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	078 A3.01	
	Importance Rating:	3.1	3.2

Proposed Question:

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

Instrument air header pressure now indicates 89 psig.

What should be the status of the instrument air compressors?

A. All compressors running and loaded.

B. Only Rotary compressors 0-5 and 0-6 running and loaded

C. Only Reciprocating compressors 0-1 through 0-4 running and loaded

D. All compressors running but only Rotary compressors 0-5 and 0-6 loaded.

Proposed Answer:

A. All compressors running and loaded.

Explanation:

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

B incorrect. 1 thru 4 started at 93 psig.

C incorrect. 5 and 6 start at 103 psig.

D incorrect. All will be running and loaded.

Technical Reference(s): STG K1, Compressed Air

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank S-1254

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Normal Operations, Continued

Normal system responses (continued)

Effect of ...	Consequence
Standby starts, IA compressors.	Compressors 0-1 through 0-4 will start if their control switch is in AUTO and: • PS-130 decreases to 93 psig, or • If any other running compressor trips. Compressor 0-5 through 0-7 will start if they are in standby, green Auto Operation light on and: • they have shutdown due to being unloaded for >20 minutes and a compressor load signal is received. or • if auto start switch is in ENABLE (ON for 0-7) and the power supply bus has lost power and power is restored

Effects of operation
Obj 19

Equipment operating effects on system indications are summarized below. Although numerous possibilities exist for the initial conditions of operating a component, the information below assumes operation from a normal configuration.

Effect of operating...	Consequence The operator should expect to see...
IA Compressor 0-1 through 0-4	compressor discharge pressure increase, oil and air temperatures increase. Instrument air header pressure should cycle from: • 98 to 105 psig with master unloader ON. • 95 to 102 psig with master unloader OFF.
IA Compressor 0-5 and 0-6	compressor discharge pressure increase, oil and air temperatures increase. Instrument air header pressure should cycle from: • 103 to 108 psig on the lead compressor, and • 100 to 104 psig on the lag compressor.

Continued on next page

1.00

If instrument air header pressure indicates 89 psig, what would be the status of the instrument air compressors?

Assume all air compressors are available and are aligned in a normal system alignment.

A. All compressors running and loaded

B. only Rotary compressors 0-5 & 0-6 running and loaded

C. only Reciprocating compressors 0-1 through 0-4 running and loaded

D. All compressors running but only Rotary compressors 0-5 & 0-6 loaded

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

68957	Explain the basic principles of operation for the Compressed Air System.
37565	Analyze Compressed Air System control logic
68959	Describe Automatic operations of the Compressed Air System.

Reference Id: S-1254
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 3.00

RO Question 55

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_1___	_____
	K/A:	103 K3.01	
	Importance Rating:	3.3	3.7

Proposed Question:

Core offload is in progress.

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

A. Both personal air lock doors are open.

B. Half the bolts are removed from the equipment hatch.

C. Welding cables are laid through the equipment hatch.

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

Proposed Answer:

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

Explanation:

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

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

C incorrect, these may be removed in short enough time (<30 minutes).

D correct, not capable of being isolated.

Technical Reference(s): TS 3.9.4

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

Learning Objective: 96971 - Identify 3.9 Technical Specification LCOs

Question Source:
Bank - INPO modified

Question History: Last NRC Exam: DCPP 10/02

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

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

Comments:

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

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch capable of being closed and held in place by four bolts;

b. One door in each air lock capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation valve.

-------------------------------NOTE--
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge and exhaust ventilation isolation valves actuates to the isolation position on an actual or simulated actuation signal.	24 months

QuestionId 22497

ExamType ILO

ExamDate 10/1/2002

AbbrevLocNam Diablo Canyon Unit 1 Cognitive Level 2

NSSSType PWR ExamLevel R

KaNumber ..103.K3.01

QuestionStem Which one of the following conditions concerning the Personnel Air Lock would exceed a Limiting Condition for Operation and require entering a Tech Spec Action Statement?

Answer The outer and inner doors are opened simultaneously for a normal transient entry into containment while in MODE 4.

Distract1 Both air lock doors fail acceptance test criteria while the plant is in MODE 6.

Distract2 Welding cables are laid through both airlock doors while the plant is in MODE 5.

Distract3 The outer door is opened for a normal transit entry into containment while in MODE 3.

RO Question 56

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	001 K3.02	
	Importance Rating:	3.4	3.5

Proposed Question:

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

Assuming no operator or automatic actions occur, when the plant stabilizes, reactor power will be ________ and average reactor coolant temperature will be ________.

A. lower, the same

B. lower, lower

C. the same, the same

D. the same, lower

Proposed Answer:

A. lower, the same

Explanation:

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

B incorrect. Temperature will not change.

C incorrect. Power will decrease.

D incorrect. Power will decrease.

Technical Reference(s): LPA12

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

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

Question Source:
Bank – F-40627

Question History: Last NRC Exam

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

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

Comments:

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

1.00

A nuclear reactor is exactly critical just below the point of adding heat when a single control rod drops into the core. Assuming no operator or automatic actions occur, when the plant stabilizes, reactor power will be ________ and average reactor coolant temperature will be ________.

A. lower; the same

B. lower; lower

C. the same; the same

D. the same; lower

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

65661	Explain reactor response to a control rod insertion.

Reference Id: F-40627
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 2.00

OP AP-12C, Dropped Control Rod

AP-12C, Scope and Symptoms

Scope
Obj 1

OP AP-12C, Dropped Control Rod, provides instructions for plant operation when a control rod becomes disengaged from its drive mechanism and drops into the core.

Symptoms or Entry Conditions
Obj 2, 8

Symptoms of dropped rod are:
- control bank rods stepping out (if in auto).
- rapid drop in T_{AVG} & power, with corresponding decrease in pressurizer level and pressure.
- rod bottom light.
- power range high rate status light (1/4 on negative rate) and PPC printout.
- alarms relevant to dropped control rod as listed in OP AP-12C (review list of alarms, as necessary).

Description of Malfunction
Obj 3

Control rods may drop for the following reasons.
- Blown stationary gripper fuse or other power failure in power cabinet.
- Failed stationary gripper coil in CRDM or open circuit to CRDM.
- High CRDM temperature causing degraded magnetic flux coupling.
- Faulty current order from slave cycler in logic cabinet or failed firing circuit in power cabinet.

Effects and Consequences of Dropped Rods
Obj 8

Some of the effects and consequences of dropped rods are:
- After power recovers when T_{AVG} decreases, a dropped rod takes on the characteristics of an excessively mispositioned rod.
- It can produce significant distortions in local core power distributions and eventual xenon changes of nearby fuel assemblies, with the potential for localized fuel damage during recovery if the rod is withdrawn too quickly or power is not reduced (depending on how long rod is dropped).
- Depending on location of dropped rod, QPTR may be severely affected and AFD affected somewhat.
- Available SDM will remain essentially constant as total power defect varies by the same amount as the worth of the dropped rod.
- However, if the rod cannot be recovered, available SDM by calculation is reduced and the RIL is raised by an amount equal to the reactivity worth of the dropped rod from the full out position.

RO Question 57

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	002 K6.07	
	Importance Rating:	2.5	2.8

Proposed Question:

Unit 2 is at 25% power.

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

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

A. Flow goes to zero. Tcold increases to approximately Thot.

B. Flow goes to zero. Thot decreases to approximately Tcold.

C. Flow goes to approximately 30%. Tcold increases to approximately Thot.

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

Proposed Answer:

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

Explanation:

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

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

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

Technical Reference(s): TH18T

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-72907, modified

Question History: Last NRC Exam: N/A

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

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

Comments:

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

P-72907# 1.00

Given the following conditions:

Unit 2 tripped from 29% power.
RCP 2-1 breaker tripped open when the busses swapped.

Which ONE of the following describes the response of T_{hot} and T_{cold} in Loop 2-1?

A. T_{hot} lowers to approximately equal T_{cold}.

B. T_{cold} rises to approximately equal T_{hot}.

C. T_{hot} rises, T_{cold} remains approximately stable.

D. T_{cold} lowers, T_{hot} remains approximately stable.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

Reference Id:	P-72907
Must appear:	No
Status:	Active
Difficulty:	3.00
Time to complete:	2
Topic:	Idle Loop Indications
Cross Reference:	LA6
Comment:	New 3/31/04

Reviewed by CNH4, 4/17/06

Changed link from LAR1 obj 6104 to STG A-6 obj 35743 since this is more of a systems / transient analysis question than a procedures question.

Ref: STG A-6 Page 3-6 (explains reverse flow in idle loop). See also TH18T Page 29 Table 18-3 for Thot and Tcold behavior if the reactor does not trip.

Stopping a Reactor Coolant Pump (RCP) with No Resultant Reactor Trip

Stopping a Reactor Coolant Pump (RCP) will normally result in a reactor trip. For this analysis, however, a reactor trip is assumed to not occur. Reactor coolant normally flows from the discharge of each RCP to the reactor vessel. The coolant exits the reactor vessel as hot coolant, enters the steam generator, then exits the steam generator as cold coolant to continue the cycle.

If an RCP is stopped, the high pressure discharge of the other three pumps forces the coolant to flow backwards through the newly idled loop. Therefore, the maximum temperature attainable, anywhere in the idle loop, is T_C of the operating loops. Thus, T_{AVG} of the idle loop can be no greater than T_C of the operating loops.

Due to the reduced backpressure of the idle loop, the flow rate in each of the operating loops increases from 100% to about 108% of rated flow. The idle loop has an equivalent flow of about 32%, in the reverse direction. Total core flow will decrease since only three RCPs are operating, and some of their flow is bypassing the core through the idle loop.

T_C in the idle loop becomes the same as the T_C of the operating loops, due to backflow. T_H in the idle loop is strongly affected by the rate at which feedwater is fed into the idle loop's steam generator, and will be less than T_C. Core ΔT will increase significantly due to core power remaining constant with reduced core flow.

The following table depicts the typical values of RCS parameters resulting from tripping one RCP from an initial power level of 60%, with no reactor trip or safety injection occurring.

Table 18-3
DCPP PLANT RESPONSE TO TRIPPING ONE RCP FROM 60% POWER

Parameter	Initial Value	Operating Loop Values	Idled Loop Values
Loop T_H	580°F	584°F	528°F
Loop T_C	544°F	541°F	541°F
Loop ΔT	36°F	43°F	–13°F
Loop T_{AVG}	562°F	562.5°F	534.5°F
Loop Flow	100%	108%	–32%
Core Flow	100%	73%	N/A
T_{SAT}	532°F	524°F	524°F

Pstm	885 psig	825 psig	825 psig
FW Flow, #/hr	2.16 M	2.76 M	0.36 M

With reactor power level maintained at 60%, and core flow reduced from 100% to 73%, the core ΔT will increase by 37% (from 36°F to 49°F). This causes the core exit temperature to increase from 580°F to 590°F. Due to this coolant mixing with the cold coolant (528°F) from the idle loop, the vessel outlet temperature will be 584°F.

The operating loops share most of the steam load previously carried by the idle loop. The steam generators in the operating loops to steam harder, causing steam pressure to decrease. The output of each operating steam generator will increase by 28%, with 8% additional RCS flow. This equates to a 19% increase in loop ΔTs (from 36°F to 43°F). The loop ΔTs do not change as much as the core ΔT due to the increased loop flow, and mixing with the idle loop return flow.

T_H in the operating loops increases (to 584°F), due to core ΔT being greater. T_C in the operating loops decreases (to 541°F), due to the increased heat removal. T_{AVG} of the operating loops, however, remains relatively constant (562.5°F compared to an initial value of 562°F). As mentioned previously, T_H in the idle loop is strongly affected by the rate at which steam generator level is allowed to recover. The T_H value of 528°F results from very closely monitored feedwater flow rates to the idle steam generator.

Also mentioned previously is the fact that the reactor normally trips when stopping a RCP from any significant power level. This is caused by a rapid reduction in steam generator level in the idle loop. If a reactor trip does not occur immediately after the RCP is stopped, another trip signal is generally initiated several minutes after the RCP is stopped. This is caused by the low flow rate of reactor coolant through the idle loop. Feedwater is fed into the idle steam generator at a reasonable rate, but is heated at a very slow rate. The steam generator level control system allows feeding the steam generator to some nominal level, then reduces the feedwater flow rate to some minimal value.

The slow heating of the water in the idle steam generator causes the water to expand, increasing the level to the hi-hi level setpoint (67%). This will cause the reactor, turbine, and main feedwater pumps to trip off the line. This may be avoided by taking manual control of the feedwater control system and feeding the steam generator very slowly.

Discounting any reactor trips associated with tripping one RCP, the most significant effects are the decreased core flow and increased core ΔT. Some control systems utilize a T_{AVG} input, but this is generally an auctioneered high T_{AVG}. Since T_{AVG} of the operating loops remains virtually unchanged, and the idle loop T_{AVG} decreases, stopping one RCP has little or no affect on these control systems.

RO Question 58

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	011 K2.02	
	Importance Rating:	3.1	3.2

Proposed Question:

Unit 2 is at 50% power in a normal power lineup.

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

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

A. Groups 1 and 2 heaters are available.

B. Groups 2 and 3 heaters are available.

C. Groups 2 and 4 heaters are available.

D. Groups 3 and 4 heaters are available.

Proposed Answer:

A. Only Groups 1 and 2 are available.

Explanation:

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

B incorrect. 3 powered from 13E.

C incorrect. 4 powered from 13E.

D incorrect. 3 and 4 powered from 13E.

Technical Reference(s): STG 4A, PPLC

Proposed references to be provided to applicants during examination: None

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

- Pressurizer heaters
- PORV Block Valves

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Pressurizer Heaters

Purpose
Obj 11

The purpose of the Pressurizer Heaters (78 electric heaters) is to maintain Pressurizer water at the saturation temperature corresponding to the desired RCS pressure.

Location

The Pressurizer Heaters are located in stainless steel sheaths that are seal welded to a heater well assembly in the lower head of the Pressurizer.

Power supplies
Obj 14

The Pressurizer Heater group power supply information is shown below. Breakers are located in buses.

Group Number	480 VAC Bus	Breaker Number	No. of Heaters	kW Capacity
Proportional group 1	13D	52-13 (23)D5	18	415
Backup group 2	13D/1G	52-13 (23)D6 52-1G(2G)-72*	21	485
Backup group 3	13E/1H	52-13E(23E)-2 52-1H(2H)-74*	21	485
Backup group 4	13E	52-13E(23E)-5	18	415

*Note: Backup vital supply.

Physical description

The Heater design characteristics are listed below.

Characteristic	Details
Total number of heaters	78 heaters in 4 groups:
Total rated capacity	1800 kW
Individual heater capacity	23.08 kW
Operating voltage, Hz	480 VAC, 60 Hz
Design pressure	2485 psig
Design temperature	680°F
Design heatup rate	55°F per hour

Continued on next page

RO Question 59

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	016 A4.02	
	Importance Rating:	2.7	2.6

Proposed Question:

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

A. PK12-10, S/G BLOWDOWN CATION CONDT'Y HI alarm in.

B. PK12-14, COND TRAY/TUBE SKT LEAK DET CONDT'Y HI alarm in.

C. Condensate pump cation conductivity >200 µS/cm - PT 19 on Tracor Westronics Secondary Chemistry Recorder.

D. Feedwater cation conductivity >0.25 µS/cm - PT 32 on Tracor Westronics Secondary Chemistry Recorder.

Proposed Answer:

D. Feedwater cation conductivity >0.25 µS/cm - PT 32 on Tracor Westronics Secondary Chemistry Recorder.

Explanation:

Only D correct.
Step 4 states the following:
Monitor for Condensate Demin Breakthrough:
Feedwater Cation Conductivity, PT 32 on Secondary Chemistry Recorder, is LESS THAN 0.25 µS/cm

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-5506

Question History: Last NRC Exam: N/A

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

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

Comments:

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

1.00

In accordance with OP AP-20, "CONDENSER TUBE LEAK," the parameter that is monitored as an indication of demineralizer (polisher) system breakthrough is:

A. Increase in feedwater cation conductivity - PT 32 on Tracor Westronics Secondary Chemistry Recorder.

B. Increase in condensate pump cation conductivity >200 µS/cm - PT 19 on Tracor Westronics Secondary Chemistry Recorder.

C. Increase in blowdown cation conductivity - PK12-10 S/G BLOWDOWN CATION CONDT'Y HI alarm in.

D. Increase in condenser tray/tube sheet leak detection conductivity - PK12-14 COND TRAY/TUBE SKT LEAK DET CONDT'Y HI alarm in.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3478	State the entry conditions for abnormal operating procedures
4976	State the actions for increasing condensate cation conductivity

Reference Id: P-5506
Must appear: No
Status: Active
User Text: 4976.050553
User Number 1: 0000003.00
User Number 2: 0000002.90
Difficulty: 2.00
Time to complete: 3
Topic: LPA20 - Demineralizer breakthrough.
Cross Reference: OP AP-20, LPA20
Comment: REF: OP AP-20 step 4.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION: FCV-230 must remain closed to maintain the Condensate Demineralizers in service. A Reactor Trip is required if FCV-230 opens for any reason.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **PREVENT Condensate Demin Bypassing:**	
a. Check FCV-230 CLOSED	a. TRIP the Reactor and GO TO EOP E-0
b. OPEN the supply breaker for FCV-230 UNIT ONE / UNIT TWO 52-15J-17 52-25I-17	
4. **MONITOR For Condensate Demin Breakthrough:** Feedwater Cation Conductivity, PT 32 on Secondary Chemistry Recorder, is LESS THAN 0.25 µS/cm	TRIP the Reactor AND GO TO EOP E-0.
5. **REDUCE Power as Required by Attachment 4.1, While Continuing On In This Procedure:**	
6. **SECURE any Liquid Radwaste Discharge in Progress in Preps for Securing a CWP**	
7. **PREVENT CST Contamination, If Necessary:** Condensate Cation Conductivity LESS THAN 15 µS/cm	1) Place HC-3 in MANUAL and CLOSE LCV-12. 2) Notify Chemistry to sample the condensate prior to rejecting the hotwell. 3) Reject the hotwell overboard per OP C-7A:III. a. Do not decrease hotwell level to LESS THAN 55 inches. b. Monitor Feedwater Suction pressure and start an additional condensate booster pump set if pressure approaches 300 psig.

RO Question 60

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	033 A3.02	
	Importance Rating:	2.9	3.1

Proposed Question:

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

A. 127 feet

B. 129 feet

C. 131 feet

D. 133 feet

Proposed Answer:

D. 133 feet

Explanation:

D correct. According to section 4.0 DESIGN FEATURES
4.3.2 Drainage
The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133 ft.

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

Proposed references to be provided to applicants during examination: None

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

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

Question Source:
DCPP Bank R-55487

Question History: Last NRC Exam: N/A

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

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

Comments:

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

The design feature of the Spent Fuel Pool will prevent inadvertant draining of the pool below elevation:

A. 125 ft.

B. 130 ft.

C. 132 ft.

D. 133 ft.

Answer: D

ASSOCIATED INFORMATION:

Associated objective(s):

9697	Identify Technical Specification LCOs
35482	Identify Tech Spec LCOs.
35694	Describe the operation of the Spent Fuel Pool Cooling system. ? Make-up ? Normal alignments ? Refueling alignments ? Malfunctions

Reference Id: R-55487
Must appear: No
Status: Active
User Text: 9555.120111
User Number 1:
User Number 2:
Difficulty: 2.00
Time to complete: 2
Topic: Design features of the spent fuel pool

4.3 Fuel Storage (continued)

4.3.2 Drainage

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

4.3.3 Capacity

The permanent spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies. For cycles 14-16, the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 154 fuel assemblies. For cycles 14-16, the total combined spent fuel pool capacity in the permanent and cask pit storage racks is limited to no more than 1478 fuel assemblies.

RO Question 61

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	034 K4.02	
	Importance Rating:	2.5	3.3

Proposed Question:

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

A. Block at motor

B. Hoist full down

C. Hoist underload

D. Upper slow zone

Proposed Answer:

C. Hoist underload

Explanation:

A incorrect, not bypassable.

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

C correct, bypassed on hoist pendant.

D incorrect, not bypassable.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Spent Fuel Pool Bridge Crane, Continued

Interlocks
Obj 4, 28

The purpose of interlocks on the spent fuel pool bridge crane are to prevent damage to spent fuel assemblies thereby preventing a radioactive release.

The table below describes the two classes of interlocks.

Class of Interlock	Prevents ...
Hoist operation	lifting a load greater that 2000 pounds (overload).
	lowering a load less that 1400 pounds (underload) • bypassed with UNDERLOAD PERMISSIVE button.
	fast speed by hoist when close to upper height limit
	lifting a load higher than the hoist is designed for.
	two blocking the hoist block and tackle (block hitting the winch).
	overextending the hoist (letting out too much cable).
Multiple actions	more than one of these operations at a time: • hoist operation or • trolley movement, or • bridge movement.

The table below summarizes the interlocks on the Spent Fuel Pool Bridge Crane.

Interlock	Is active when..	Action
Power transfer switch	not in the OFF position.	One hoist may be energized at a time.
Upper slow zone limit switch	hoist increases to 58 on readout	Shifts hoist speed to slow prior to reaching upper limit switch.
Overload on hoist	load monitor senses greater than 2000 pounds load.	• Stops hoist operation in raise direction. • Energizes overload lt.
Full up limit switch *	hoist increases to 59.7 on readout.	Stops hoist operation in raise direction.

Continued on next page

Spent Fuel Pool Bridge Crane, Continued

Interlocks (continued)

Interlock	Is active when..	Action
Block at motor limit switch	block activates limit switch at hoist motor.	
Underload on hoist (bypassed with UNDERLOAD PERMISSIVE button)	load monitor senses less than 1400 pounds load	• Stops hoist operation in lower direction. • Energizes underload lt.
Full down limit sw. *	hoist decreases to lower limit.	Stops hoist operation in lower direction.
Hoist	hoist is in operation.	Stops trolley or bridge motion.
Bridge	bridge is in operation.	Prevents trolley motion.
Trolley	trolley is in operation.	Prevents bridge motion.

* All GEMCO limit switch height interlocks may be bypassed by use of a LIMIT SWITCH BYPASS push-button located in the GEMCO control box on the bridge end support.

Operation
Obj 4, 19

The spent fuel pool bridge crane operations are described in the table below.

Step	Action
1	Close circuit breaker 52-12M-21 to provide power
2	Energize controls by closing disconnect transfer switch for the hoist to be used
3	Move the bridge and trolley to the location desired
4	Lower the hoist to pick up the desired item
5	Pick up the desired item, monitor load
6	Move the bridge and trolley to the desired location
7	Lower the hook until the load is minimized, un-hook the load.
8	Raise the hook clear

RO Question 62

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	035 A2.04	
	Importance Rating:	3.6	3.8

Proposed Question:

Unit 1 is at 50% power.

The following events occur:

- PK09-02, SG 1-2 PRESS, LVL, FLOW alarms.
- Steam flow has increased above feed flow.
- Reactor power is increasing
- It is confirmed a safety on the steam generator has failed open.

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

A. Trip the reactor.

B. Place rods in MANUAL and verify rod motion stops.

C. Reduce turbine load as necessary to maintain Tave on program.

D. Take manual control of feedwater and restore steam generator levels to program.

Proposed Answer:

A. Trip the reactor.

Explanation:

Per AR PK09-02, if a steam generator safety is open, the action is to trip the reactor.

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

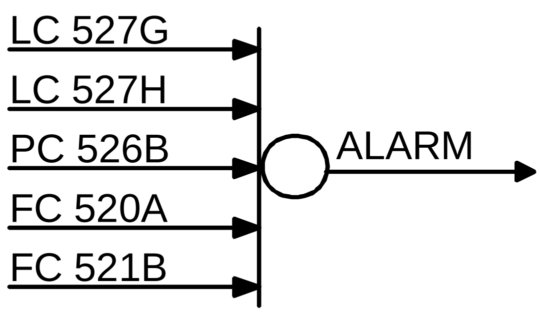
TITLE: SG 1-2 PRESS, LVL, FLOW

1

07/26/00
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. LOGIC DIAGRAM



2. ALARM INPUT DESCRIPTION

DEVICE NUMBER	ALARM INPUT	ANNUNCIATOR TYPEWRITER PRINTOUT	SETPOINT
FC 520A	354	Stm Gen 1-2 Stm Flo less than FW Flow	0.70 x 10E6 lbs/hr
PC 526B	419	Stm Gen 1-2 Press Lo	LT 650 psig
LC 527G	342	Stm Gen 1-2 Lvl Hi From Ref	+ 5%
LC 527H	343	Stm Gen 1-2 Lvl Lo From Ref	- 5%
FC 521B	735	Stm Gen 1-2 FW Flow less than Stm Flow	0.70 10E6 lbs/hr

3. PROBABLE CAUSE

   3.1 Stm. gen. level deviation.

   3.2 Stm. flow LT feedwater flow.

   3.3 Stm. gen. low pressure.

   3.4 Feedwater flow LT Stm. flow.

4. AUTOMATIC ACTIONS

   None

5. OPERATOR ACTIONS

5.1 If Reactor Trips, GO TO EOP E-0.

5.2 Perform channel check of Stm. Gen. level, press., and flow indications.

5.3 Verify steam generator water level control is performing properly in AUTO.

5.4 Take Manual control of Stm. Gen. water level and restore Stm. Gen. level and/or steam flow/feed flow as necessary.

5.5 If a safety valve has lifted and fails to reseat, trip the reactor and GO TO EOP E-0.

5.6 If a malfunction of a Stm. Gen. pressure, level, or flow channel is indicated, GO TO OP AP-5.

RO Question 63

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	045 K5.17	
	Importance Rating:	2.5	2.7

Proposed Question:

GIVEN:

- The plant is at 90% power
- Control Rods are at 228 steps
- Boron concentration is 1600 ppm.

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

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

A. The RCS temperature change would be smaller.

B. More positive reactivity would be added.

C. The power change would be smaller.

D. The power defect would be smaller.

Proposed Answer:

A. The RCS temperature change would be smaller.

Explanation:

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

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

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

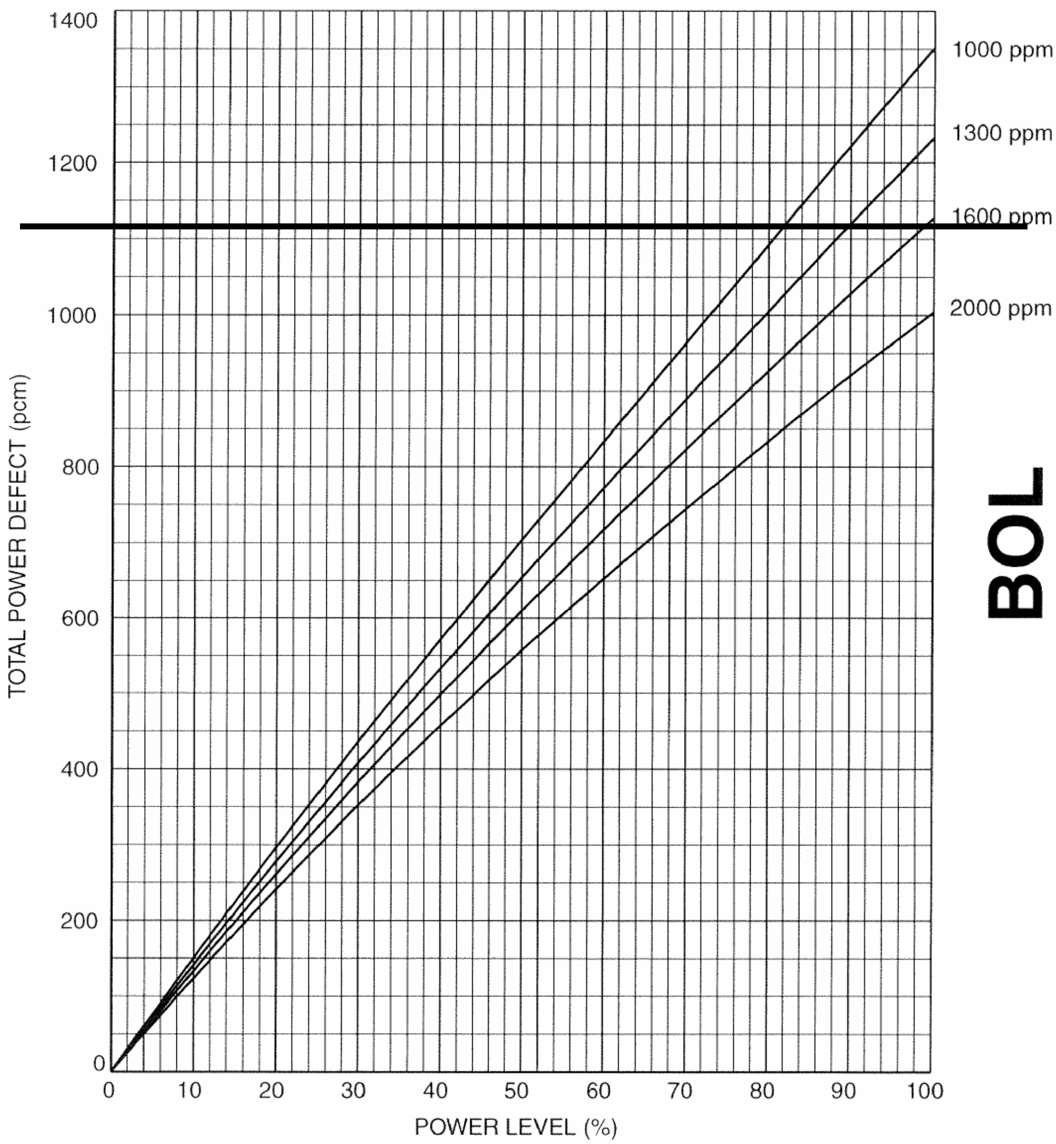
Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis X

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

Comments:

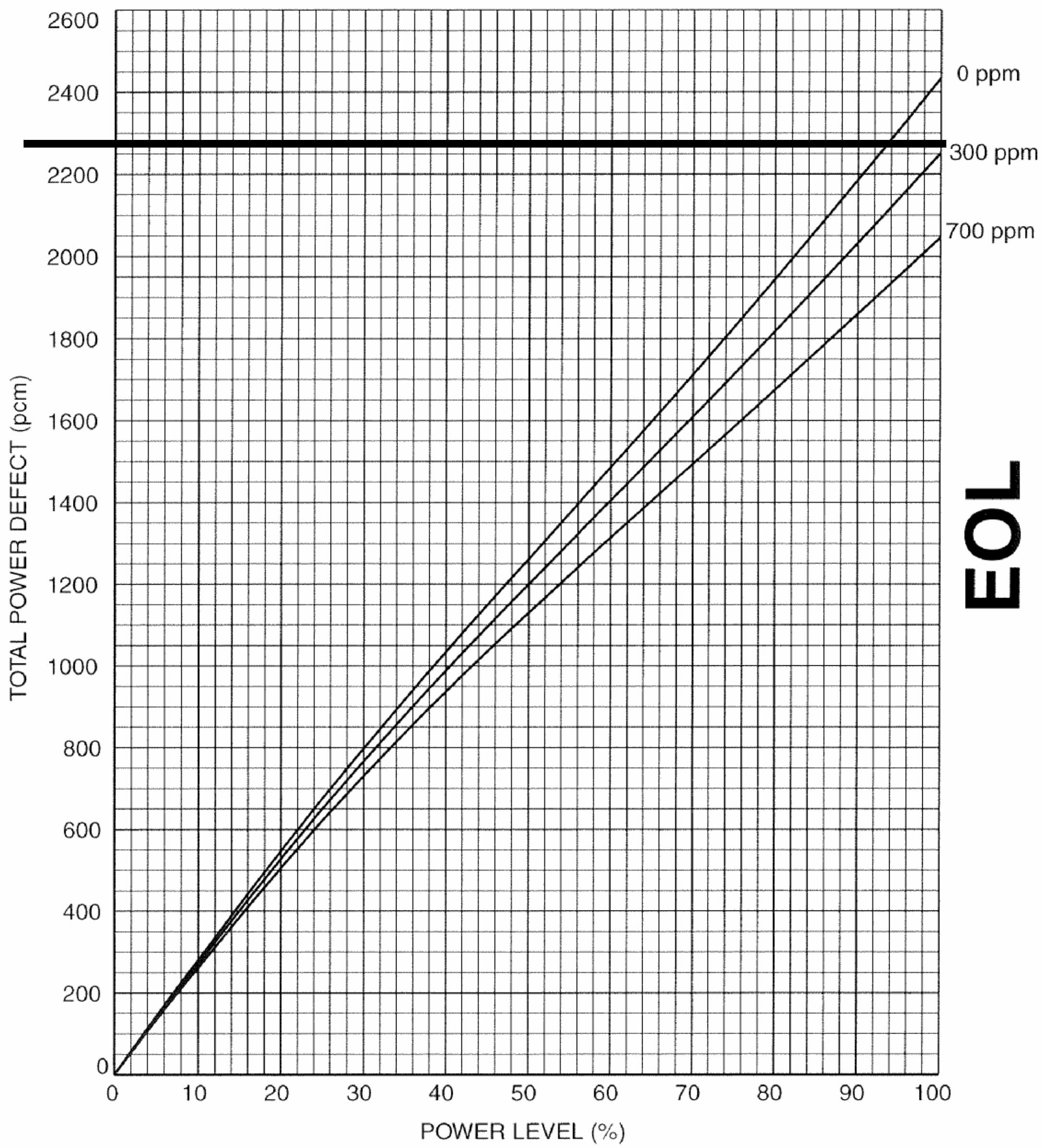
K/A: 045 K5.17 - Knowledge of the operational implications of the following concepts as the apply to the MT/B System: Relationship between moderator temperature coefficient and boron concentration in RCS as T/G load increases (2.5/2.7)

DIABLO CANYON POWER PLANT OPERATION DATA
FIGURE R17-1F-3
Total Power Defect as a Function of Power Level at BOL
Cycle 14 for Burnup 0 - 8000 MWD/MTU



SOURCE: WCAP-16466-P, Rev 0, Figure D.4-1

DIABLO CANYON POWER PLANT OPERATION DATA
FIGURE R17-1F-3
Total Power Defect as a Function of Power Level at EOL
Cycle 14 for Burnup 16000 MWD/MTU - EOL



SOURCE: WCAP-16466-P, Rev 0, Figure D.4-3

RO Question 64

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	068 G2.4.49	
	Importance Rating:	4.0	4.0

Proposed Question:

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

The discharge:

A. must be immediately secured.

B. may continue unchanged.

C. must immediately be reduced.

D. must be diverted to the discharge of the other unit.

Proposed Answer:

A. must be immediately secured.

Explanation:

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank NS-54684

Question History: Last NRC Exam

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

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

Comments:

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

1.00

The liquid radwaste system is discharging to the ocean when a reactor trip occurs. The discharge flow:

A. must be terminated.

B. may continue unchanged.

C. must be reduced.

D. must be diverted to the other Unit's discharge path.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

69249	Explain the basic principles of operation for the Liquid Radwaste System and the major components and equipment.
4800	Explain liquid rad waste system operation

Reference Id: NS-54684
Must appear: No
Status: Active
User Text: 4800
User Number 1:
User Number 2:
Difficulty: 3.00

NOTE: Chemistry chart used in the following step can be found in EDMS under "NPG Library, Chemistry, Chemistry S/U & S/D".

- Adjust chemical injection for current total AFW flowrate, Chemistry instructions and the Chemistry chart. (See Attachment 9.3)
- Adjust make-up water flow to the CST & TT per the Control Room.
- Channel check FR-53 and RM-23.
- For a Unit 1 reactor trip, notify SCARP that Zinc Injection has been secured and Argon Gas Injection will need to be isolated.

6.4 AUXILIARY BUILDING SENIOR WATCH

6.4.1 Immediate Actions for Auxiliary Building Senior Watch

- **Secure any liquid or gaseous radwaste discharges in progress.**
- Select Boric Acid Evaporator to RECYCLE if in service. Manually isolate Auxiliary Steam to the Evaporator Preheater and Shell by closing AXS-39.
- Check the system's status as a bus transfer may have caused the fluidic logic to change its alignment.
- If "Containment Isolation" alarm is in, secure RCDT Pps, Containment Structure Sump Pps and Rx Cavity Sump Pps. Log pump status in Aux Senior Daily Log and Abnormal Status board. Notify SFM of pump status as soon as plant conditions stabilize.

6.4.2 SUBSEQUENT ACTIONS FOR AUXILIARY BUILDING SENIOR WATCH

None

6.5 INTAKE/OUTSIDE WATCH

6.5.1 Immediate Actions for Intake/Outside Watch

- Performing a normal shutdown of the CW Bay Chlorination Skid. Reference OP E-3:II as necessary.
- If less than two of the four CWPs remain in service, shutdown all ASW continuous chlorination.
- Verify Intake Cooling Pump in service to running CWP.
- Check status of CWP that tripped.
- Walkdown the intake to check for other abnormalities, shutdown any unnecessary equipment that auto started.
- **Secure overboard discharge of WHAT Tank if in progress.**

RO Question 65

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_2___	_____
	Group:	_2___	_____
	K/A:	071 A1.06	
	Importance Rating:	2.5	2.8

Proposed Question:

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

A. PCV-94, 95 and 96 (GDT'S fill valves) will close.

B. FCV-407, 408, 409 (GDT'S purge outlet header isolations) will close.

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

D. FCV-410 (GDT'S Outlet Header to Plant Vent) will close.

Proposed Answer:

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

Explanation:

A incorrect. No auto closure on high radiation.

B incorrect. No auto closure on high radiation.

C correct, only RCV-17 closes (and is prevented from opening).

D incorrect. No auto closure

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

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

- Control switches
- Interlocks and logic
- Automatic actions

Question Source:

Bank R-52308

Question History: Last NRC Exam: N/A

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

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

Comments:

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

Which one of the following actions will automatically occur if the gaseous radwaste discharge header radiation monitor (RE-22) detects high radiation?

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

B. FCV-410 (GDT'S Outlet Header to Plant Vent) will close.

C. Gas decay tank vent valves will close.

D. Gas decay tank fill valves will close.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

37706	Describe the operation of the gaseous radwaste vent valves, including: ? Control switches ? Interlocks and logic ? Automatic actions
69285	Describe Gaseous Radwaste System automatic features.

Reference Id: R-52308
Must appear: No
Status: Active
User Text: 7425.110711
User Number 1: 0000002.10
User Number 2: 0000003.40
Difficulty: 1.00

GDT Outlet HDR to Plant Vent FCV-410, Continued

Interlocks
Obj 9, 12

FCV-410 has an administrative control with a mechanical key interlock for its switch in order to open and close the valve.

Indications
Obj 9

The following indications are available for FCV-410 in the Aux. Building on the Aux. Control Board, 85′ elevation (Unit 2 is the same):

FCV-410 Key Control Switch		
Indicating Lights	**Meaning**	**Normal Status**
Red	Valve is OPEN	OFF/ON
Green	Valve is CLOSED	ON/OFF

Alarm

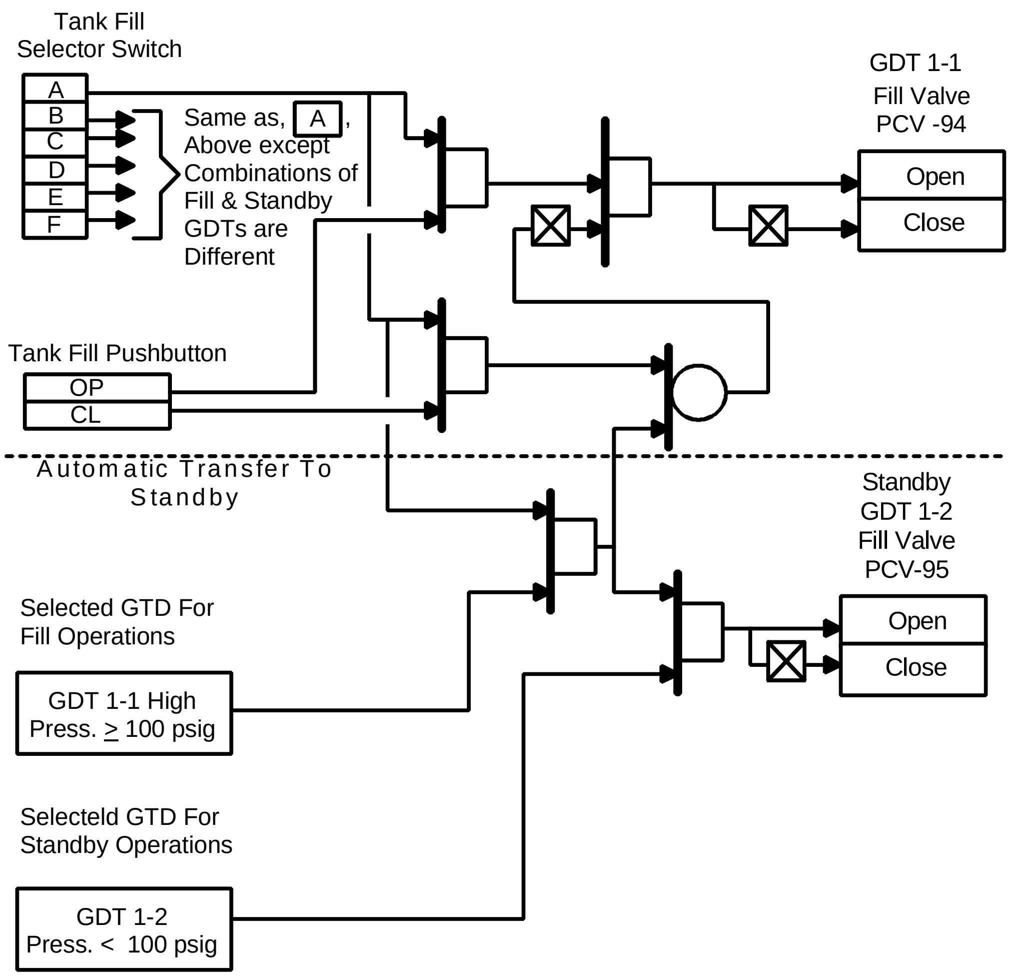
The following alarm exists for the vent header down stream of FCV-410 and FCV-101 as described below.

Parameter	**Source**	**Location**
HIGH vent pressure	PS-183	Aux. Board

* This condition is common when FCV-410 is first opened until PCV-101 responds. If the alarm persists then the operator should close FCV-410 and remove the key.

Gas Decay Tank Fill Valves PCV-94, 95, 96, Continued

Logic diagram The following diagram represents a typical logic for the selected GDT for fill, standby, and automatic transfer operations.



WGS-17

Interlocks

- The positions on the Tank Fill Selector switch allow only one GDT to be filled at a time.
- The interlocks prevent the GDT selected for fill from being vented or purged.
- The standby tank can be purged while being held in standby.

Continued on next page

GDT to Plant Vent Valve RCV-17/RE-22, Continued

Controls
Obj 9, 12

Each unit's valve RCV-17 is controlled by its respective RE-22. RCV-17 will open, and remains open whenever:
- radiation monitor RE-22 is reset, and
- there is no high radiation alarm actuated.

Automatic Actions
Obj 11, 12

The automatic actions and function of RCV-17 are described below.

Automatic actions IF...	Function THEN...
High radiation setpoint on RE-22 is exceeded: • RCV-17 automatically CLOSES or • RCV-17 is prevented from OPENING.	The dose, at any time, at and beyond the site boundary, from gaseous effluents will be within the prescribed dose limits of 10CFR20, Appendix B.
RCV-17 fails closed on loss of instrument air.	It ensures that an uncontrolled or unmonitored release is prevented.

Logic

The control logic for RCV-17 is described in the table below. RE-22 is located in area K at 54′ elevation of the Auxiliary Building.

Control	Function
The setpoint on RE-22 is variable.	It is set prior to each discharge to ensure that the administrative limit for off site dosage is not exceeded. • Whenever the setpoint is exceeded, the alarm must be reset from the Radiation Monitoring System control panel to reopen RCV-17

Interlocks
Obj 12

RE-22 must be reset for RCV-17 to open.

Continued on next page

GDT Outlet HDR to Plant Vent FCV-410, Continued

Interlocks
Obj 9, 12

FCV-410 has an administrative control with a mechanical key interlock for its switch in order to open and close the valve.

Indications
Obj 9

The following indications are available for FCV-410 in the Aux. Building on the Aux. Control Board, 85′ elevation (Unit 2 is the same):

FCV-410 Key Control Switch		
Indicating Lights	**Meaning**	**Normal Status**
Red	Valve is OPEN	OFF/ON
Green	Valve is CLOSED	ON/OFF

Alarm

The following alarm exists for the vent header down stream of FCV-410 and FCV-101 as described below.

Parameter	Source	Location
HIGH vent pressure	PS-183	Aux. Board

* This condition is common when FCV-410 is first opened until PCV-101 responds. If the alarm persists then the operator should close FCV-410 and remove the key.

RO Question 66

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_1___	_____
	K/A:	G2.1.9	
	Importance Rating:	2.5	4.0

Proposed Question:

GIVEN:

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

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

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

A. Ask the Unit 2 CO.

B. Locate and ask the Unit 2 Shift Foreman.

C. Go to the Shift Manager for permission.

D. Enter the area; an operator with a current license may enter the area at any time.

Proposed Answer:

A. Ask the Unit 2 CO.

Explanation:

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

B incorrect, SFM permission not required.

C incorrect, the CO may grant permission.

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New
New

Question History: Last NRC Exam: N/A

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

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

Comments:

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

PACIFIC GAS AND ELECTRIC COMPANY
OPERATIONS SECTION POLICY

Number: A-21
Revision: 6
Date: 02/09/05

DIABLO CANYON POWER PLANT UNITS 1 AND 2

TITLE: **CONTROL ROOM PROTOCOL**

APPROVED: ______________________________
Operations Manager

SCOPE

This policy establishes the basic expectations for the conduct of personnel in the control room. The intent is to minimize distractions to the control room watchstanders and to allow them to focus on plant parameters. The expectation is that all activities performed in the control room are conducted in a professional and businesslike manner. The on watch Operations shift team is responsible for the appearance and formality of the control room.

AUTHORITY

The Shift Manager shall implement this policy and is responsible for the conduct of all personnel in the control room. Additionally, the Shift Manager and the Shift Foremen are responsible for implementing any additional requirements consistent with the spirit of this policy.

REQUIREMENTS

The following list is intended to establish a formal control room environment. It is not intended to be all-inclusive. Personnel in the control room are expected to comply with the spirit of this policy for all areas not specifically covered.

1. ***A formal request to enter the "at the controls" (see OP1.DC12 for details) area should be made by everyone except the ON WATCH operators and Control Room Assistant. The request should be made to the SFM. If the SFM is not available, the BOPCO should be asked. The CO may be asked if no other licensed operator is available on that unit.***

   Other on shift Operations personnel (OST and Asset Team operators, WWMs, etc.) may enter the elevated area to use or file OVIDs and procedures. This is not intended to be a short cut to the SFM or to be used as a passageway to the kitchen area.

RO Question 67

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_1___	_____
	K/A:	G2.1.20	
	Importance Rating:	4.3	4.2

Proposed Question:

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

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

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

The procedural response to this situation was:

A. Correct, FR-S.1, Step 1, RNO instructs the operator to de-energize 480v buses 13D and 13E.

B. Correct, direct entry in FR-S.1 is allowed because the reactor failed to automatically trip.

C. Incorrect, E-0, Step 1, should have been performed prior to going to FR-S.1.

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

Proposed Answer:

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

Explanation:

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

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

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

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-68832

Question History: Last NRC Exam

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

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

Comments:

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

While operating at 100% power, a valid reactor trip signal is received. The REACTOR TRIP INITIATED alarm is in, but the REACTOR TRIP ACTUATED alarm is NOT in, the reactor did not automatically trip.

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

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

The procedural response to this situation was:

A. Incorrect, since E-0 should be continued once the reactor has been tripped in Step 1, RNO.

B. Incorrect, since E-0 , Step 1, should have been performed prior to going to FR-S.1.

C. Correct, direct entry in FR-S.1 is allowed based on the indications.

D. Correct, since FR-S.1, Step 1, RNO states "Manually trip reactor."

Answer: A

ASSOCIATED INFORMATION:
Associated objective(s):

38107	Apply the Rules of Usage in EOPs for the CSFSTs and FRGs, including: the six status trees the priority of use of the status trees the priority of use of the color of each CSF when to monitor and/or implement the CSFSTs and FRGs

Reference Id: P-68832
Must appear: No
Status: Active
Difficulty: 3.00
Time to complete: 2
Topic: Ability to execute procedural steps\

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. VERIFY Reactor Trip:	Manually Trip Reactor
a. Reactor trip and bypass breakers - OPEN	IF The reactor trip breakers are closed,
b. Reactor Power - DECREASING	THEN Manually de-energize 480V Buses 13D and 13E (52-HD-13 and 52-HE-4).
c. DRPI indicates	
1) Rods - FULLY INSERTED	IF Reactor will NOT trip,
2) Rod Bottom lights – LIT	THEN GO TO FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION / ATWS
	WHEN The reactor trips,
	THEN Perform the following:
	1) Re-energize 480V Buses 13D and 13E (52-HD-13 and 52-HE-4).
	2) Locally open the reactor trip breakers.
	- -
2. VERIFY Turbine Trip:	
a. All Turb Stm Stop Vlvs - CLOSED	a. Manually Trip Turbine,
	IF At Least one Stop vlv OR Governor vlv in each lead is NOT Closed
	THEN Close MSIVs and MSIV Bypass vlvs.
	- -

RO Question 68

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	______
	Group:	_2___	______
	K/A:	G2.2.12	
	Importance Rating:	3.0	3.4

Proposed Question:

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

What type of surveillance is this considered?

A. Periodic

B. Recurring

C. Conditional

D. Mode Specific

Proposed Answer:

C. Conditional

Explanation:

Only C correct. Per AD13.DC1 - Conditional Surveillance
Any surveillance that is performed on other than the normal periodic time based surveillance interval. Conditional surveillances may be required for events such as inoperable alarms and monitors or for changing plant conditions such as entering a new mode.

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source:
Bank P-69640

Question History: Last NRC Exam: N/A

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

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

Comments:

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

1.00

A surveillance is required only because the unit transitioned to another mode. This is therefore a __________ surveillance.

A. conditional

B. periodic

C. mode-specific

D. recurring

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3588	Identify the definitions associated with the surveillance testing and inspection program

Reference Id: P-69640
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 1.00

2.4 The RT W/O module of PIMS allows the plant to keep track of testing activities that are to be performed on a recurring basis. Next due dates are generated based on the last test performance date and the test frequency. The library W/O contains the permanent record of administrative requirements for a given test, for example, the test title and procedure number, which organization performs the test, who is responsible for the procedure, which plant modes the test is required to be current, the test frequency and reference to applicable license requirements. The active W/O is cloned from the library W/O and is used to plan, schedule, assign and update the satisfactory performance of the test. Each test performance will have its history documented on an individual active W/O.

2.5 Technical Specification required surveillance requirements can be categorized as conditional or periodic.

2.5.1 This procedure only applies to conditional surveillances required to be performed by Tech Spec surveillance requirements. They are to be specifically referenced in a controlling procedure such as an operating, abnormal, emergency, equipment or function control procedure.

2.5.2 All periodic surveillances conducted at a frequency less than seven days are controlled by shift turnover lists, or other controlling procedures. All periodic surveillances required on a frequency of seven days or greater are controlled by the master surveillance schedule and are scheduled by designated test coordinators for each section.

2.6 Sections other than the responsible section are often required to assist and support in the conduct or analysis of a surveillance procedure. Surveillance tests and inspections often refer to other procedures which are required to accomplish the test. Assistance and support requirements shall be outlined within the controlling surveillance procedure.

3. DEFINITIONS

3.1 Conditional Surveillance

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

3.2 Periodic Surveillance

A surveillance conducted on a routine surveillance interval.

3.3 Recurring Task Scheduler (RTS)

A computerized scheduling program is used to track accomplishments of the STPs in the master surveillance schedule. This program is the PIMS recurring task scheduler (RTS) and is used for tracking STPs with a test frequency of 7 days or greater.

3.4 Master Surveillance Schedule

The data in RTS that defines the surveillance test program. The data includes frequencies, license requirements, required in "MODE", work "MODE", last performed date, next due date and Tech Spec date. When a procedure step refers to updating the master surveillance schedule, it refers to the act of updating the activity completion data or "taking a work order activity to ACTCMP."

RO Question 69

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_2___	_____
	K/A:	G2.2.23	
	Importance Rating:	2.6	3.8

Proposed Question:

Unit 1 is at full power.

PK15-21, PPC, alarms. The operator reports both computers are down.

Which of the following surveillances must be logged by the operator every hour to satisfy a Technical Specification or ECG?

A. AFD

B. Rod height

C. DRPI and group step demand

D. Primary to Secondary tube leakage

Proposed Answer:

A. AFD

Explanation:

A correct. AFD is logged every hour.

B incorrect. Logged every 4 hours.

C incorrect. Logged every 4 hours.

D incorrect. Logged every 4 hours.

Technical Reference(s): AR PK15-21, PPC

Proposed references to be provided to applicants during examination: None

Learning Objective: 66066 - Discuss the requirements of System 37 ECGs.
4883 - State admin requirements associated with PPC operations

Question Source:
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:

55.41.10 - Administrative, normal, abnormal, and emergency operating procedures for the facility.
55.43.2 - Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments:

K/A: G2.2.23 – Ability to track limiting conditions for operations. (2.6/3.8)

1

TITLE: OOS PPC Data Sheet

PPC OOS Time: ____________________ Date: ____________________________

This attachment is used to manually log TS required data normally monitored by the PPC. This form is set up to log data for a 24 hour period.

When completed, attach this data sheet to the shift log.

1 HOUR surveillance required per ECG 37.2: For each operable excore channel, check AFD within limits of COLR when ≥50% RTP.

DATE	TIME	INIT	STATUS	DATE	TIME	INIT	STATUS
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT
			SAT/UNSAT				SAT/UNSAT

Remarks: ____

____

____

4 HOUR surveillance required:

1. Perform Attachment 9.1 of OP O-4, "Primary to Secondary Steam Generator Tube Leak Detection." Refer to ECG 2.1.

2. Verify and log that DRPI and group demand agree within 12 steps. Refer to FSAR 15.2.3.1.

DATE	TIME	INIT	S/D Banks (SAT/UNSAT)				Control Banks (SAT/UNSAT)			
			SB A	SB B	SB C	SB D	CB A	CB B	CB C	CB D

3. Verify that individual rod positions are within insertion limits specified in the COLR. Refer to TS 3.1.6. and FSAR 15.2.3.1.(T36105)

DATE	TIME	INIT	S/D Banks (SAT/UNSAT)				Control Banks (SAT/UNSAT)			
			SB A	SB B	SB C	SB D	CB A	CB B	CB C	CB D

Remarks: ____

RO Question 70

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_2___	_____
	K/A:	G2.2.34	
	Importance Rating:	2.8	3.4

Proposed Question:

The actual critical rod position, at MOL, will be HIGHER than the estimated critical position (ECP) if the:

A. Startup occurs 18 hours after a trip from 50% power instead of 16 hours.

B. Steam dump controller is set at 990 psig instead of the normal 1005 psig.

C. Actual boron concentration is 10 ppm lower than that recorded on the ECP.

D. Power defect was recorded as a trip from full power instead of 50% power.

Proposed Answer:

D. Power defect was recorded as a trip from full power instead of 50% power.

Explanation:

A incorrect. 2 hours later Xenon is decaying, adding positive reactivity. Result is lower critical rod height.

B incorrect. Steam dump controller at a lower temperature will lower RCS temperature. Lower critical rod height.

C incorrect. Less boron – positive reactivity, lower critical rod height.

D correct. Less power defect results in insufficient compensation for reactivity. Critical rod height will be higher.

Technical Reference(s): RC17T, Estimated Critical Position

Proposed references to be provided to applicants during examination: None

Learning Objective: 12429 - DESCRIBE how independent changes in each of the four reactivity parameters will affect the results of the ECP calculation

Question Source:
Bank F-1832

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.43.1 - Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

55.43.6 - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Comments:

K/A: G2.2.34 – Knowledge of the process for determining the internal and external effects on core reactivity (2.8/3.4)

1.00

The actual critical rod position, at MOL, will be higher than the estimated critical position (ECP) if the:

A. power defect was recorded as a trip from full power instead of 50% power.

B. steam dump controller is set at 990 psig instead of the normal 1005 psig.

C. actual boron concentration is 10 ppm lower than that recorded on the ECP.

D. startup occurs 18 hours after a trip from 50% power instead of 16 hours.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

12429	DESCRIBE how independent changes in each of the four reactivity parameters will affect the results of the ECP calculation

Reference Id: F-1832
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 2.00

1.2 EFFECTS OF REACTIVITY PARAMETER CHANGES ON THE ECP

The ECP calculation evaluates the changes that have occurred in each of the four reactivity parameters, then determines the boron concentration required to make the reactor critical at the specified control rod position. This section discusses the effect on the ECP calculation as a result of additional changes in the each of the four reactivity parameters, or as a result of common calculation errors associated with the four reactivity parameters. Two of the reactivity parameters, xenon and samarium, are time-dependent. Time factors are discussed for these parameters.

A very important concept is that the net change in reactivity, associated with any parameter, is determined by subtracting the reference reactivity value from the critical reactivity value. In other words, the net change in reactivity is the final reactivity value minus the initial reactivity value. Reversing the order reverses the sign of the result, leading to an erroneous calculation. The actual ECP calculation is discussed in detail in Section 2.1 of this chapter.

Power Defect

While entering reactivity data for an ECP calculation, an incorrect middle-of-cycle (MOC) reference power defect value of –1600 pcm is used instead of the correct end-of-cycle (EOC) value of –2400 pcm. Using the smaller negative value of reference power defect equates to a less positive total net reactivity change calculation.

The result of this error is that the calculated critical boron concentration is lower than that required to offset the actual net reactivity change. More important than the calculation error is the fact that the reactor would achieve criticality at a control rod position lower than the specified control rod position.

RO Question 71

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_3___	_____
	K/A:	G2.3.1	
	Importance Rating:	2.6	3.0

Proposed Question:

10CFR20 limits the radiation exposure (dose) to a qualified radiation worker to ______ per year.

Diablo Canyon guideline limits the radiation dose to a qualified worker to _____ per year without extension.

	10CFR20 Limit	Diablo Canyon Guideline
A.	5000 mrem	2000 mrem
B.	5000 mrem	4500 mrem
C.	3000 mrem	2000 mrem
D.	3000 mrem	2700 mrem

Proposed Answer:

A. 5000 mrem 2000 mrem

Explanation:

A correct. Per RP1.ID6, admin guideline is 2 rem. Federal limit is 5 rem.

B incorrect. 4500 is 90% of the Federal limit and this is the administrative limit. Anything above the guideline requires permission.

C incorrect. 3 rem is old Federal limt.

D incorrect. 3 rem is old Federal limit. 2700 is 90% of this “old” limit.

Technical Reference(s): RP1.ID6, attachment 8.1

Proposed references to be provided to applicants during examination: None

Learning Objective: GET

Question Source:
INPO - modified

Question History: Last NRC Exam: Kewaunee 2/06

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

10 CFR Part 55 Content: 55.41.12 - Radiological safety principles and procedures.
Comments:

K/A: G2.3.1 – Knowledge of 10 CFR: 20 and related facility radiation control requirements. (2.6/3.0)

5.12 Additional Exposure Authorization

5.12.1 Annual Administrative Limit for whole body radiation exposure is 4.5 rem per year. In unusual cases, plant management may wish to permit exposures up to, but not exceeding, the Federal Limit (5 rem per year).

a. To exceed an Administrative Limit, Attachment 8.3, "Additional Exposure Authorization for Administrative Limits," should be initiated by the requesting organization and approved per procedure.

b. The person initiating this form should be the Supervisor of the worker for whom the extension is being requested.

5.12.2 Annual Administrative Guideline for whole body exposure to radiation is 2 rem per year. This guideline may be extended up to the Administrative Limit of 4.5 rem per year.

a. To exceed an Administrative Guideline at the plant, Attachment 8.2, "Additional Exposure Authorization for Administrative Guidelines," should be initiated by the requesting organization and approved per procedure.

b. The person initiating this form should be the Supervisor of the worker for whom the extension is being requested.

5.12.3 Lifetime Guideline: For a worker who has exceeded the Lifetime Guideline (numerical value, in rem, of an individual's age, in years), an extension is required for the worker to receive greater than 1 rem/yr at the plant for contractors OR 1 rem/yr total for company employees.

5.13 INSTRUCTIONS / GUIDANCE for completing *PAGE 1*, Attachment 8.2 and 8.3:

"Additional Exposure Authorization for Administrative Guidelines" - Attachment 8.2.

"Additional Exposure Authorization for Administrative Limits" - Attachment 8.3.

NOTE: The requesting organization should complete Page 1 before asking the worker to read, sign and date the Employee Acknowledgment:

5.13.1 Worker information: Requester to provide worker information.

5.13.2 New Exposure Guideline Requested: Requester to provide requested exposure limit.

TITLE: Permissible Levels of Exposure

1. Annual Limits and Guidelines

Part of Body Irradiated	Federal Limit	Administrative Limit	Administrative Guideline
Whole body - head, trunk, gonad, arms above the elbow and legs above the knee - (rem TEDE)	5.0	4.5	2.0
Skin & any Extremity - (rem SDE)	50	45	N/A
Lens of the eye - (rem LDE)	15	13.5	N/A
Any individual organ or tissue other than lens of the eye - (rem TODE)	50	45	N/A

2. **Five Year Guideline**

The whole body exposure (TEDE) for a five year period should not exceed 10 rem. This is implemented as the 2 rem per year Administrative Guideline.

3. **Lifetime Guideline**

Total accumulated lifetime occupational exposure (TEDE) should not exceed the numerical value (in rem) of an individual's age (in years).

NOTES 1: Individuals who exceed the Lifetime Guideline are restricted to an additional exposure of 1 rem / year.

NOTES 2: In-processing personnel may be restricted to lower limits until their records are complete.

NOTES 3: Declared Pregnant Workers and Minors are restricted to 10% of the Federal Limits listed above.

DEFINITIONS:	
TEDE	Total Effective Dose Equivalent which is the summed total of the Deep Dose Equivalent (external whole body exposure) and Committed Effective Dose Equivalent.
SDE	Shallow Dose Equivalent to the skin and extremities at a depth of 0.007 cm
LDE	Eye Dose Equivalent to the lens of the eye at a depth of 0.3 cm
TODE	Total Organ Dose Equivalent which is the summed total of the Deep Dose Equivalent (external whole body exposure) and the Committed Dose Equivalent (internal dose to the organs or tissues from internal exposure from non-stochastic ALI), when CDE is not zero.

NUCLEAR POWER GENERATION
RP1.ID6
ATTACHMENT 8.2

TITLE: Additional Exposure Authorization for Administrative Guidelines

This page should be completed by the Requesting Organization and forwarded to Dosimetry:

Individual's Name: ______________________ SSN: ____/____/____

Department/Company: ______________________ Time/Date: ______/______

New Exposure Guideline Requested: (Enter new guidelines requested)

Annual Administrative Guideline - Annual radiation worker exposure guideline is 2 rem (TEDE) per calendar year. The guideline is extendible to 4.5 rem. Request: ____________

_____: **Request lowered by plant ALARA personnel** Allowed: ____________

Lifetime Administrative Guideline - Lifetime exposure is greater than age in years and individual will receive greater than 1 rem/yr at DCPP. The guideline is extendible beyond this exposure, when required. Request: ____________

_____: **Request lowered by plant ALARA personnel** Allowed: ____________

Justification: Discuss below (attach additional pages, if needed), with adequate detail, why no one else can reasonably perform the task(s) for which the additional exposure is being requested. Additionally, provide information to justify the amount of the exposure being requested and what actions will be taken on the job to limit the individual's exposure to ALARA (i.e., restrictions to performing lower exposure tasks).

ALARA REVIEW:

[ ] Denied [ ] Acceptable for further processing: ______________________/____________
(RP Engineering/Supervision) (Date)

Employee Acknowledgment

I acknowledge that I have been informed of a request for additional exposure authorization above the normal guidelines established for the plant and have reviewed the above justification and actions to be taken to restrict my exposure to as low as reasonably achievable. Further, I have reviewed my radiation exposure history and certify that I have received no other occupational exposure at another facility since the beginning of this work assignment at the plant (history information in Dosimetry should be complete).

Signature: ______________________ Time/Date: ____________/____________

30401	2/2/2006	PWR
Kewaunee Unit 1	1	

10CFR20 limits the radiation exposure (dose) to a qualified radiation worker to _______ per year.

Kewaunee Power Station limits the radiation dose to a qualified radiation worker to _______ per year WITHOUT special authorization.

10CFR20 limit	KPS limit
5000 mrem	2000 mrem
3000 mrem	1200 mrem

3000 mrem corresponds to the second line administrative limit (the limit employed after first level of approval, at 60% of the NRC normal limits). Previous 10CFR20 limits also used to specify 3 rem/qtr limit. 1500 mrem correspond to the typical first line limit employed at 40% of the NRC allowed limit.

3000 mrem	1800 mrem

3000 mrem corresponds to the 2nd stage limit (the limit employed after first level of approval). Previous 10CFR20 limits also used to specify 3 rem/qtr limit. 1800 mrem correspond to the typical second line limit employed at 60% of the NRC allowed limit.

5000 mrem	3000 mrem

5000 mrem corresponds to the correct NRC limit. 3000 mrem correspond to the typical second line limit employed at 60% of the NRC allowed limit.

RO Question 72

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_3___	_____
	K/A:	G2.3.10	
	Importance Rating:	2.9	3.3

Proposed Question:

Both units are at full power.

PK15-06, Cont Room Vent alarms on Unit 1 and Unit 2.

UNIT 1 Radiation Monitors	Unit 2 Radiation Monitors
RE-25 and RE-26 - 2.5 mrem/hr.	RE-25 and RE-26 - 5.0 mrem/hr.
RE-51 and RE-52 - 2.3 mrem/hr.	RE-53 and RE-54 – 5.2 mrem/hr

What action should be taken by the operators for both units?

A. Place Unit 1 selector switches in MODE 3. Place Unit 2 selector switches in MODE 4.

B. Place Unit 1 selector switches in MODE 4. Place Unit 2 selector switches in MODE 3.

C. Place the selector switches in MODE 3 on both Units 1 and 2.

D. Place the selector switches in MODE 4 on both Units 1 and 2.

Proposed Answer:

B. Place Unit 1 selector switches in MODE 4. Place Unit 2 selector switches in MODE 3.

Explanation:

A incorrect. Unit 1 radiation level at the pressurization (RE-51 and 52) intake is lower than Unit 2's (RE-53 and 54).

B correct. The Unit with the lower reading (Unit 1) selector switches are placed in MODE 4. The other unit is placed in MODE 3.

C incorrect. Both units are placed in different modes depending on radiation levels.

D incorrect. Both units are placed in different modes depending on radiation levels.

Technical Reference(s): AR PK15-06, STG H5, CRVS

Proposed references to be provided to applicants during examination: AR PK15-06

Learning Objective: 4714 - Explain the operation of the Control Room Ventilation system.

Question Source:

New

Question History: Last NRC Exam: N/A

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

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

Comments:

K/A: G2.3.10 - Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (2.9/3.3)

CONTROL ROOM VENT

06/15/05
Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
1055	Cont Rm Vent Supply Fan S-35 and S-36 Flo Lo	FS-5021 AND FS-5023	no flow AND no flow	2.1
1102	CRVS Hi Rad and/or Mode 4 Actuation Trn A	RE25X (U1)	≥ 1.6 mR/hr	2.1
		RE25X (U2)	≥ 1.6 mR/hr	
		RE51X (U1)	≥ 1.6 mR/hr	
		RE53X (U2)	≥ 1.6 mR/hr	
1103	CRVS Hi Rad and/or Mode 4 Actuation Trn B	RE26X (U1)	≥ 1.6 mR/hr	2.1
		RE26X (U2)	≥ 1.6 mR/hr	
		RE52X (U1)	≥ 1.6 mR/hr	
		RE54X (U2)	≥ 1.6 mR/hr	
1135	Control Room Press Htr TEH27A	TS-5005	≤ 75°F	2.1
1168	Control Room Press Htr TEH27B	TS-5007	≤ 75°F	2.1
1170	Control Room Dampers 2, 3, 7 or 8 in Man or UV	42X-2	manual control or UV detected	2.1
		42X-3	manual control or UV detected	
		42X-7	manual control or UV detected	
		42X-8	manual control or UV detected	

(Continued)

UNIT 1

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
1171	Control Room Press Fan S-99 Trouble	42X-1	loss of power	2.1
		42X-1A	loss of power	
		42X-99	loss of power	
1214	Radiation Instrument Failure	74HRCM1	UV / failure	2.1
1215	Control Room Dampers 2A, 3A, 7A, 8A/Htr HE-N Troub	42X-2A	manual control or UV detected	2.1
		42X-3A	manual control or UV detected	
		42X-7A	manual control or UV detected	
		42X-8A	manual control or UV detected	
1216	Control Room Press Fan S-98 Trouble	42X-1B	loss of power	2.1
		42X-1C	loss of power	
		42X-98	loss of power	
1230	Control Rm Vent Fan OC Trip	49-A-1	thermal overload	2.1
		49-A-2	thermal overload	
		49-A-3	thermal overload	
		49-B-1	thermal overload	
		49-B-2	thermal overload	
		49-B-3	thermal overload	
		49X-A-4	thermal overload	
		49X-B-4	thermal overload	

2. OPERATOR ACTIONS

2.1 General Actions **(All Inputs)**

2.1.1 IF high radiation exists as indicated on the RMS Panel on Unit 2 side of the Control Room,
THEN:

CAUTION: Do not manually select both units for Mode 4 operation, as this will start two pressurization fans.

a. Check Mode 4 status lights illuminated on both units. [ ]

NOTE: Pressurization fans are automatically swapped due to high radiation at intake of running fan. This will occur only once.

b. IF both units have a high radiation condition,
THEN:

1. On the unit having the lowest pressurization intake radiation, place BOTH selector switches in "MODE 4". [ ]
2. On the other unit, place BOTH selector switches in "MODE 3". [ ]
3. Periodically repeat step 2.1.1b to ensure that the operating fan is selected to supply from the unit with the lowest radiation. [ ]

c. IF only one unit has a pressurization intake high radiation condition,
THEN:

1. On the unit without the alarming monitor, place BOTH selector switches in "MODE 4". [ ]
2. On the other unit, place BOTH selector switches in "MODE 3". [ ]

d. Check that damper failure status lights are OFF to confirm proper system lineup and operation. [ ]

e. IF any damper failure status lights are ON,
THEN select the unselected subtrain. [ ]

f. IF CRVS streamers are hanging down,
THEN on the affected unit, select the unselected subtrain. [ ]

g. IF Mode 4 is initiated,
THEN verify Unit 2 pressurization fan S96 OR S97 is running. [ ]

h. REFER TO reporting requirements in XI1.ID2, "Regulatory Reporting Requirements and Reporting Process," and make any necessary reports. [ ]

i. WHEN the high radiation condition is cleared,
THEN reset ventilation mode PER OP H-5:IV, "Control Room Ventilation System Mode Changes." [ ]

> **NOTE 1**: **Input 1055** may alarm during CRVS testing in Mode 4 due to flow switch design.[Ref 4.2]
>
> **NOTE 2**: For CRVS fan OC alarm, breakers are in Panels PPCA and PPCB in the CRVS Equipment Room

2.1.2 IF the alarm was NOT a result of radiation,
THEN perform the following:

a. Determine the affected equipment by observing printout, VB-4 position, and failure panel indicating lights. [ ]

b. Dispatch an Operator to inspect the affected equipment. [ ]

2.1.3 IF neither train of Control Room cooling are operating,
THEN:

a. Perform a walkdown of equipment cabinets in the Control Room to identify local hot areas. [ ]

b. Open cabinet doors as necessary to provide additional cooling. [ ]

c. REFER TO CP M-10, "Fire Protection of Safe Shutdown Equipment," for guidance on providing temporary cooling to the Control Room. [ ]

2.1.4 REFER TO TS 3.7.10. [ ]

2.1.5 REFER TO the following as necessary to diagnose the problem: [ ]

- OP H-5:IV, "Control Room Ventilation System Mode Changes"
- STP M-6A, "Routine Surveillance Testing of Control Room Ventilation System"

2.1.6 Probable Causes

- High radiation or radiation monitor failure
- Supply fans S-35 and S-36 failed
- Pressurization fans S-98 or S-99 failed
- Damper failure
- Dampers 2, 2A, 3, 3A, 7, 7A, 8, 8A placed in manual or breaker open
- Pressurization system HEPA filter inlet heater failure
- Overcurrent trip of S-35, CR-35, CP-35, S-36, CR-36, CP-36, S-39 or S-40
- Overcurrent trip alarm caused by deenergizing CP-35 or CP-36 control circuit[Ref 4.3]

3. AUTOMATIC ACTIONS

3.1 If Mode 3 initiated, both units will shift to Mode 3.

3.2 High radiation at the normal intake or Phase A isolation will initiate Mode 4:

3.2.1 One unit will be in ventilation Mode 4, the other unit in Mode 3.

3.2.2 Failure of a pressurization fan will start an alternate fan.

3.3 If the opposite intake is not already in alarm for high radiation, a high radiation condition at either North or South pressurization intakes will cause a swap to the opposite intake.

3.4 If only one Control Room ventilation (normal intake) radiation monitor is in alarm, only the associated train of equipment will shift:

If RM in alarm is:	these dampers close:	and these dampers possibly fail:
25	2, 3, 7, 8	2A, 3A, 7A, 8A
26	2A, 3A, 7A, 8A	2, 3, 7, 8

4. REFERENCES

4.1 Technical Specification 3.7.10, "Control Room Ventilation System (CRVS)"

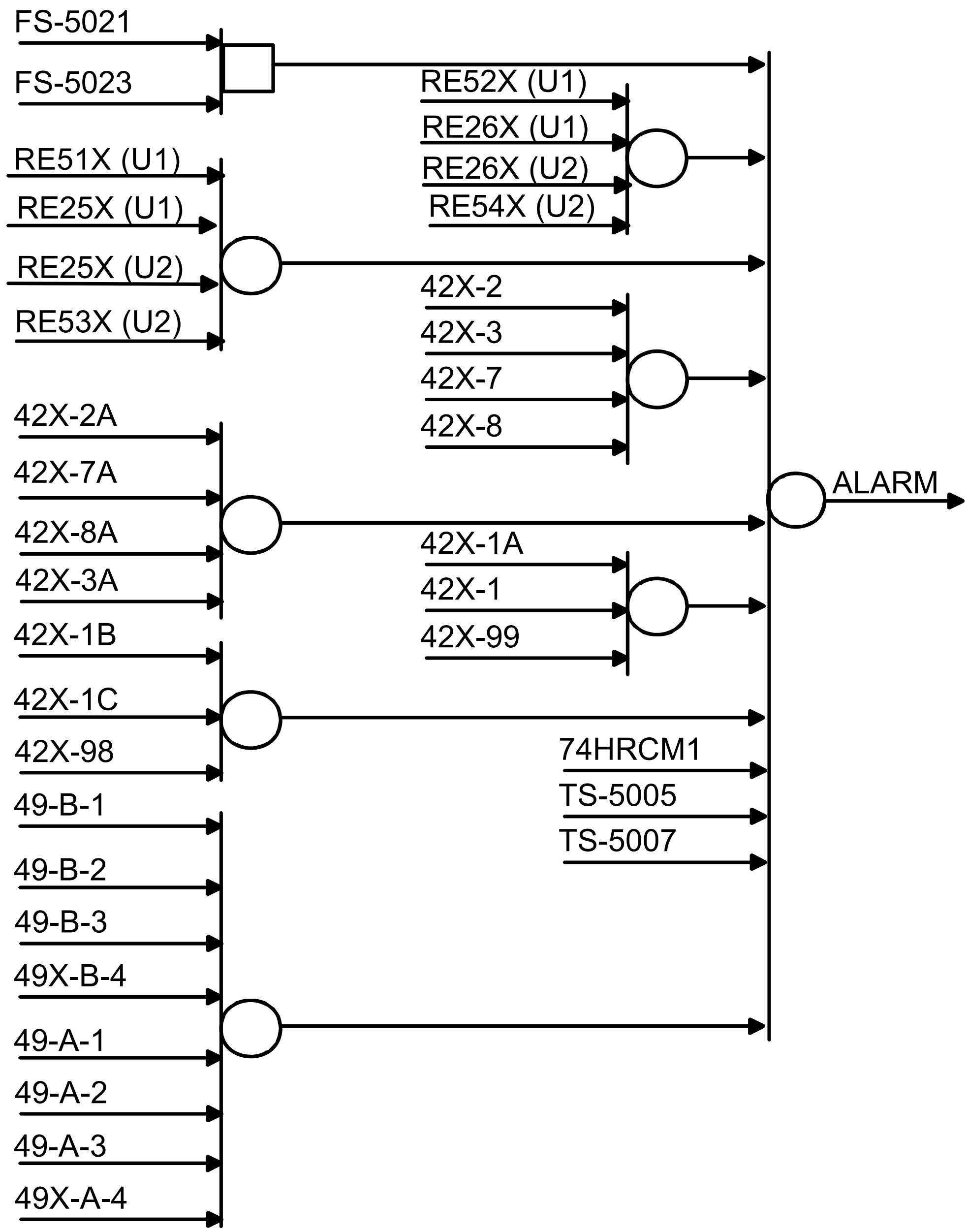
4.2 A0426513, "S-35/36 Flow Low"

4.3 A0467182, "Control Room Vent. Fan OC Trip is in Alarm (CBAR)"

4.4 Main Annunciator Schematics (Electrical Drawing Section 8)

- 501134
- 501141 (Input 1055)
- 501143 (Inputs 1102, 1103)
- 502115 (Inputs 1135, 1168)
- 502116 (Inputs 1170, 1171, 1214, 1215, 1216, 1230)

5. <u>LOGIC DIAGRAM</u>



Mode 4 Operation, Continued

Fan Select Logic
Obj 12

The control circuit attempts to minimize radiation exposure to the Control Room Operators by selecting the best pressurization fan to run.

1. When a Phase A Isolation occurs OR if high radiation is detected at the normal Control Room Ventilation Intake (RE-25 or RE-26), relays will close to start the SELECTED fan on the OPPOSITE Unit. If the opposite Unit's pressurization fan is not running and the radiation at the Pressurization Fan (RE-51, 52, 53, or 54) is not in alarm, then the selected fan will start.

2. If the SELECTED fan fails to start within 15 seconds, the NON-SELECTED fan will start, provided that the opposite Unit's pressurization fan is not running and the radiation at the Pressurization Fan is not in alarm.

3. If high radiation prevents either fan from running, relays will start the selected fan on the affected unit. *This will happen regardless of the radiation level at this fan.*

4. If a pressurization fan is running with a high radiation, the operator may start a pressurization fan on the other end by:
   a. Selecting Mode 3 on the side to be shutdown.
   b. Selecting Mode 4 on the side to be run.

   The fan to be run will then start.
   The previous fan will then shutdown. ("Opposite Fans Shutdown" signal is lost.)
   The fan will continue to run as long as a high radiation exists at the shutdown fan.

If radiation levels are normal, selecting Mode 4 will start the selected pressurization fan.

Continued on next page

RO Question 73

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_3___	_____
	K/A:	G2.3.11	
	Importance Rating:	2.7	3.2

Proposed Question:

GIVEN:
- The crew has entered E-3, Steam Generator Tube Rupture.
- The ruptured S/G has been identified.

What action will the operator perform on the ruptured steam generator 10% Steam Dump controller?

A. The setpoint will be raised to 1040 psig to minimize radiological releases.

B. The controller will be placed in MANUAL and closed to minimize radiological releases.

C. The setpoint will be raised to 1040 psig to increase ruptured steam generator pressure and decrease primary to secondary leakage.

D. The controller will be placed in MANUAL and closed to increase ruptured steam generator pressure and decrease primary to secondary leakage.

Proposed Answer:

A. The setpoint will be raised to 1040 psig to minimize radiological releases.

Explanation:

A correct. The valve setpoint is raised but left in AUTO to actuate if necessary.

B incorrect. Controller left in AUTO.

C incorrect. Action is not performed to minimize primary to secondary leakage.

D incorrect. Controller left in AUTO, not performed to minimize primary to secondary leakage.

Technical Reference(s): E-3, Background step 3

Proposed references to be provided to applicants during examination: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:
Bank - P-27379 - modified

Question History: Last NRC Exam: N/A

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

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

Comments:

K/A: G2.3.11 - Ability to control radiation releases. (2.7/3.2)

#	1.00

A reactor Trip and SI have occurred on Unit 1 due to a Steam Generator Tube Rupture. E-3, Steam Generator Tube Rupture, is currently in progress. The ruptured S/G has been identified, and its 10% Steam Dump controller is in AUTO and set for 1040 PSIG.

Why is the ruptured S/G 10% Steam Dump controller setpoint increased to 1040 PSIG?

A. This will minimize the radiological releases while preventing lifting of the code safety valves.

B. If the S/G goes water solid, the 10% Steam Dump valve can discharge water.

C. To increase ruptured S/G pressure in order to decrease the primary to secondary leakage.

D. To ensure that S/G pressure can <u>NOT</u> increase above 1055 PSIG.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7920	Explain basis of emergency procedure step

Reference Id: P-27379
Must appear: No
Status: Active

Difficulty: 3.00

KNOWLEDGE:

o The PORV on the ruptured steam generator should remain available to limit steam generator pressure unless it fails open. This will minimize any challenges to the code safety valve.

o Means of closing and isolating plant specific valves between the main steamline isolation valve and turbine stop valve, such as steam dump valves, Moisture Separator Reheater (MSR) valves, sample line valves, etc. These valves provide a backup means of isolating the ruptured steam generator if the associated main steamline isolation valve should fail. These valves may be closed in parallel with subsequent recovery steps after the ruptured SG(s) are isolated from the intact SG(s) by closing the appropriate main steam isolation and bypass valves.

o If it is necessary to trip the turbine-driven AFW pump, in order to quickly terminate the radiological release from and depressurization of the ruptured SG, it might still be possible through local actions to achieve selective isolation of the ruptured SG from the turbine-driven AFW pump. In this context selective isolation means that the ruptured SG and only the ruptured SG is isolated from the turbine-driven AFW pump. Selective isolation is feasible only if at least one SG capable of supplying the turbine-driven AFW pump is intact. In terms of the reference plant, in which the turbine-driven AFW pump can be supplied from two SGs, each supply-steam line contains an air-actuated isolation valve normally operated from the control room and a locally operated manual isolation valve. Downstream of the isolation valves the two steam lines join to supply the turbine-driven AFW pump. Local actions can be directed toward closing the manual isolation valve from the ruptured SG and/or toward local-manual operation of the air-actuated isolation valve from the ruptured SG, which the control room operator tried but failed to close while attempting to isolate flow from the ruptured SG. In either case, once the ruptured SG is selectively isolated from the turbine-driven AFW pump, the pump can be restarted if necessary.

PLANT-SPECIFIC INFORMATION:

o Identify valves between main steamline isolation valve and turbine stop valves.

o Actions for isolating the ruptured steam generator should consider any difference in the time delay between closing individual main steamline isolation valves (slow close) and closing all main steamline valves (fast close). The benefit of steam dump to condenser must be weighed against the possibility of steam generator overfill due to a delay in stopping primary-to-secondary leakage.

o Action should be taken to identify and isolate steam traps upstream of the MSIVs. Unisolated traps could result in a steam release causing the ruptured generator to slowly depressurize. An unisolated steam trap could also result in a small release of activity to the secondary side of the plant.

o (O.03) Setpoint for SG PORV controller (typically 25 psi below lowest safety valve set pressure).

This setpoint should be greater than no-load pressure in order to minimize atmospheric releases from the ruptured steam generator and less than the minimum safety valve setpoint to prevent lifting of the code safety valves. The 25 psi margin is a typical value to allow for opening of the PORV prior to lifting of the safety valve.

o If the turbine-driven AFW pump will not trip or cannot be tripped from the control room, then actions to isolate it from the ruptured SG must be pursued locally. These actions depend on the plant-specific design, and range from local-manual trip of the turbine-driven AFW pump to local operation of manual isolation valves. Priority should first be given to terminating the radiological release from and depressurization of the ruptured SG through the turbine-driven AFW pump. Then time and effort can be directed toward selective isolation of the ruptured SG from the turbine-driven AFW pump. If selective isolation of the ruptured SG can be achieved, then the turbine-driven AFW pump can be restarted if necessary.

RO Question 74

Examination Outline Cross-Reference:

Level	RO	SRO
Tier:	_3___	_____
Group:	_4___	_____
K/A:	G2.4.21	
Importance Rating:	3.7	4.3

Proposed Question:

Which of the following Critical Safety Function RED paths can be directly caused by procedurally directed operator action?

A. Subcriticality

B. Core Cooling

C. Heat Sink

D. Containment

Proposed Answer:

C.Heat Sink

Explanation:

A incorrect. Caused by power generation greater than 5%.

B incorrect. Caused by overheating thermocouples.

C correct. Can be caused by the operator throttling AFW flow below 435 gpm while no steam generator is above 6%, as in the case of ECA-2.1 or FR-P.1

D incorrect. Caused by high containment pressure.

Technical Reference(s): FR-H.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 9704 - Identify entry conditions for the FRPs

Question Source:
New

Question History: Last NRC Exam: N/A

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

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

Comments:

K/A: G2.4.21 - Knowledge of the parameters and logic used to assess the status of safety functions including:
1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.
(3.7/4.3)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. CHECK Total Feedflow LESS THAN 435 GPM DUE TO OPERATOR ACTION:	
a. RETURN TO procedure and step in effect	a. GO TO Step 2.

****

CAUTION: Feedflow should not be reestablished to any Faulted S/G if a Nonfaulted S/G is available.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. CHECK If Secondary Heat Sink Is Required:	
a. RCS Pressure - GREATER THAN ANY Intact S/G	a. Return to Procedure and Step in Effect.
b. RCS Hot Leg Temperature - GREATER THAN 350°F	b. IMPLEMENT OP B-2:V, RHR - PLACE IN SERVICE DURING PLANT COOLDOWN while continuing in this procedure. IF Adequate cooling with RHR System established, THEN RETURN TO procedure and step in effect.
3. CHECK CCP Status - AT LEAST ONE AVAILABLE	Perform the following: a. Stop all RCPs. b. GO TO Step 12 (Page 13). OBSERVE CAUTION prior to step.

CAUTION: If total feed flow is less than (S.02) gpm due to operator action, this guideline should not be performed.

PURPOSE: To alert the operator that the performance of FR-H.1 is required only if minimum feed flow capability is lost

BASIS:

During the performance of certain guidelines (e.g., ECA-2.1, FR-P.1, FR-S.1 and FR-Z.1), it is possible that the SG level is below the narrow range and the total feed flow is throttled to less than the minimum AFW flow requirement. If the feed flow is reduced due to operator action to minimize feed flow as instructed in these guidelines and the capability of providing the minimum feed flow is available (i.e., pumps and valves in the feedwater system are capable of being used if necessary), then a transfer from these guidelines is not required and guideline FR-H.1 should not be performed.

ACTIONS:

Determine if total feed flow is less than (S.02) gpm due to operator action

INSTRUMENTATION:

- o Total feed flow indication
- o Feed flow control valve position indication
- o Feed pump status indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

The operator must be aware that the availability to provide feed flow to the steam generators determines if FR-H.1 is to be performed.

RO Question 75

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_3___	_____
	Group:	_4___	_____
	K/A:	G2.4.48	
	Importance Rating:	3.5	3.8

Proposed Question:

Unit 1 is at full power, in a normal lineup with no equipment out of service.

For which of the following set of valid conditions could the BOPCO take corrective action without obtaining prior permission, and then report their actions to the SFM "after the fact"?

A. The operating CCP 1-2 trips on overcurrent.

B. Containment temperature increases 5°F with 3 CFCUs running in LOW.

C. Main Feedwater pump 1-1 trips.

D. PK01-01, “ASW SYS HX dp/HDR PRESS” alarms with only 1 ASW heat exchanger in service.

Proposed Answer:

C. Main Feedwater pump 1-1 trips

Explanation:

A, B and D are abnormal conditions which would be promptly reported to the SFM.

C correct. An immediate action in AP-15 requires a reactor trip if power is greater than 80%.

Technical Reference(s): OP1.DC11 – Conduct of Operations – Abnormal Plant Conditions.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3705 - Explain the responsibilities of the control room staff positions during emergencies

Question Source:

Bank – Modified R-55094

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41.7 - Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: G2.4.48 - Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions. (3.5/3.8)

1.00

Under which of the following set of conditions would the BOPCO be expected to take the specified corrective action without obtaining prior permission and then report his actions to the SFM "after the fact"?

A. Starting the standby condensate/booster set upon its failure to automatically start, during an LTB.

B. Placing the standby ASW/CCW HX in service in response to alarm AR PK01-01, "ASW SYS HX dp/HDR PRESS."

C. Starting CCP 1-1 following an overcurrent trip of operating CCP 1-2.

D. Starting a standby CFCU due to an observed increase in containment temperature.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

3705	Explain the responsibilities of the control room staff positions during emergencies

Reference Id: R-55094
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 2.00

5.2.3 The shift foreman has control room command and provides overall direction and supervision of the unit while maintaining an overall view of the plant situation. The SFM should remain in the "controls" areas when the plant is in an unstable condition.

5.2.4 Two licensed operators should be assigned to perform control board manipulations and perform necessary verifications or respond to information requests. To best perform these duties these operators should:

a. Communicate in accordance with the operations communication standard (OP1.DC12).

b. Organize their activities so that one operator is responsible for the control console and vertical boards one and two, and the other operator is responsible for vertical boards three, four, and five.

c. Promptly report all unexpected or unexplained equipment operation or plant response to the shift foreman.

d. Utilize self-verification techniques for all equipment operation.

e. <u>Take time critical actions to ensure nuclear safety, prevent injury or damage to property, or to maintain service of critical equipment</u>.

f. Actuate those safety systems/components which have failed to respond to actuation signals and promptly inform the SFM.

5.2.5 The WCSFM shall report to the Control Room during abnormal plant operations.

a. Following a reactor trip, the WCSFM shall assist the crew by assessing plant response and informing the SFM appropriately. The WCSFM should use plant instrumentation, SPDS displays and the main annunciator system to monitor:

- Reactor coolant system (RCS) and pressurizer (PZR) heatup/cooldown rates
- RCS pressure for unexpected changes
- Containment parameter changes
- PORV lifts or leakage
- Safety valve lifts or leakage
- Equipment malfunctions the crew is unaware of
- Condenser availability
- Unexpected annunciator alarms

SRO Question 76

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_1___
	K/A:	_____	EPE 009 EA2.21
	Importance Rating:		3.9

Proposed Question:

Unit 1 is operating at 100% power in a normal full power, steady state lineup. Charging pump 1-1 is in service.

The following events occur:

- PK11-21, High Radiation, input 669, In Core Seal Table Area Mon Hi Rad RE 7, alarms.
- The operator reports RM-7 is reading 55 mr/hr and trending up on the PPC.
- Additionally, trends are increasing for Containment Radiation Monitors, RM-11 and RM-12.
- Containment temperature is increasing.
- Charging flow is 115 gpm.
- RCP seal injection flow is 8 gpm to each RCP.
- Pressurizer level is 60% and steady
- Letdown is not isolated

Based on the radiation monitors and system response, what action should be taken by the Shift Foreman?

A. The leakage is small. Enter OP O-21, “Isolating Leaking Thimble Tubes” to find and isolate the leak.

B. The leak is in excess of leakage allowed by Technical Specifications. Enter AP-25, “Rapid Load Reduction” to quickly shutdown the unit.

C. The leak within the capacity of normal charging. Enter AP-1, “Excessive Reactor Coolant System Leakage” and attempt to isolate the leak.

D. Leakage is in excess of allowable for continued operation. Direct a reactor trip, SI actuation and enter E-0, “Reactor Trip or Safety Injection.”

Proposed Answer:
C. The leak within the capacity of normal charging. Enter AP-1, “Excessive Reactor Coolant System Leakage” and attempt to isolate the leak.

Explanation:
A incorrect, increasing containment radiation and size of the leak precludes using O-21.

B incorrect, RCS leakage is not an entry condtioin for AP-25.

C correct, leakage is less than 50 gpm and within the capacity of normal charging.

D incorrect, this is for larger leaks (greater than 50 gpm).

Technical Reference(s): O-21, Isolating Leaking Thimble Tubes.
AP-1, Excessive Reactor Coolant System Leakage

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

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: EPE 009 EA2.21 - Ability to determine or interpret the following as they apply to a small break LOCA: Containment radiation trend recorder (3.9)

TITLE: Excessive Reactor Coolant System Leakage

07/19/06
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure covers RCS leakage conditions where the charging system is capable of maintaining normal PZR level while the PZR heaters maintain normal system pressure. The goal is to limit the release of radioactive material by isolating defective component or reduce the magnitude of the leakage to within Tech Spec limits. ITS 3.4.13, "RCS Operational Leakage," applies in Modes 1-4.

2. SYMPTOMS

2.1 Irregular RCP seal flow (FR-157 or 159, VB2)

2.2 Charging/letdown flow mismatch (FI-134A, VB2; FI-128A, CC2)

2.3 Decreasing VCT level (LI-112, VB2)

2.4 Abnormal seal injection flow (FI-144, 143, 115, 116; VB2)

2.5 Abnormal letdown flow (FI-134A, VB2)

2.6 PRT high level LI-470, high press PI-472, or high temp TI-471 (VB2)

2.7 High RCDT level (LI-188, Aux Board), or increased pumpdown frequency

2.8 PZR safety valve or PORV discharge line high temp (TI-465, 467, 469, or 463; VB2)

2.9 Rx vessel flange leakoff temp high (TI-401, VB-2)

2.10 High Containment Sump or Rx Cavity Sump levels (LR-60, 61, 62; PAM1)

2.11 High Containment air temp or pressure (YR-26; VB1)

2.12 Increased Containment radiation (RM-11 or 12, RMS CAB II)

2.13 Increasing rad levels (RE-17A/B) with possible autoclosure of RCV-16 (VB1)

2.14 Increasing CCW surge tank level (VB1)

2.15 Possible autoclosure of FCV-357 (VB1)

2.16 Increasing CCW header "C" flow (FI-46, VB1)

2.17 Increasing letdown temp (TI-130, VB2)

2.18 TCV-130, CCW to Letdown HX, goes full open (VB2)

2.19 Possible flashing in letdown line (PI-135, VB2)

2.20 Possible Main Annunciator Alarms:

2.20.1 (PK01-07) CCW SYS SURGE TK LVL/MK-UP

2.20.2 (PK01-08) CCW HEADER C

2.20.3 (PK01-16) CONTMT ENVIRONMENT P250

2.20.4 (PK01-17) CFCU DRAIN LEVEL HI

2.20.5 (PK04-24) VCT PRESS/LVL/TEMP

2.20.6 (PK05-01 to 04) RCP NO.____

2.20.7 (PK05-14) RCDT LEVEL HI-LO

2.20.8 (PK05-17) PZR PRESSURE LOW

2.20.9 (PK05-20) PZR RELIEF/SAFETY VLVS OPEN

2.20.10 (PK05-21) PZR LEVEL HI/LO

2.20.11 (PK05-22) PZR LEVEL HI/LO Control

2.20.12 (PK05-23) PZR SAFETY OR RELIEF LINE TEMP

2.20.13 (PK05-25) PRT PRESS/LVL/TEMP

2.20.14 (PK11-02) RX FLANGE LEAK-OFF TEMP HI

2.20.15 (PK11-06) SJAE HI-RAD

2.20.16 (PK11-17) SG BLOWDOWN HI RAD

2.20.17 (PK11-18) MAIN STM LINE HI RAD

2.20.18 (PK11-21) HIGH RADIATION

TITLE: Operating Order 0-21: Isolating Leaking Thimble Tubes

01/06/00
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

   1.1 This Operating Order provides instructions for isolating leaking incore thimble tubes from their respective 10 Path Rotary Transfer Device. These instructions should be implemented whenever the Moveable Incore Detector System leak detection system alarm activates and the alarm is considered to be valid.

2. DISCUSSION

   2.1 This procedure ensures that thimble tube leaks are properly verified and isolated in a safe and timely manner.

3. RESPONSIBILITIES

   3.1 Shift Foreman (SFM) for implementing this procedure.

   3.2 Radiation Protection Section for High Rad Area support and deconning support, if required.

   3.3 Maintenance Services (MS) for troubleshooting assistance (removal of 10 Path Rotary Transfer Device side covers).

4. PREREQUISITES

   4.1 All Moveable Incore Detectors have been placed in their STORAGE locations.

   4.2 An SWP is written and approved for Seal Table Room entry.

   4.3 Handheld pyrometer is available for tube identification.

5. PRECAUTIONS AND LIMITATIONS

   5.1 A Special Work Permit is required for work performed in the Seal Table Room under these instructions.

6. INSTRUCTIONS

****

CAUTION: If at any time during the implementation of this procedure it is observed that Reactor Coolant System charging flow increases to maintain PZR level, implement OP AP-1 and discontinue this procedure. The Seal Table Room conditions would prohibit entry for isolation of the leak.

****

6.1 Verify that all Moveable Incore Detectors are placed in their STORAGE locations.

6.2 Notify Radiation Protection Section that a containment entry will be required to isolate a thimble tube leak in the Seal Table Room.

6.3 Attempt to quantify the leak rate while performing subsequent steps in this procedure.

6.3.1 Perform STP R-10C.

6.3.2 Refer to Technical Specifications 3.4.6.2 (ITS 3.4.13), Reactor Coolant System - Operational Leakage.

6.4 Monitor RM-7 for changes in Seal Table Room radiation levels as an indication of whether the leak is large since the leak detection system drains onto the seal table.

6.5 Make an entry with Radiation Protection and MS personnel to locate and isolate the leaking thimble tube. Take a 10-foot ladder, a handheld pyrometer, and a straight-slot screwdriver suitable for use on #10-24 pan-head screws.

6.6 Upon entering the Seal Table Room, visually inspect the six 10 Path Rotary Transfer Devices for obvious signs of leakage. If no leakage is observed, proceed to Step 6.7. If a leak is observed at one of the 10 Path Rotary Transfer Devices proceed as follows:

6.6.1 Immediately close all of the isolation valves (9 or 10) above the seal table for those thimble tubes associated with the 10 Path Rotary Transfer Device which is leaking.

6.6.2 Exit the Seal Table Room and notify Control Room personnel to reset the leak alarm at the MIDS Power Distribution Panel.

NOTE: It may be necessary to reset the leak detection system alarm numerous times if there is some residual water still draining out of the 10 Path Rotary Transfer Device.

6.6.3 After a few minutes verify that the alarm stays out and that no water is leaking out of the open-ended drain tubing.

6.6.4 Verify that the 10 Path Rotary Transfer Device is no longer leaking.

SRO Question 77

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	__1__
	Group:	_____	__1__
	K/A:	_____	APE 015/017G2.4.4
	Importance Rating:		4.3

Proposed Question:

GIVEN:

- PK 05-01, RCP No. 11 in alarm due to inputs:
    - 1393 RCP 1-1 Seal Leakoff Flo Hi
    - 1259 RCP 1-1 No. 2 Seal Leakoff Flo Hi
- RCP 1-1 Number 1 seal DP is 1200 psid
- RCP 1-1 Number 1 seal leakoff flow is 5.8 gpm.
- RCP 1-1 bearing outlet and number 1 seal outlet temperatures are stable.
- The crew has entered AP-28, RCP Malfunction, section B, RCP No. 1 Seal Failure

Which of the following actions should be taken by the Shift Foreman?

A. Based on the number 1 seal leakoff flow high alarm, direct a reactor trip and a trip of RCP 1-1 and go to E-0.

B. Based on the number 2 seal leakoff flow high alarm, direct a reactor trip and a trip of RCP 1-1 and go to E-0.

C. Based on total seal leakoff flow greater than allowed, direct a reactor trip and a trip of RCP 1-1 and go to E-0.

D. Based on high number 1 seal leakoff flow, direct a unit shutdown per AP-25, Rapid Load Reduction.

Proposed Answer:
B. Based on the number 2 seal leakoff flow high alarm, direct a reactor trip and a trip of RCP 1-1 and go to E-0.

Explanation:
A incorrect. Number 1 seal leakoff flow does not warrant a reactor trip and RCP trip unless bearing outlet or seal outlet temperatures are increasing.

B correct, the presence of the alarm for number 2 seal leakoff flow high requires a reactor and RCP trip.

C incorrect, total leakoff flow is less than 8 gpm based on number 1 seal leakoff less than 7 gpm.

D incorrect, AP-25 entry is warranted if seal leakoff is greater than 6 but less than 7 gpm.

Technical Reference(s): AP-28, RCP Malfunction, rev2.

Proposed references to be provided to applicants during examination: OP AP-28, section B

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: APE 015 G2.4.4 - RCP Malfunction - Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (4.3)

SECTION B: RCP NO. 1 SEAL FAILURE (Continued)

ACTION / EXPECTED RESPONSE | **RESPONSE NOT OBTAINED**

1. **CHECK No. 1 Seal Leakoff Flow on Affected RCP(s)**
   - FR-159, RCP 1 **OR** RCP 2 SEAL No. 1 RTN FLO
   - FR-157, RCP 3 **OR** RCP 4 SEAL No. 1 RTN FLO
     - Decision Table

<table>
<tr><th></th><th>RCP Radial Bearing Outlet Temperature
AND
No. 1 Seal Outlet Temperatures are:
STABLE</th><th>RCP Radial Bearing Outlet Temperature
OR
No. 1 Seal Outlet Temperatures are:
INCREASING</th></tr>
<tr><td>No. 1 Seal Leak Off Less Than 0.8 GPM</td><td>1. Shutdown Unit per OP AP-25 within 8 hours AND continue with this section.
2. Monitor for RCP vibration trip criteria
3. When Stable in Mode 3
Then a. Stop affected RCP
b. Close valve affected spray if RCP 1 or RCP 2</td><td rowspan="3">Perform ALL of the Following:
1. Trip the reactor AND GO TO EOP E-0, "Reactor Trip or Safety Injection" while completing this step.
2. Stop affected RCP(s)
3. Close spray valve (if RCP 1 or RCP 2).
4. Close Seal No. 1 outlet isolation valve 3 to 5 minutes after RCP is stopped.
5. Monitor Attachment 4.1 parameters for affected RCP(s).</td></tr>
<tr><td>No. 1 Seal Leak Off is:
0.8 GPM to 6 GPM</td><td>1. Continue pump operation.
2. Check seal injection flow GREATER THAN No. 1 seal leakoff flow.
3. Monitor Attachment 4.1 parameters for affected RCP(s).
4. Contact Operations Management and EFIN for further evaluation.
5. Return to procedure and step in effect.</td></tr>
<tr><td>No. 1 Seal Leak Off Greater Than 6 GPM AND
Less Than 7 GPM</td><td>1. Shutdown Unit per OP AP-25, "Rapid Local Reduction" within 8 hours AND continue with this section.
2. Monitor for RCP vibration trip criteria.
3. Stop affected RCP(s)
4. Monitor Attachment 4.1 parameters for affected RCP(s).
5. Close Seal No. 1 outlet isolation valve 3 to 5 minutes after RCP is stopped.
6. Close spray valve (if RCP 1 or RCP 2).</td></tr>
</table>

SECTION B: RCP NO. 1 SEAL FAILURE (Continued)

NOTE 1: Total Seal Leakoff Flow is the sum of No. 1 and No. 2 Seal Leakoff indications for affected RCP(s).

NOTE 2: It is acceptable to assume a No. 2 seal leakoff of less than 1 gpm if PK05 **"RCP SEAL No. 2 LKOFF FLO HI"** is NOT IN.

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. CHECK RCP No. 2 Seal Leakoff Flow LESS THAN 1 GPM	Perform ALL of the following:
• RCP Seal No. 2 Leakoff FLO HI Alarm NOT IN	a. Trip Reactor **AND** GO TO EOP E-0, "Reactor Trip or Safety Injection" while completing this step.T35185 b. TRIP affected RCP(s) c. Between 3 to 5 minutes after RCP stopped, CLOSE Seal No. 1 outlet Isolation Valve for affected RCP(s). d. Monitor CCW temperature and flow to affected RCP(s). e. Close spray valve (if RCP 1 or RCP 2).
3. CHECK Total Seal Leakoff Flow - LESS THAN 8.0 GPM	Perform ALL of the following:
a. RCP No. 1 SEAL LKOFF **FLO** Less Than 7 GPM: • FR-159, RCP 1 or RCP 2 • FR-157, RCP 3 or RCP 4	a. Trip Reactor **AND** GO TO EOP E-0, "Reactor Trip or Safety Injection" while completing this step.T35185 b. TRIP affected RCP(s) c. Between 3 to 5 minutes after RCP stopped, CLOSE Seal No. 1 outlet Isolation Valve for affected RCP(s). d. Monitor CCW temperature and flow to affected RCP(s). e. Close spray valve (if RCP 1 or RCP 2).

SECTION B: RCP NO. 1 SEAL FAILURE (Continued)

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. CHECK RCP Parameters within OPERATING LIMITS per Attachment 4.1	Perform ALL of the following: a. Trip Reactor **AND** GO TO EOP E-0, "Reactor Trip or Safety Injection" while continuing this section. b. TRIP affected RCP(s). c. **IF** RCP RAD BRG OUT TEMP **OR** SEAL WTR OUTLET TEMP INCREASING, **THEN** Between 3 to 5 minutes after RCP stopped, CLOSE Seal No. 1 Outlet Isolation Valve for affected RCP(s). d. Monitor CCW temperature and flow to affected RCP(s). e. Close spray valve (if RCP 1 or RCP 2).
5. PERFORM STP R-10C, "Reactor Coolant System Inventory" or Estimate to Determine if RCS Leakage Excessive	Implement AP-1, "Excessive Reactor Coolant System Leakage" while completing this step.
6. VERIFY RCPs Thermal Barrier Cooling In Service:	Perform the following:
a. CCW-FCV-750 RCP Thermal Barrier Return IC Valve - OPEN	a. Verify Seal Injection flow to all RCPs greater than or equal to 6 gpm per pump.
b. CCW-FCV-357 RCP Thermal Barrier Return OC Valve - OPEN	b. **IF** Seal Injection flow to all RCPs less than 6 gpm per pump **THEN** GO TO Section F, pg. 29, of this procedures
c. RCP No. 1 Seal Outlet temperatures-NORMAL	
d. RCP Radial Brg Outlet temperatures-NORMAL	
e. CCW Header C alarm (PK01-08) - OFF	

SRO Question 78

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_1___
	K/A:	_____	APE026 AA2.06
	Importance Rating:		3.1

Proposed Question:

Unit 1 is operating at 100% power.

The following alarms are received:
- PK05-01 (RCP No. 11)
- PK05-02 (RCP No. 12)
- PK05-03 (RCP No. 13)
- PK05-04 (RCP No. 14)
- PK01-08, CCW Header C

The CO reports the following RCP conditions:
- The cause for PK01-08 is FCV-357, CCW Thermal Barrier Return valve is closed.
- Cause for the RCP alarms is radial bearing temperature above 160°F
- RCP Radial Bearing Outlet temperatures are all reading approximately 205°F and stable
- RCP Seal Outlet temperatures are all reading approximately 215°F and stable
- RCP Motor Shaft Vibrations are all 2 to 3 mils and stable
- RCP Pump Shaft Vibrations are all 4 to 5 mils and stable

What action should be taken by the Shift Foreman for the current plant conditions?

A. RCP temperatures are greater than allowed. Immediately direct a reactor trip, all RCPs tripped and go to E-0, Reactor Trip or Safety Injection.

B. A loss of RCP thermal barrier cooling has occurred. If flow is not restored within 5 minutes, direct a reactor trip, all RCPs tripped and go to E-0, Reactor Trip or Safety Injection.

C. RCP Radial bearing temperatures are high. Direct the operator to lower temperature by opening the Number 1 Seal Bypass Valve for each RCP in accordance with the alarm response for each RCP.

D. A loss of RCP thermal barrier cooling has occurred. Enter OP AP-11, Malfunction of the Component Cooling Water System to investigate and correct the loss.

Proposed Answer:
D. A loss of RCP thermal barrier cooling has occurred. Enter OP AP-11, Malfunction of the Component Cooling Water System to investigate and correct the loss.

Explanation:
A incorrect, all temperatures are below the threshold to trip. Temperature is above motor bearing setpoint but not radial bearing setpoint (225°F) or seal outlet temperature (235°F)

B incorrect, a trip is required in 5 minutes if all CCW has been lost to the RCPs. Only thermal barrier flow has been lost (motor oil cooler still supplied by CCW)

C incorrect, the bypass valves are not opened if RCS pressure is above 1000 psig.

D correct, per AP-11, there is no trip criteria for a loss of CCW to thermal barrier. The cause is investigated, and AP-1 implemented.

Technical Reference(s): AP-11, Malfunction of CCW System, SECTION E: LOSS OF CCW FLOW TO RCPs

Proposed references to be provided to applicants during examination: None

Learning Objective: 8121 - State the effects of RCP operations during loss of CCW

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: APE 026 AA2.06 - Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The length of time after the loss of CCW flow to a component before that component may be damaged. (3.1)

RCP
NO. 11

12/27/05

Effective Date

QUALITY RELATED

CONTAINS REACTOR TRIP CRITERIA

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS		DEVICE	SETPOINT	STEP
358	RCP 1-1 Seal Leakoff Flo Lo		FC-159A	<0.9 gpm	2.1
359	RCP 1-1 Seal Water Delta P Lo		PC-188	<250 psi	2.1
361	RCP 1-1 Pp Radial Brg Temp Hi		TC-155	>160°F	2.1
366	RCP 1-1 Seal Water Inj Flo Lo		FC-144	<6.5 gpm	2.1
370	RCP 1-1 No. 1 Seal Outlet Temp Hi		TC-148	>170°F	2.1
500	RCP 1-1 Temp PPC		Y0401C		2.1
	INPUTS TO Y0401C	PPC ID			
	RCP 1-1 Stator Temp	T0412A		≥248°F	
	RCP Mtr 1-1 Top Radial Brg Temp	T0413A		≥180°F	
	RCP Mtr 1-1 Top Thrust Brg Temp	T0414A		≥180°F	
	RCP Mtr 1-1 Lo Radial Brg Temp	T0415A		≥180°F	
	RCP Mtr 1-1 Lo Thrust Brg Temp	T0416A		≥180°F	
501	RCP 1-1 OC Trip		51XVE3	max ≥400 A pri inst ≥3680 A pri th ≥2160 A pri	2.1
			51XVE3R	max ≥440 A pri inst ≥4000 A pri	
502	RCP 1-1 Fdr Grd		50NVE3	grounded	2.1
503	RCP 1-1 Standpipe Lvl Hi		LC-486	>123 ft 7 in el.	2.1
504	RCP 1-1 Standpipe Lvl Lo		LC-487	<119 ft 9 in el.	2.1

(Continued)

2.2.5 Probable Causes

- Number 1 or Number 2 Seal failure
- Loss of injection water
- Crud in Number 1 Seal
- Low differential pressure across Number 1 Seal

2.3 RCP 1-1 Seal Water Low Differential Pressure **(Input 359)**

2.3.1 IF RCP seal water differential pressure is below 255 psid, THEN perform the following:

a. TRIP the Reactor. [ ]

b. While completing this step, GO TO EOP E-0, "Reactor Trip or Safety Injection." [ ]

c. TRIP RCP 1-1. [ ]

d. Place in "MANUAL" and CLOSE PCV-455A, Pressurizer Spray Valve. [ ]

e. IMPLEMENT OP AP-28, Section B, "RCP Number 1 Seal Failure". [ ]

2.3.2 Raise RCS pressure OR lower VCT pressure. [ ]

2.3.3 Probable Causes

- Low RCS pressure
- High VCT pressure
- Number 1 Seal Failure

2.4 RCP Radial Bearing Temperature High **(Input 361)**

2.4.1 IF RCS pressure is above 1000 psig, THEN GO TO OP AP-28, Section E, "Loss of CCW to an RCP or RCP High Temperature." [ ]

2.4.2 IF ALL of the following conditions exist:

- Seal injection flow to each RCP—ABOVE 8 gpm
- Number 1 Seal leakoff flow—BELOW 1 gpm
- RCS pressure—BETWEEN 100 and 1000 psig
- CVCS-1-8141A, B, C, D, Number 1 Seal Leakoff Valves—ALL OPEN

THEN OPEN CVCS-1-8142, Number 1 Seal Bypass Valve, as needed to lower temperature. [ ]

2.4.3 IF ALL of the above conditions are normal, THEN direct maintenance to check for an RTD failure. [ ]

SECTION E: LOSS OF CCW FLOW TO RCPs

SYMPTOMS

1. Thermal barrier and lube oil cooler cooling water return high temperature indication.
2. Possible Main Annunciator Alarms
   a. CCW HEADER C (PK01-08)
      1) RCP L.O. Clr CCW Flo Lo
      2) RCP Thermal Barrier CCW Flo Lo
      3) CCW Hdr C Flo Lo
   b. RCP No. _____ (PK05-01, 02, 03, 04)
      1) RCP _____ Temp PPC

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****
CAUTION: IF RCP No. 1 Seal Outlet Temperature exceeds 235°F **OR** RCP Radial Bearing Temperature exceeds 225°F, DO NOT restore RCP seal cooling.
****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. VERIFY CCW Flow To All RCP Lube Oil Coolers: a. Verify CCW Vlvs - OPEN • FCV-355 • FCV-356 • FCV-749 • FCV-363 b. RCP L.O. Clr CCW Flow LO Alarm (PK01-08) – NOT IN c. RCP Temp PPC Alarm (PK05-01, 02, 03, 04) - NOT IN	IF CCW Flow to RCP(s) CANNOT be restored to Lube Oil Coolers within 5 minutes, THEN 1) TRIP reactor. 2) TRIP affected RCP. 3) GO TO EOP E-O, REACTOR TRIP OR SAFETY INJECTION.

SECTION E: LOSS OF CCW FLOW TO RCPs (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. VERIFY RCP Seal Injection - IN SERVICE: • RCP Seal Injection Flow between 8 and 13 GPM • RCP Seal No. 1 Outlet Temp - NORMAL • RCP Radial Brg Outlet Temp - NORMAL	IF Both thermal barrier and seal injection flow are lost AND CANNOT be immediately restored, THEN 1) Manually TRIP reactor. 2) TRIP affected RCP(s). 3) GO TO EOP E-O, REACTOR TRIP OR SAFETY INJECTION while implementing the next three steps, as applicable. 4) Isolate seal injection to the RCP(s) before restarting a charging pump by locally CLOSING the following valves: • CVCS-8382A, Seal Inj Filter 1-2 (2-1) Outlet • CVCS-8382B, Seal Inj Filter 1-1 (2-2) Outlet • CVCS-8387A, Filter Bypass 5) If all RCPs are affected, close FCV-357, RCP Thermal Barrier CCW Return Isolation. 6) Initiate cooldown to MODE 5 per OP L-5 "Plant Cooldown From Minimum Load to Cold Shutdown" after exiting the EOPs
3. VERIFY CCW Flow To All RCP Thermal Barriers - NORMAL:	

****

CAUTION: If FCV-357 closed on high flow, do not attempt to open FCV-357 until condition causing high flow is cleared.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Verify FCV-357 did not close on high flow	a. GO TO Section B Step 5.b, Page 6.
b. Verify Thermal Barrier Return Vlvs FCV-750 and FCV-357 - OPEN	
c. Verify RCP Thermal Barrier CCW Flow Lo Alarm (PK01-08) - NOT IN	

SECTION B: CCW SYSTEM INLEAKAGE (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. DETERMINE Leak Location: (Continued)	
b. RCP Thermal barriers	b. Isolate as follows: 1) Verify FCV-750 CLOSED 2) Locally ISOLATE Thermal Barrier CCW return (inside containment) by closing as applicable: • RCP 1: CCW-234 • RCP 2: CCW-242 • RCP 3: CCW-251 • RCP 4: CCW-262 3) Monitor containment sump for expected level increase 4) IMPLEMENT OP AP-1, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE
c. Excess Letdown Heat Exchanger	c. Isolate as follows: 1) ISOLATE RCS flow to Heat Exchanger (VB2) • Close CVCS-8166 - OR - Close CVCS-8167 • Close HCV-123 2) ISOLATE CCW flow to Heat Exchanger: • Locally Close CCW-426 • Locally Close CCW-431 - OR - Close FCV-361 3) Adjust charging flow to minimum or restore normal letdown.

THIS STEP CONTINUED ON NEXT PAGE

SRO Question 79

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_1__
	K/A:	_____	APE 054 G2.1.33
	Importance Rating:		4.0

Proposed Question:

The crew is stabilizing the plant at approximately 50% following a ramp from 70% in accordance with AP-15, Loss of Feedwater Flow, due to the trip of a Main Feed pump.

Why must Chemistry be notified?

A. To determine current boron concentration.

B. To determine if RCS pH adjustment is required.

C. A sample for DOSE EQUIVALENT I-131 is required by Technical Specifications.

D. A sample for gross specific activity, $100/\bar{E}$, of the RCS is required by Technical Specifications.

Proposed Answer:
C. A sample for DOSE EQUIVALENT I-131 is required by Technical Specifications.

Explanation:
A incorrect, a boron concentration will not have changed.

B incorrect, pH not changed.

C correct, IAW Tech Spec SR 3.4.16, a sample for DE I-131 following a power change of 15% or more in one hour is required 2 to 6 hours after the power change.

D incorrect. This is determined on a periodic frequency, not after power changes.

Technical Reference(s):
OP AP-15, step 18 of section A.
Technical Specification 3.4.16

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

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:

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

Comments:

K/A: APE 054 G2.1.33 – Loss of Main Feedwater – ability to recognize indications for system operating parameters which are entry conditions for technical specifications. (4.0)

SECTION A: ONE MFP TRIPS WITH BOTH MFPs OPERATING (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

16. **Reset LOAD TRANSIENT BYPASS RELAY**[T35511]

17. **COMPLETE Secondary Realignment:**[T35511]

    a. Notify the Condensate Polisher Watch and CLOSE FCV-230.

    b. CLOSE FCV-55 AND place in AUTO.

    c. Place the following mode selector switches from CONT ONLY to AUTO:

    - TCV-23
    - FCV-31
    - FCV-30
    - LCV-12

    d. Shut down one condensate/booster pump set AND return to AUTO.

18. **NOTIFY Chemistry If Power Reduction Exceeded 15% Within One Hour**

19. **GO TO OP L-4, Normal Operation At Power:**

    a. Verify applicable steps have been performed and comply with applicable Precautions and Limitations.

- END -

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity > 1.0 µCi/gm.	-------------------NOTE----------------- LCO 3.0.4c is applicable. -- A.1 Verify DOSE EQUIVALENT I-131 specific activity within the acceptable region of Figure 3.4.16-1. <u>AND</u> A.2 Restore DOSE EQUIVALENT I-131 specific activity to within limit.	 Once per 4 hours 48 hours
B. Gross specific activity of the reactor coolant $100/\bar{E}$ µCi/gm.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 specific activity in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} to < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	7 days
SR 3.4.16.2	----------------------------NOTE-------------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period.
SR 3.4.16.3	----------------------------NOTE-------------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

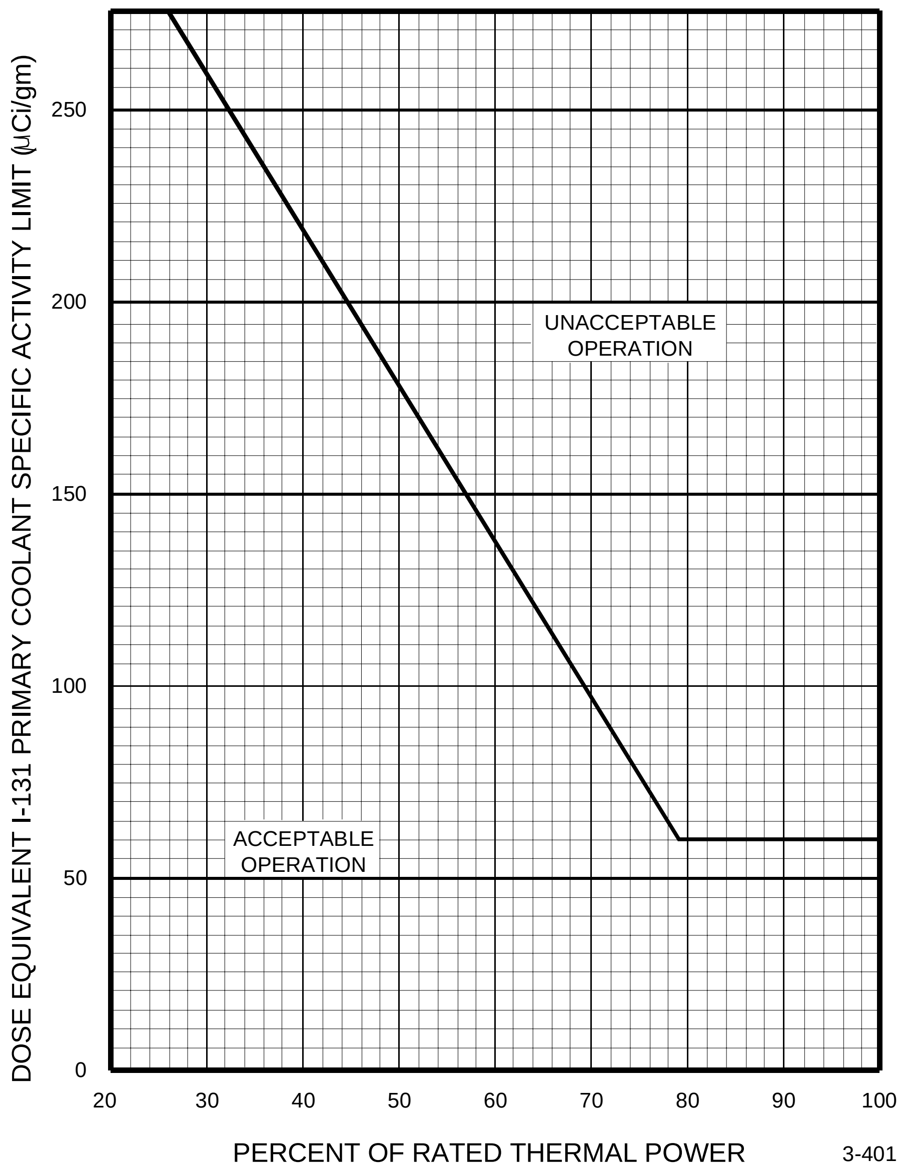


Figure 3.4.16-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1 μCI/GRAM DOSE EQUIVALENT I-131.

SRO Question 80

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_1__
	K/A:	_____	APE 058 AA2.03
	Importance Rating:		3.9

Proposed Question:

Unit 1 is at full power.

The following events occur:
- A loss of Vital DC bus 1-2 occurs and the crew enters AP-23, Loss of Vital DC Bus.
- The reactor is tripped. The Shift Foreman is performing the actions of AP-23.
- The Work Control Shift Foreman is implementing E-0, Reactor Trip or Safety Injection.

What action should be taken by the Work Control Shift Foreman at step 3 of E-0?

A. Verify from the BOPCO that Vital Buses F and H are energized and implement AP-27, Loss of a Vital 4 kV and/or 480 V Bus.

B. Verify the BOPCO has closed the breaker for Emergency Diesel 12 to energize bus G, that buses F and H are energized, then continue in E-0.

C. Verify from the BOPCO that Vital Buses F and H energized and GO TO EOP ECA-0.3.

D. Verify from the BOPCO that Vital Buses F and H energized and refer to EOP ECA-0.3 for restoration of Bus G while continuing in E-0.

Proposed Answer:

D. Verify from the BOPCO that Vital Buses F and H energized and refer to EOP ECA-0.3 for restoration of Bus G while continuing in E-0.

Explanation:
A incorrect. 2 vital buses energized from offsite, however, E-0 has action in the RNO to energize the de-energized bus/

B incorrect. 2 vital buses are energized. No control over the diesel at this time.

C incorrect. Do not transition to ECA-0.3. Immediate action step is to refer to ECA-0.3

D incorrect. Action is to refer to ECA-0.3

Technical Reference(s): E-0, step 3. AP-23, Loss of Vital DC Bus, and attachment B.

Proposed references to be provided to applicants during examination: None

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

Question Source:
Modified DCPP Bank A-0735

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: APE 058 AA2.03 - Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems. (3.9)

With the unit at 100% power, a loss of a vital DC bus will:

A. Result in loss of a 4kV vital bus when the unit trips.

B. Prevent a manual Unit Trip.

C. Result in a loss of DC power to the main generator air side seal oil back-up pump.

D. Prevent a train of phase "A" isolation valves from closing.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

7116	Explain the consequences of loss of DC vital bus

Reference Id: A-0735
Must appear: No
Status: Active
User Text: 7116.07633
User Number 1: 0000003.70
User Number 2: 0000004.30
Difficulty: 3.00
Time to complete: 3
Topic: LPA-23 Loads supplied by vital DC
Cross Reference: DUTY AREA 63, AP-23 APPX A/B/C

TITLE: Loss of Vital DC Bus **1 AND 2**

05/02/03
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure covers the steps to be taken in the event of a loss of one vital DC electrical bus. Steps to stabilize the plant are presented first. Loss of each DC bus and subsequent recovery will then be covered separately in appendices. The appendices are written for loss of DC during any mode. It should be noted that vital AC power associated with the lost DC bus may also be lost and corrective actions are taken in this procedure.

2. SYMPTOMS

1.1 Possible Annunciator Alarms

1.1.1 PK20-18 "125V DC Panels 11 (21), 12 (22), or 13 (23)"

(0122) "125V DC PNL PD-11 (21) FDR BKR OPN OR UV"
(0202) "125V DC PNL PD-12 (22) FDR BKR OPN OR UV"
(0278) "125V DC BUS 13 (23) SYS UV"

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.

APPENDIX B

LOSS OF VITAL DC BUS 12 (22)
RECOVERY AND RESTORATION

SYMPTOMS

a) Train B FW Reg solenoids de-energize, fail closed, plant trips on Lo-Lo SG level

b) 35% and 40% Steam Dumps fail closed, 10% Steam Dumps functional

c) MFP #2 trips, MFP #1 running with recircs failed closed, manual trip of MFPs required

d) PCBs fail to open, DEHC sense PCBs open and drive Gov Valvs closed which may cause Pzr Hi Pressure Trip if this occurs from high power level, Unit trip required

e) 4kV Bus G does not transfer to startup

- PY-12 lost with Bus G
- Voltage transient of PY-14 during transfer to startup could result in 2/4 SI and MSLI

f) 4kV and 12kV Bus Es fail to transfer to startup

g) Letdown isolated, Excess LD not available, Pressurizer level controlled by start/stop of charging pump

h) PORV-455C failed closed

i) Pzr Htrs Groups 3 and 4 unavailable due to loss of power, vital backup power available for Group 3 only

j) No control of D/G 1-2

INSTRUCTIONS

****

CAUTION: Reenergizing a DC bus without first stripping the bus of its loads may result in unexpected adverse equipment operation.

****

1. Evaluate plant status and take appropriate action for loss of the following components/functions:

   a. CVCS-8471 and Charging Injection are the only boration flowpaths available when 480V Bus G is deenergized.

   b. If the Turbine Driven AFW pump 1-1 (2-1) is needed:

      1) FCV-95, Steam Isolation valve, will have to be manually opened.

      2) Turbine Driven AFW pump motor operated valves LCV-106, 107, 108, and 109 will be deenergized if 480V bus G is lost and personnel will have to be dispatched locally to control AFW flow.

   c. Place PY17 on backup supply from 480V bus F to keep Rotary Air Compressor master unloader energized if PY-17 is de-energized.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. VERIFY Vital 4KV Bus Status:	
a. Verify ALL vital 4KV buses - ENERGIZED	a. Perform the following: 1) IF All vital buses are deenergized, THEN GO TO EOP ECA-0.0, LOSS OF ALL VITAL AC POWER. 2) IF Only one vital bus is energized, THEN IMPLEMENT EOP ECA-0.3, RESTORE 4KV BUSES. 3) IF Two vital buses are energized, THEN Perform the following: o REFER TO EOP ECA-0.3, RESTORE 4KV BUSES, while continuing in this procedure. o REFER TO Appendix A for guidance on available ESF equipment. -

SRO Question 81

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_1__
	K/A:	_____	W/E11 EA2.2
	Importance Rating:		4.2

Proposed Question:

GIVEN:
- The crew is currently implementing ECA-1.1, Loss of Emergency Coolant Recirculation.
- Bus F is de-energized
- RWST level is 30%
- RCS Pressure is 450 psig
- Containment pressure is 24 psig.

Per ECA-1.1, what action, if any, needs to be taken to achieve the REQUIRED combination of Containment Fan Cooler Units and Containment Spray Pumps for the current plant conditions?

A. No action required, the required number of Containment Spray pumps are running.

B. Stop either Containment Spray pump 11 or 12.

C. Stop both Containment Spray pumps 11 and 12.

D. Stop Containment Spray pump 12, Containment Spray pump 11 is de-energized.

Proposed Answer:

B. Stop either Containment Spray pump 11 or 12.

Explanation:
A incorrect. CS pumps 11 and 12 are powered from Bus G and H respectively. Required number is 1 (2 CFCUs are de-energize), one must be secured. This would be correct if either pump was powered from bus F.

B correct. CS pumps are both running. Required number for plant conditions is 1 because 2 CFCUs are powered from Bus F.

C incorrect. This would be correct if 4 CFCUs were still available, instead of 3.

D incorrect. Neither CS pump is powered from bus F.

Technical Reference(s): ECA-1.1 step 6 and attachment A

Proposed references to be provided to applicants during examination: ECA-1.1 (or at least page 3 (step 6)

Learning Objective: SIM0152 Demonstrate the Proper Use of the EOPs

Question Source:
Modified - Indian Point NRC Exam 12/04

Question History: Indian Point 12/04

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

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: W/E11 EA2.2 Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. (4.2)

ACTION/EXPECTED RESPONSE | **RESPONSE NOT OBTAINED**

5. CHECK RWST Level GREATER THAN 4%

GO TO Step 29 (Page 15).

6. DETERMINE Containment Spray Requirements:

a. Determine number of Spray Pps required using table:

RWST LEVEL	CONTAINMENT PRESSURE	TOTAL NUMBER OF FAN COOLERS RUNNING	NUMBER OF SPRAY PUMPS REQUIRED
GREATER THAN 33%	GREATER THAN 47 PSIG	N/A	2
	BETWEEN 22 PSIG AND 47 PSIG	2 or LESS	2
		3 or 4	1
		ALL	0
	LESS THAN 22 PSIG	N/A	0
LESS THAN 33%	GREATER THAN 22 PSIG	3 or LESS	1
		4 or 5	0
	LESS THAN 22 PSIG	N/A	0
LESS THAN 4%	N/A	N/A	0

b. Spray Pps running - EQUAL TO NUMBER REQUIRED

b. Reset Spray Pps and manually operate Spray Pps as necessary to obtain number required.

IF Containment Pressure increases,

THEN Start additional Containment Spray Pps.

IF Spray Pps CANNOT be operated,

THEN Continue efforts to establish Containment Fan Cooler Operation.

- -

APPENDIX A

BLACKOUT EMERGENCY LOADING OF VITAL BUSES

1. If the vital buses lose voltage prior to resetting the Safety Injection Signal, the vital buses will automatically load the vital equipment given below. Verify the equipment has been loaded by observing breaker lights on the control board.

2. If the vital buses lose voltage after the Safety Injection Signal has been reset, load or verify loaded the equipment given below onto the vital buses manually. Allow approximately 4 seconds between loading of each piece of equipment onto a given vital bus. Load or verify that the CFCU are running in Low Speed.

VITAL BUS F (D/G No. 3)	VITAL BUS G (D/G No. 2)	VITAL BUS H (D/G No. 1)
CCP No. 1	RHR Pp No. 1	SI Pp No. 2
SI Pp No. 1	CCW Pp No. 2	RHR Pp No. 2
CCW Pp No.1	ASW Pp No. 2	CCW Pp No. 3
ASW Pp No.1	CCP No. 2	AFW Pp No. 2
AFW Pp No.3		
CFCU No. 1	CFCU No. 3	CFCU No. 4
CFCU No. 2	CFCU No. 5	
MCC F	MCC G	MCC H

3. Load the Containment Spray Pps only if they were running prior to the blackout.

VITAL BUS G	VITAL BUS H
Contmt Spray Pp No. 1	Contmt Spray Pp No. 2

The following sequence of events occurs:

"Unit 2 was operating at 100% power
"Small break LOCA occurred 25 minutes ago.
"The Team is currently implementing ECA 1.1, Loss of Emergency Coolant Recirculation due to loss of recirculation capability.
"RCS Pressure is 450 psig
"Containment pressure is 4 psig.

Given the attached reference from ECA 1.1, Loss Of Containment Sump Recirculation, which of the following indicates the REQUIRED correct combination of Containment Fan Cooler Units and Containment Spray Pumps that are required to be operating under these conditions?

D.0 FCU, 2 Spray Pumps

A.4 FCUs, 0 Spray Pumps

B.1 FCU, 1 Spray Pump

C.0 FCU, 1 Spray Pump

SRO Question 82

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_2__
	K/A:	_____	APE 003 G2.4.49
	Importance Rating:		4.0

Proposed Question:

Unit 1 is recovering from a dropped control rod in accordance with OP AP-12C, “Dropped Control Rod.”

At step 13, while opening the lift coils to recover the dropped rod, a second control rod drops into the core.

The Shift Foreman directs the CO to terminate the rod recovery.

What is the next action the Shift Foreman should take?

A. Direct a trip of the reactor and GO TO EOP E-0, Reactor Trip or Safety Injection.

B. Call the Plant Reactor Engineering Group for guidance.

C. Direct the operator not to reset any URGENT alarms and contact I&C immediately to investigate the problem.

D. Within one hour, verify SDM to be within the limits provided in the COLR and make preparations to be in MODE 3 within the next 6 hours.

Proposed Answer:

A. Direct a trip of the reactor and GO TO EP E-0, Reactor Trip or Safety Injection.

Explanation:

Only A is correct. Currently the crew would be at step 13 of the procedure, preparing the dropped rod for recovery. When the second one drops, there are now two dropped rods. Per step 1 of the procedure (not a continuous action), if there are 2 dropped rods, then the reactor is tripped. It would not be appropriate to continue with the recovery with a second dropped rod.

Technical Reference(s): AP-12C step 1 and step 13, rev10.

Proposed references to be provided to applicants during examination: None

Learning Objective: 9934 - Explain the operator actions if more than 1 control rod drops

Question Source:
Bank DCPP P-5466

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:

55.43.6 - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Comments:

K/A: APE 003 G2.4.49 – Dropped Control Rod, Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (4.0)

Unit 1 is recovering from a dropped control rod in accordance with OP AP-12C, DROPPED CONTROL ROD. While attempting to recover the dropped rod, a second control rod drops into the core. Your actions are to:

A. Manually trip the reactor and GO TO EP E-O, REACTOR TRIP OR SAFETY INJECTION.

B. Initiate a technical specification shutdown in accordance with Tech. Specs. 3.1.3.1.d.

C. Terminate rod movement, do NOT reset any URGENT alarms, and contact I&C immediately to investigate the problem.

D. Terminate rod motion and call the Plant Reactor Engineering Group for guidance.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

9934	Explain the operator actions if more than 1 control rod drops

Reference Id: P-5466
Must appear: No
Status: Active
User Text: 9934.010013
User Number 1: 0000003.40
User Number 2: 0000004.60
Difficulty: 2.00
Time to complete: 3
Topic: AP-12C, Actions to be done if two rods drop.
Cross Reference: LPA-12, AP-12C
Comment: Reviewed 11/8/96 jpsj

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ONLY One Control Rod Dropped	Trip the Reactor and GO TO EOP E-0, REACTOR TRIP OR SAFETY INJECTION
2. PLACE Rod Control in MANUAL	
3. STOP Any Load Change In Progress AND Allow Conditions To Stabilize:	
a. Reactor is critical.	a. Place the reactor in Mode 3 by fully inserting the control rods. GO TO OP L-5, PLANT COOLDOWN FROM MINIMUM LOAD TO COLD SHUTDOWN

NOTE: The initial determination of SDM (STP R-19) is simplified if dilution is not used to raise temperature.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. STOP Any RCS Boration or Dilution In Progress	
5. ADJUST Turbine Load To Match TAVG AND TREF	
6. CHECK Axial Flux Difference Within Tech Spec Limits	Refer to Tech Spec 3.2.3.
7. Calculate QPTR per STP R-25:	
a. Verify LESS than 1.02.	a. Refer to Tech Spec 3.2.4.A.
8. VERIFY SDM within COLR limits:	
a. Perform STP R-19 Data Sheet 4, within 1 hour	Refer to Tech Spec 3.1.1 and 3.1.4
9. VERIFY Rod Control System Had No Urgent Failure:	
a. Verify Rod Cont Urgent Failure (PK03-17) - OFF	a. If PK03-17 is ON, THEN 1. Do not attempt to move rods or reset the Urgent Failure. 2. Contact maintenance for trouble shooting. 3. Refer to AR PK03-17, ROD CONT URGENT FAILURE.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. FIND And Correct Cause Of The Dropped Rod:

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

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

b. Contact maintenance to initiate troubleshooting and repairs

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

11. CONTACT Reactor Engineering Regarding the Dropped Rod to Obtain:[T30977]

a. Guidance on rate of control rod movement during recovery. Inform Reactor Engineering if fuel has not been conditioned at the current power level and estimated time rod has been misaligned.

b. Power level at which recovery should be performed

c. Use of other control rods during recovery.

12. ESTABLISH Initial Recovery Conditions:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check the time since the rod dropped - LESS THAN ONE hour	a. Reduce reactor power to LESS THAN 50%. Continue with Step 12.c when power is LESS THAN 50%.
b. Reduce power as necessary such that the steady state power level attained after the rod is recovered is LESS THAN 90% Reactor Power	
c. Adjust T_{AVG} 1.0 to 1.5°F below T_{REF} by inserting control bank rods as necessary	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

13. PREPARE For Rod Withdrawal:

a. Select the affected rod bank on the Bank Selector Switch

b. Record the step counter position on the affected group

Bank ____ Group _____ Step _____

c. Reset the step counter to zero on the affected group only

d. Locally open lift coil disconnect switches on all rods in the affected bank except the dropped rod. (Lift Coil Disconnect Cabinet 115' Elev Aux Bldg)

e. Check dropped rod is in a control bank

RESPONSE NOT OBTAINED: e. GO TO Step 13.g (page 5).

NOTE: A ROD LO LO INSERTION LIMIT ALARM (PK03-14) may occur during the performance of this step.

f. Locally reset the pulse to analog converter to ZERO for the affected control bank as follows: (P/A Converter Cabinet, 115' Elev, Aux Bldg)

1) Select the affected bank on the digital display switch

2) Record the position displayed

Bank ___________

Position _________

3) Hold the AUTO-MANUAL switch in the MANUAL position

4) Depress the DOWN push button the required number of times to rezero the digital display

5) Release the AUTO-MANUAL switch and verify it spring returns to AUTO position

g. UPDATE the Plant Process Computer Rod Bank Step Count to ZERO for the affected Bank. (Refer to Appendix B for address)

SRO Question 83

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_2__
	K/A:	_____	APE 036 AA2.02
	Importance Rating:		4.1

Proposed Question:

Unit 1 refueling is in progress.

The RED overload light comes on at the Containment Manipulator Crane Console. The SRO suspects the fuel assembly being moved has been damaged.

Which of the following would be a positive indication to the SRO that the fuel element has been damaged?

A. Increased drag indicated on the Dillon Cell.

B. Increasing radiation levels on RE-58 and RE-59.

C. Report from the Control Room of increasing Source Range counts.

D. Gas bubbles coming to the surface of the water in the refueling cavity.

Proposed Answer:

D. Gas bubbles coming to the surface of the water in the refueling cavity.

Explanation:

A incorrect, increased drag would be an indication of the assembly possibly resisting movement but not that it has actually been damaged. NOTE, if the limit is exceeded, then it’s a possible, additional indication.

B incorrect, these radiation monitors are in the SFP area.

C incorrect, this would be an indication of approaching criticality but not fuel damage.

D correct, a damaged fuel assembly would release the gas in the assembly. The gas comes to the surface which would indicate the integrity of the assembly is compromised.

Technical Reference(s): AP-21, rev 10, step 2 - Symptoms.

Proposed references to be provided to applicants during examination: None

Learning Objective: 6619 - Explain the actions for fuel damage, actual or suspected

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.7 - Fuel handling facilities and procedures.

Comments:

K/A: APE 036 AA2.02 - Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurrence of a fuel handling incident (4.1)

Irradiated Fuel Damage

05/06/03
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

This procedure gives guidance in the event an irradiated fuel assembly is damaged during fuel handling or is found to have existing damage sustained from the previous cycle.

2. SYMPTOMS

2.1 Refueling crew observation of fuel damage.

2.2 While in the process of withdrawing a fuel assembly from the core, the refueling crew might observe release of gas bubbles from the fuel assembly with the following possible additional symptoms:

2.2.1 Drag withdrawal limit exceeded (Dillon Cell at Manipulator Crane Control Console).

2.2.2 Red overload light is ON (Manipulator Crane Control Console).

2.2.3 Null meter indicates excessive positive drag accompanied by an audible alarm (Manipulator Crane Control Console).

2.2.4 Auto Stop of upward hoist movement.

2.3 Local alarm/revolving red beacon light from area radiation monitors or portable continuous air monitors (e.g., SPING-3A/AMS-3/IM-11), as applicable.

2.4 Fuel Handling Building (FHB) evacuation horn is automatically sounded, if fuel handling incident is in the FHB.

2.5 Possible Main Annunciator Alarms:

2.5.1 FHB HIGH RADIATION RE-58 and 59 (PK11-10)

2.5.2 HIGH RADIATION (PK11-21)

2.5.3 CONTMT VENT ISOLATION (PK02-06)

2.5.4 If in Unit 1

CONTMT RADIATION (PK11-19)

2.5.5 If in Unit 2

HI-LEVEL RAD MONITOR SYSTEM (PK11-19)

2.6 Possible Containment Ventilation Isolation (CVI) if fuel handling incident is in Containment.

2.7 FHB ventilation transfers to Iodine removal mode if incident is in FHB. (As this is an expected action along with the automatic sounding of the FHB evacuation horn whenever RE-58 or 59 alarm is activated, Control Room call to the crew is to confirm the incident.)

SRO Question 84

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	__1__
	Group:	_____	__2__
	K/A:		APE 037 G2.1.33
	Importance Rating:		4.0

Proposed Question:

GIVEN:

- Unit 1 is at full power
- RCS IDENTIFIED and primary to secondary leakage at 1200 is as follows:
  - Total IDENTIFIED LEAKAGE is 9.5 gpm
  - Primary to Seconday leakages:
    - 0.03 gpm to Steam Generator 11
    - 0.03 gpm to Steam Generator 12
    - 0.00 gpm to Steam Generator 13
    - 0.04 gpm to Steam Generator 14

Subsequent Steam Generator leakages are determined as follows:

TIME	1230	1300	1330
Steam Generators 11	0.06	0.08	0.08
Steam Generator 12	0.08	0.09	0.11
Steam Generator 13	0.00	0.00	0.00
Steam Generator 14	0.06	0.08	0.09

For the times above, when, if at all, was the Shift Foreman required to enter LCO 3.4.13, RCS OPERATIONAL Leakage, ACTION A, for leakage in excess of Technical Specifications?

A. 1230

B. 1300

C. 1330

D. No leakage limits have been exceeded as of 1330.

Proposed Answer:

C. 1330

Explanation:

A incorrect. At 1230, Identified is 9.4 gpm + 0.20 gpm or 9.6 gpm. No steam generators are above 150 gpd.

B incorrect. At 1300 Identified is 9.4 + 0.25 or 9.65. No steam generators are above 150 gpd.

C correct. At 1330, identified is 9.4 + 0.28 for a total of 9.68, however, primary to secondary limits have been exceeded in the 12 steam generator - limit is 150 gpd or .104 gpm. Currently leakage is 0.11 gpm or 158.4 gpd.

D incorrect. The LCO was entered at 1330.

Technical Reference(s): TS 3.4.13, TS definitions

Proposed references to be provided to applicants during examination: None

Learning Objective: 9697D - Identify 3.4 Technical Specification LCOs

Question Source:
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:

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

Comments:

K/A: APE 037 G2.1.33 – Steam Generator Tube Leak - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (4.0)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR	B.1 Be in MODE 3. AND	6 hours
Pressure boundary LEAKAGE exists.	B.2 Be in MODE 5.	36 hours

* For MODES 3 and 4, if steam generator water samples indicate less than the minimum detectable activity of 5.0 E-7 microcuries/ml for principal gamma emitters, the leakage requirement of specification 3.4.13.d. may be considered met.

LEAKAGE

LEAKAGE shall be:

a. *Identified LEAKAGE*

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

LEAKAGE (continued)

3. *Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System.*

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

SRO Question 85

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_1__
	Group:	_____	_2__
	K/A:	_____	W/E16 EA2.1
	Importance Rating:		3.3

Proposed Question:

GIVEN:

- A large LOCA has occurred
- Containment pressure is 3 psig
- ECCS system is aligned for Cold Leg Recirculation
- The BOPCO reports Containment radiation is 10 R/hr on RM-30/31.
- After leaving E-1.3, Transfer to Cold Leg Recirculation, the Shift Foreman enters FR-Z.3, Response to High Containment Radiation Level

When informed of the Containment radiation levels, the TSC recommends starting a Containment Spray pump to reduce the radiation levels.

Which of the following actions should be taken by the Shift Foreman?

A. Direct the BOPCO to start a Containment Spray pump.

B. Inform the TSC that Containment Spray is not an option at this time and return to E-1, Loss of Reactor or Secondary Coolant.

C. Refer to FR-Z.1, Response to High Containment Pressure, for guidance on starting a Containment Spray pump.

D. Refer to E-1.3, Transfer to Cold Leg Recirculation, for guidance on starting a Containment Spray pump.

Proposed Answer:

D. Refer to E-1.3, Transfer to Cold Leg Recirculation, for guidance on starting a Containment Spray pump.

Explanation:

A incorrect. Because the plant is in recirc, starting the pump at this time will be starting it without a suction path.

B incorrect. Containment spray is an option.

C incorrect. Guidance is in E-1.3.

D correct. Because of the current plant alignment, guidance in E-1.3 is necessary.

Technical Reference(s): FR-Z.3, page 2.

Proposed references to be provided to applicants during examination: None

Learning Objective: 6177 - State the means of reducing containment radiation levels

Question Source:
Bank P-49251, modified

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: E16 EA2.1 - Ability to determine and interpret the following as they apply to the (High Containment Radiation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (3.3)

1.00

A LBLOCA has occurred, and the ECCS system is in the Cold Leg Recirc phase. Containment radiation is noted to be 7 R/hr on RM-30/31. The EEC has identified the status of the CSFSTs and the TSC has recommended starting a Containment Spray pump. Which of the following should be done:

A. Refer to EOP E-1.3, Transfer to Cold Leg Recirc, for instructions.

B. Start the containment spray pumps.

C. Stop the iodine removal units, then start the containment spray pumps.

D. Return to the procedure and step in effect.

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

68453	Explain the basic principles of operation for the Containment Fan Cooling Units.
69406	Explain the basic principles of operation for the Containment Systems.
6177	State the means of reducing containment radiation levels

Reference Id: P-49251
Must appear: No
Status: Active
User Text:
User Number 1:
User Number 2:
Difficulty: 3.00
Time to complete: 2

*** UNCONTROLLED PROCEDURE - DO NOT USE TO PERFORM WORK or ISSUE FOR USE ***

TITLE: Response to High Containment Radiation Level

1

06/29/06
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure provides actions to respond to High Containment Radiation Level.

1.2 The major actions in EOP FR-Z.3 are:

- Verify containment ventilation isolation,
- Verify Containment Fan Cooler Units are in service
- Notify the plant engineering staff of containment radiation level.

2. VERIFY ENTRY CONDITION FOR EOP FR-Z.3

2.1 EOP F-0.5, YELLOW Condition

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
1. CHECK Containment Vent Isol:	Do one of the following:
a. Containment Vent Isol portion of Monitor Light Box B: • Red Activated Light - ON • White Status Lights - OFF	• Manually Actuate CONTMT ISOL PHASE A OR • Manually Close the CVI Vlvs with White Status Lights - ON

****

CAUTION: The Iodine Removal Units must not be placed in service during a Loss of Primary OR Secondary Coolant.

****

2. CHECK Containment Fan Cooler Units, RUNNING IN SLOW SPEED:

a. Monitor Light Box C:

- Red Activated Light - ON
- White Status Light - OFF

NOTE: If the TSC recommends starting or restarting Containment Spray to reduce the Radiation Level and the ECCS System is in the Recirc Mode, REFER TO EOP E-1.3, TRANSFER TO COLD LEG RECIRCULATION.

3. NOTIFY TSC Of Containment Radiation Level To Obtain Recommended Action:

4. RETURN TO Procedure And Step In Effect

- END -

3. APPENDICES

   None

4. ATTACHMENTS

   None

SRO Question 86

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2__
	Group:	_____	_1___
	K/A:		003 A2.05
	Importance Rating:		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 Shift Foreman 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 increase seal leakoff flow.

Technical Reference(s): E-0.2, attachment B

Proposed references to be provided to applicants during examination: None

Learning Objective: 4892 - State the cause/effect relationship between VCT and RCPs

Question Source:
Diablo Canyon 2005 SRO

Question History: Last NRC Exam: DCPP 2005

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

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: 003 A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of VCT pressure on RCP seal leakoff flows (2.8)

Previous exam question

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	2
	Group #	_____	1
	K/A #		003 A2.05
	Importance		2.8

The crew is making preparations to start a RCP using Attachment B in E-0.2, "Natural Circulation Cooldown". The CO reports Seal Leakoff flow is low. All other conditions for starting the RCP are met.

Which of the following actions should the SFM direct the operator to perform?

A. Start the RCP.

B. Increase VCT level.

C. Increase charging flow.

D. Decrease VCT pressure.

Proposed Answer:

D. Decrease VCT pressure.

Explanation:
A incorrect, this is not a procedure in which RCPs are started if normal conditions are not met.
B incorrect, increasing VCT level will increase VCT pressure, which will further decrease seal leakoff flow.
C incorrect, increasing charging will not affect appreciably affect seal leakoff.
D correct, decreasing VCT pressure will decrease the DP and increase seal leakoff flow.

Technical Reference(s):
E-0.2, Natural Circulation Cooldown, attachment B, Restart of Reactor Coolant Pump

Proposed references to be provided to applicants during examination: none

Learning Objective: 4892 - State the cause/effect relationship between VCT and RCPs

Question Source:

New X

Question History: Last NRC Exam N/A

Question Cognitive Level:

Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

K/A: 003 A2.05 – Reactor Coolant Pump - Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of VCT pressure on RCP seal leakoff flows

Appendix B: Restart of Reactor Coolant Pump

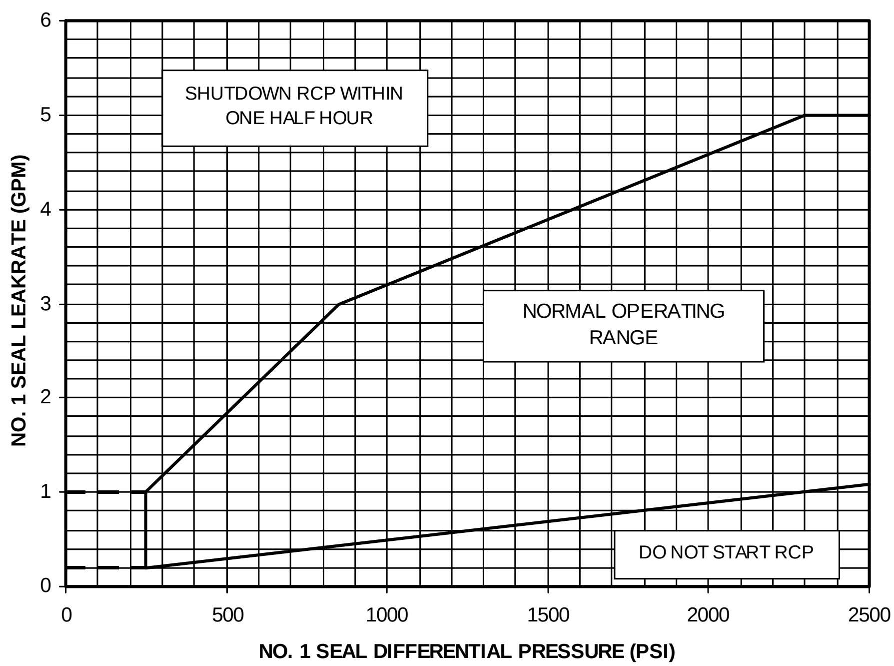
NOTE: RCP No. 2 (preferred) or RCP No.1 should be given priority for purpose of normal PZR Spray. If RCP 1 is desired for PZR Spray then RCP 3 and 4 should also be started to provide sufficient PZR Spray DELTA-P.

****

CAUTION: RCP Seals may be damaged if RCP Seal Cooling was lost AND an RCP is started WITHOUT an RCP Seal Status Evaluation.

****

1. Start oil lift pump and run for 2 minutes.
2. Reset SI, Phase A or Phase B as necessary to provide RCP support systems.
3. Verify the following CCW valves open to RCP thermal barrier and oil coolers:
    - FCV-355
    - FCV-356
    - FCV-749
    - FCV-363
    - FCV-750
    - FCV-357
4. Verify seal DP GREATER THAN 255 PSID, Depressurize VCT as necessary.
5. Verify Seal Injection flow between 8 GPM TO 13 GPM.
6. Verify Seal Leak Off flow WITHIN limits shown on graph.
7. Verify closed PCV-455A and B, Normal PZR Spray Vlvs, AUTO optional.
8. Start RCP. Observe RCP Pump Motor AMPS and FLOW to verify NORMAL operation.



SRO Question 87

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2___
	Group:	_____	_1___
	K/A:		006 G2.4.31
	Importance Rating:		3.4

Proposed Question:

Unit 1 is at full power.

The following events occur:
- PK 02-05, ACCUM PRESSURE HI-LO and PK 02-10, ACCUM LEVEL HI-LO alarm
- The operator reports Accumulator 1-1 pressure is 590 psig, level is 59% and both are decreasing slowly.
- The level and pressure stabilize at 585 psig and level is 50%, once the Nuclear Operator checks NSS-1-9352A, Accumulator Sample Line Isolation, closed.

Which of the following describes the OPERABILITY status of Accumulator 1-1 and the action that should be taken first by the Shift Foreman?

A. Inoperable. Refer to AR PK02-05 to raise pressure.

B. Inoperable. Refer to AR PK02-10, to raise level.

C. OPERABLE. Refer to AR PK02-05 to raise pressure.

D. OPERABLE. Refer to AR PK02-10, to raise level.

Proposed Answer:

B. Inoperable. Refer to AR PK02-10, to raise level.

Explanation:

A incorrect. Accumulator is inoperable, however, the action that should be taken is to raise level and then if necessary adjust pressure.

B correct. Accumulator is inoperable. The action to take is to raise level, which will raise pressure. Once level is restored, pressure is adjusted, if necessary.

C and D incorrect. Per STP 1-1A, TS level is 52%. Pressure is above the required pressure, however, Accumulator are inoperable due to the low level.

Technical Reference(s): TS 3.5.1, STP 1A, PK02-05 and PK02-10

Proposed references to be provided to applicants during examination: STP I-1A (at least Accumulator page) and TS 3.5.1

Learning Objective: 35326 - State the ECCS parameters that produce alarms in the control room.

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: 006 G2.4.31 – ECCS - Knowledge of annunciators alarms and indications, and use of the response instructions. (3.4)

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. AND C.2 Reduce RCS pressure to $\leq$ 1000 psig.	6 hours 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)

APPL MODE	TECH SPEC REFERENCE	**A. CHECK / VERIFICATION FROM CONTROL ROOM**	PERF
1	SR 3.5.1.2 SR 3.5.1.4	1. RCS Accumulator Volume	

a. Record current RCS accumulator volume and latest STP C-2 volume, then calculate volume change.

NOTE: PPC Quality Code color must not be light blue (cyan) for use in this surveillance.

Check the indicator used.

RCS Accumulator	RCS ACCUMULATOR VB2 IND OR PPC Point	CURRENT LEVEL %		C-2 LEVEL %		% Level Change	PERF
1-1	LI-950 [ ] LI950R [ ]	______	-	______	=	_____	
	LI-951 [ ] LI951R [ ]	______	-	______	=	_____	
1-2	LI-952 [ ] LI952R [ ]	______	-	______	=	_____	
	LI-953 [ ] LI953R [ ]	______	-	______	=	_____	
1-3	LI-954 [ ] LI954R [ ]	______	-	______	=	_____	
	LI-955 [ ] LI955R [ ]	______	-	______	=	_____	
1-4	LI-956 [ ] LI956R [ ]	______	-	______	=	_____	
	LI-957 [ ] LI957R [ ]	______	-	______	=	_____	_____

b. Verify all current RCS accumulator levels are ≥ 59% and ≤ 75%. [52% to 82%] ______

c. Verify all RCS accumulator level changes are ≤ +3% [5.6%]. ______

d. If any operational draining or filling of an accumulator occurs or if level increase is unacceptable, request chemistry to perform STP C-2 within 6 hours for affected accumulators. N/A [ ] ______

ACCUM PRESSURE HI-LO

07/06/05
Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
351	Accum 1-1 Press Lo PC-961B	PC961B	< 595.5 psig	2.1
404	Accum 1-3 Press Lo PC-965B	PC965B	< 595.5 psig	2.1
405	Accum 1-4 Press Lo PC-967B	PC967B	< 595.5 psig	2.1
409	Accum 1-2 Press Lo PC-963B	PC963B	< 595.5 psig	2.1
1381	Accum 1-1 Press Hi PC-961A	PC961A	> 647.5 psig	2.1
1386	Accum 1-2 Press Hi PC-963A	PC963A	> 647.5 psig	2.1
1388	Accum 1-1 Press Hi PC-960A	PC960A	> 647.5 psig	2.1
1389	Accum 1-1 Press Lo PC-960B	PC960B	< 595.5 psig	2.1
1395	Accum 1-2 Press Hi PC-962A	PC962A	> 647.5 psig	2.1
1396	Accum 1-2 Press Lo PC-962B	PC962B	< 595.5 psig	2.1
1406	Accum 1-3 Press Hi PC-965A	PC965A	> 647.5 psig	2.1
1407	Accum 1-4 Press Hi PC-967A	PC967A	> 647.5 psig	2.1
1415	Accum 1-3 Press Hi PC-964A	PC964A	> 647.5 psig	2.1
1416	Accum 1-3 Press Lo PC-964B	PC964B	< 595.5 psig	2.1
1417	Accum 1-4 Press Hi PC-966A	PC966A	> 647.5 psig	2.1
1418	Accum 1-4 Press Lo PC-966B	PC966B	< 595.5 psig	2.1

2. OPERATOR ACTIONS

2.1 General Actions **(All Inputs)**

NOTE:	The high and low pressure alarm setpoints are more accurate than the Control Board instrument readings.

2.1.1 Check both pressure indicators to confirm the alarm and the high or low pressure condition. [ ]

2.1.2 REFER TO TS 3.5.1 to determine accumulator OPERABILITY. [ ]

2.1.3 IF Annunciator PK02-10, "ACCUM LEVEL HI-LO," is ON,
THEN GO TO AR PK02-10, "Accum Level Hi-Lo," for further action. [ ]

2.1.4 GO TO the applicable section:

IF alarm is due to:	THEN GO TO:	
High Pressure	Section 2.2	[ ]
Low pressure	Section 2.3	[ ]

2.2 High Pressure **(Inputs 1381, 1386, 1388, 1395, 1406, 1407, 1415, 1417)**

2.2.1 IF high pressure is due to a level increase
AND attempting to lower pressure by draining is desired,
THEN perform the following:

a. OPEN accumulator drain to RCDT valve on affected accumulator until accumulator level reaches approximately 67%: [ ]

- Accumulator 1: SI-1-8876A
- Accumulator 2: SI-1-8876B
- Accumulator 3: SI-1-8876C
- Accumulator 4: SI-1-8876D

b. Direct chemistry to sample accumulator. [ ]

c. Determine if either of the following procedures should be performed:

- STP V-5A1, "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8819 A-D and 8956 A-D" [ ]
- STP V-5A2, "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8948 A-D and 8818 A-D" [ ]

2.2.2 Depressurize the affected accumulator as necessary PER OP B-3B:I, "Accumulators - Fill and Pressurize". [ ]

> **NOTE**: A malfunction in SI-1-PCV-199, N_2 Pressure Regulator To Accumulators, could cause high pressure nitrogen to enter the accumulator through a leaking fill and vent valve.

2.2.3 IF after depressurizing, accumulator pressure continues to rise, THEN dispatch an Operator to check SI-1-PCV-199, N_2 Pressure Regulator To Accumulators, for possible malfunction. [ ]

2.2.4 Probable Causes

- Accumulator and/or associated gas piping/valves leakage
- Accumulator level increase due to temperature change

2.3 Low Pressure **(Inputs 351, 404, 405, 409, 1389, 1396, 1416, 1418)**

2.3.1 IF low pressure is due to a level drop
AND attempting to raise pressure by filling is desired,
THEN fill the affected accumulator PER OP B-3B:I, "Accumulators - Fill and Pressurize". [ ]

2.3.2 Attempt to pressurize the affected accumulator as necessary PER OP B-3B:I, "Accumulators - Fill and Pressurize". [ ]

2.3.3 IF the nitrogen regulator did not raise accumulator pressure
THEN perform the following:

a. Check HP nitrogen supply pressure at Aux Control Board or at 115' nitrogen bottle racks. [ ]

b. Verify CLOSED all accumulator nitrogen fill and vent valves: [ ]

- Accumulator 1: SI-1-8875A
- Accumulator 2: SI-1-8875B
- Accumulator 3: SI-1-8875C
- Accumulator 4: SI-1-8875D

c. CLOSE SI-1-8880, Accumulator Nitrogen Fill Header Isolation Valve. [ ]

> **CAUTION**: Opening HCV-943 fully reduces pressure very quickly and can result in pressure being less than the Technical Specification minimum.

d. Cycle SI-1-HCV-943, Accumulator Nitrogen Vent Valve, as necessary to lower pressure downstream of the nitrogen regulator in small slow increments. [ ]

CAUTION:	Do not open more than one accumulator nitrogen fill and vent valve (8875A-D) valve at the same time. This cross-connects the accumulator nitrogen supplies and may result in a loss of design basis injection capability.[T35830]

e. Perform the next two steps in rapid succession:

1. OPEN SI-1-8880, Accumulator Nitrogen Fill Header Isolation Valve. [ ]

2. OPEN the nitrogen fill and vent valve on the affected accumulator: [ ]
   - Accumulator 1: SI-1-8875A
   - Accumulator 2: SI-1-8875B
   - Accumulator 3: SI-1-8875C
   - Accumulator 4: SI-1-8875D

f. WHEN pressure returns to normal,
THEN CLOSE appropriate accumulator nitrogen fill and vent valve. [ ]

2.3.4 IF accumulator pressure continues to drop,
THEN dispatch an Operator to check the following for leakage: [ ]
- Safety valve
- Manual vent
- Blank flange

2.3.5 Probable Causes
- Accumulator and/or associated gas piping/valves leakage
- Accumulator level decrease due to temperature changes
- Inadvertent opening of accumulator vent valve
- Accumulator discharge
- Sampling of accumulator volume

3. AUTOMATIC ACTIONS

NONE

4. REFERENCES

4.1 AR PK02-10, "Accumulator Level Hi-Lo"

4.2 OP B-3B:I, "Accumulators - Fill and Pressurize"

4.3 501122, "Schematic Diagram – Main Annunciator" (Electrical Drawing Section 8)

**ACCUM
LEVEL
HI-LO**

07/20/05
Effective Date

QUALITY RELATED

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
352	Accum 1-1 Lvl Hi LC-950A	LC950A	> 72.6%	2.1
353	Accum 1-2 Lvl Hi LC-952A	LC952A	> 72.6%	2.1
406	Accum 1-3 Lvl Hi LC-954A	LC954A	> 72.6%	2.1
407	Accum 1-4 Lvl Hi LC-956A	LC956A	> 72.6%	2.1
1382	Accum 1-1 Lvl Hi LC-951A	LC951A	> 72.6%	2.1
1383	Accum 1-1 Lvl Lo LC-951B	LC951B	< 60.8%	2.1
1384	Accum 1-2 Lvl Hi LC-953A	LC953A	> 72.6%	2.1
1385	Accum 1-2 Lvl Lo LC-953B	LC953B	< 60.8%	2.1
1390	Accum 1-1 Lvl Lo LC-950B	LC950B	< 60.8%	2.1
1391	Accum 1-2 Lvl Lo LC-952B	LC952B	< 60.8%	2.1
1408	Accum 1-3 Lvl Hi LC-955A	LC955A	> 72.6%	2.1
1409	Accum 1-3 Lvl Lo LC-955B	LC955B	< 60.8%	2.1
1410	Accum 1-4 Lvl Hi LC-957A	LC957A	> 72.6%	2.1
1411	Accum 1-4 Lvl Lo LC-957B	LC957B	< 60.8%	2.1
1419	Accum 1-3 Lvl Lo LC-954B	LC954B	< 60.8%	2.1
1420	Accum 1-4 Lvl Lo LC-956B	LC956B	< 60.8%	2.1

2. OPERATOR ACTIONS

2.1 General Actions **(All Inputs)**

NOTE:	The high and low level alarm setpoints are more accurate than The Control Board instrument readings.

2.1.1 Check both level indicators on the affected accumulator to confirm the alarm and the high or low level condition. [ ]

2.1.2 REFER TO TS 3.5.1 to determine accumulator OPERABILITY. [ ]

2.1.3 Verify CLOSED the following:

a. NSS-1-9357A, Accumulator Sample Line Inside Containment [ ]

b. NSS-1-9357B, Accumulator Sample Line Outside Containment [ ]

c. Accumulator Drains To RCDT:
- SI-1-8876A [ ]
- SI-1-8876B [ ]
- SI-1-8876C [ ]
- SI-1-8876D [ ]

d. Accumulator Fill Lines From SI Pump:
- SI-1-8878A [ ]
- SI-1-8878B [ ]
- SI-1-8878C [ ]
- SI-1-8878D [ ]

2.1.4 Dispatch an Operator to verify CLOSED NSS-1-9352A, B, C, and D, Accumulator Sample Line Isolations (Aux Bldg, 100 ft Primary Sample Panel). [ ]

2.1.5 IF accumulator level is high,
THEN return level to normal as follows:

a. OPEN accumulator drain to RCDT valve on affected accumulator until accumulator level reaches approximately 67%: [ ]
   - Accumulator 1: SI-1-8876A
   - Accumulator 2: SI-1-8876B
   - Accumulator 3: SI-1-8876C
   - Accumulator 4: SI-1-8876D

b. Direct chemistry to sample accumulator. [ ]

c. Determine if either of the following procedures should be performed:
   - STP V-5A1, "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8819 A-D and 8956 A-D" [ ]
   - STP V-5A2. "Emergency Core Cooling System Check Valve Leak Test, Post-Refueling/Post-Maintenance Valves 8948 A-D and 8818 A-D." [ ]

2.1.6 IF accumulator level is low,
THEN raise affected accumulator level PER OP B-3B:I, "Accumulators - Fill and Pressurize". [ ]

2.1.7 IF PK02-05, "ACCUM PRESSURE HI-LO", alarm is ON,
THEN GO TO AR PK02-05, "Accum Pressure Hi-Lo". [ ]

2.1.8 Probable Causes

a. Low level
   - One or more open/leaking accumulator drains
   - One or more open/leaking sample valves
   - One or more open/leaking fill line valves
   - Accumulator system piping leakage
   - Accumulator discharge to Primary System

b. High level
   - Discharge check valve leakage
   - Fill line valve leakage

SRO Question 88

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2__
	Group:	_____	_1___
	K/A:	_____	061 G2.2.25
	Importance Rating:		3.7

Proposed Question:

According to the Bases in Technical Specifications for AFW, what is the MINIMUM amount of feedwater flow available to remove decay heat following an assumed feedline break accident?

A. Two motor driven AFW pumps or the TDAFW pump able to deliver flow to a total of 3 steam generators.

B. One motor driven AFW pumps and the TDAFW pump able to deliver flow to a total of 3 steam generators.

C. One motor driven and TDAFW pump each able to deliver flow to two steam generators.

D. One motor driven AFW pump able to deliver flow to two steam generators.

Proposed Answer:

D. One motor driven AFW pump able to deliver flow to two steam generators

Explanation:

A incorrect. Only one MDAFW pump to 2 SGs assumed.

B incorrect. Flow assumed to only 2 SGs.

C incorrect. TDAFW pump not assumed.

D correct. According to TS Bases 3.7.5 - The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and *a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the broken MFW header at the pump maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.*

Technical Reference(s): B3.7.5

Proposed references to be provided to applicants during examination: None

Learning Objective: 9694G - Discuss 3.7 Technical Specification bases

Question Source:
Bank – DCPP B-0401

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments:

K/A: 061 G2.2.25 - AFW, Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (3.7)

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take normal suction through valve MU-671 on the single suction line from the condensate storage tank (CST) (LCO 3.7.6) (this valve must remain open for the applicable accident analysis assumptions to be valid) and are capable of being aligned to the firewater storage tank (FWST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the condenser steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity, and the turbine driven pump provides 200% of the required capacity to the steam generators, with 100% capacity defined as the flow required to two steam generators during the AFW design basis accident analysis (loss of normal feedwater flow (Ref. 1). The pumps are equipped with recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators, although each pump has the capability to be manually realigned to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

(continued)

BASES

BACKGROUND (continued)

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with vital AC powered control valves. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The AFW System (both the one turbine-driven and two motor-driven AFW pumps) actuates automatically upon actuation of the anticipated transient without scram mitigating system actuation circuitry (AMSAC). The motor-driven pumps are additionally actuated by: (1)safety injection; (2) an associated bus transfer to the diesel generator signal; (3) a trip of both MFW pumps; or (4) steam generator water level—low-low in one of four SGs. The turbine-driven pump is additionally actuated by 12 kV bus undervoltage or steam generator low-low level in two of four SGs via ESFAS (LCO 3.3.2).

The AFW System is discussed in the FSAR, Section 6.5 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to at least two steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3% tolerance plus 3% accumulation within 1 minute after event initiation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and AFW spillage through feedwater line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater (FWLB) or Main Steam Line Break (MSLB); and

b. Loss of MFW (the coincident loss of offsite power is a less limiting transient since RCP heat input is lost).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	In addition, the minimum available AFW flow and system characteristics must be considered in the analysis of normal cooldown and small break loss of coolant accident (LOCA) due to their potential impact. The AFW System is also designed for decay heat removal following a Steam Generator Tube Rupture (SGTR). As such the steam turbine driven AFW pump has redundant steam supplies to assure continued availability following a SGTR or MSLB event. <u>The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the broken MFW header at the pump maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.</u> The ESFAS automatically actuates the AFW turbine driven pump when required to ensure an adequate feedwater supply to the steam generators during loss of power. Vital AC power operated valves are provided for each AFW line to control the AFW flow to each steam generator. The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of decay and residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses and having the third AFW pump powered by a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs. To assure steam turbine driven AFW pump operability via redundant steam supplies, steam traps 104, 105 and 106 on the supply lines must be operable or bypassed to ensure adequate condensate removal and check valves MS-5166 and MS-5167 must be operable. The AFW System supply is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps,

(continued)

1.00

What is the MINIMUM acceptable auxiliary feedwater alignment to ensure that adequate feedwater flow is available to remove decay heat?

A. One motor driven AFW pump delivering flow to two S/Gs

B. One motor driven AFW pump delivering flow to one S/G

C. Two motor driven AFW pumps delivering flow to two S/Gs

D. One motor driven and the steam driven AFW pump delivering flow to 2 S/Gs

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

9694	Discuss Technical Specification bases
9694G	Discuss 3.7 Technical Specification bases

Reference Id: B-0401
Must appear: No
Status: Active
User Text: 9694.13ALLN
User Number 1: 0000002.70

SRO Question 89

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2___
	Group:	_____	_1___
	K/A:		064 A2.19
	Importance Rating:		2.7

Proposed Question:

GIVEN:
- Unit 1 is at 50% power
- Diesel Generator 1-2 is fully loaded and running in accordance with STP-M9A, Diesel Engine Generator Routine Surveillance Test.
- Load is 2.5 MW and 1.2 MVARs out

The crew is preparing to start Circulating Water pump 1-1.

Which of the following actions should be taken by the Shift Foreman?

A. Direct the BOPCO to reduce MVARs, then allow the pump to be started.

B. Direct the BOPCO to reduce MW load, then allow the pump to be started.

C. Allow the pump to be started, there should be no effect on diesel MW loading or MVARs.

D. Direct the BOPCO to closely monitor MVARs and adjust as necessary while the pump is started.

Proposed Answer:

A. Direct the BOPCO to reduce MVARs, then allow the pump to be started.

Explanation:

A correct. Paralleled to the grid, starting a large load can affect the VAR loading of the diesel. The procedure recommends reducing MVAR loading to less than 0.5 MVAR prior to starting large 12 KV or 4KV loads.

B incorrect. Real load should not be affected by the start of the pump.

C incorrect. While not powered from the diesel, the reactive load of the pump will affect the diesel because it is currently paralleled with aux power.

D incorrect. Action is to reduce VARs prior to starting a large load to prevent exceeding limits.

Technical Reference(s): M-9A, Diesel Engine Generator Routine Surveillance Test

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: 064 A2.19 - Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of high VARS on ED/G integrity (2.7)

INITIALS

10.7 There should be fuel oil in the priming tank. If there is not, the priming tank should be filled using the magnetic pump. Document problem in an AR. _______

10.8 The fuel oil pressure should increase to above 40 PSIG within 60 seconds of engine start. Gauge response is about 15 seconds. _______

10.9 Do not violate the following limits during normal steady-state operation:

10.9.1 Maximum continuous generator current is 451 amperes. _______

10.9.2 Maximum stator temperature is 240°F. _______

10.9.3 Minimum lube oil pressure is 60 PSIG. _______

10.9.4 Maximum lube oil temperature is 195°F. _______

10.9.5 Power factor: 1.0 to 0.8 lag (see Appendix 8.1). _______

10.9.6 Load: 2.60 MW at 0.8 PF (see Appendix 8.2 for maximum limits). _______

NOTE: The D/G may operate at > 2.5 and ≤ 2.75 MW at 0.8 PF for up to 2000 hours per year.

10.10 Whenever a D/G is paralleled to the grid and operating at full load, the MVAR output should be reduced prior to starting large loads. Diesel generators are highly responsive to starting large motor loads and could potentially exceed their KVA rating. The following good operating practices apply: (AR A0602120)

10.10.1 If a large motor load will be started while the D/G is paralleled and at full load, reduce to 0.5 MVARs or less. _______

10.10.2 These loads include 12KV pumps, condensate / booster pump set, or the #2 heater drain pump. _______

10.10.3 After the load has been started, MVARs may be raised. _______

10.11 Normal shutdown of a D/G requires DC control power. If it becomes necessary to shutdown the D/G without control power, manually operate the trip lever on the northwest corner of the engine, forward of the fuel oil filters. _______

10.12 Do not operate more than one D/G at a time paralleled to any transformer (startup or unit auxiliary) in MODE 1, 2, 3, or 4 (SR 3.8.1.3 Note 3). _______

10.13 The applicable D/G MODE SEL switch on VB4 shall be in MANUAL prior to paralleling the D/G to the bus. _______

SRO Question 90

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2___
	Group:	_____	_1___
	K/A:	_____	103 G2.1.33
	Importance Rating:		4.0

Proposed Question:

The plant is operating at 100% power.

GIVEN:

- Narrow range containment pressure recorder (YR 26) and PPC Point (P0706A) are reading 0.9 psig
- Containment temperature:
  - TE – 85, 120°F
  - TE – 87, 122°F
  - TE – 89, 117°F
  - TE – 91, 118°F

What Technical Specification LCO, if any, has been exceeded?

A. None. Containment parameters are within Technical Specifications.

B. Only Technical Specification 3.6.4, Containment Pressure.

C. Only Technical Specification 3.6.5, Containment Temperature.

D. Technical Specifications 3.6.4, Containment Pressure and 3.6.5, Containment Temperature.

Proposed Answer:

A. None. Containment parameters are within Technical Specifications.

Explanation:

A is correct. Pressure limit (upper) is 1.2 psig. Temperature limit is 120F, average. Current temperature average is just over 119F.

B incorrect. Limit is 1.2 psig (note lower limit is *-1.0* psig)

C incorrect. Temperature limit is an *average* of the temperature readings.

D incorrect. Both are within specs.

Technical Reference(s): Tech Spec 3.6.4 and 3.6.5

Proposed references to be provided to applicants during examination: None

Learning Objective: 9697F - Identify 3.6 Technical Specification LCO

Question Source:
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments:

K/A: 103 G2.1.33 - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (4.0)

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. AND B.2 Be in MODE 5.	6 hours 36 hours

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.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SRO Question 91

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2___
	Group:	_____	_2___
	K/A:		017 A2.02
	Importance Rating:		4.1

Proposed Question:

GIVEN:
- The crew has entered FR-C.1, Response to Inadequate Core Cooling from step 4 of E-1, Loss of Reactor or Secondary Coolant.
- All RCPs are secured.
- RCS pressure is 1800 psig.
- Charging pump flow of 150 gpm has been established.
- Core Exit Thermocouples are 1125°F
- RVLIS full range level is 29% and increasing slowly

Which of the following actions should be taken by the Shift Foreman?

A. Return to step 4 of E-1.

B. Return to step 2 of FR-C.1.

C. Verify ECCS valve alignment and go to step 8 of FR-C.1

D. Go to step 9 of FR-C.1.

Proposed Answer:

B. Return to step 2 of FR-C.1.

Explanation:

A incorrect. RVLIS level must be increased. Cannot transition from C.1 at this time.

B correct. With increasing RVLIS, a loop back through the first steps of C.1 is performed.

C incorrect. This is required if RVLIS level is lowering.

D incorrect. This is required if thermocouples are greater than 1200°F.

Technical Reference(s): FR-C.1, step 6 and 7

.

Proposed references to be provided to applicants during examination: FR-C.1, step 1 through 7.

Learning Objective: 9703 - Identify exit conditions for the FRPs

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: 017 A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Core damage (017)

TITLE: Response to Inadequate Core Cooling

4/18/06
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1.0 SCOPE

1.1 This procedure provides actions to restore core cooling.

1.2 The actions in this procedure are considered time critical and therefore an abbreviated tailboard should be conducted.

1.3 The major actions in EOP FR-C.1 are:

- Establish ECCS flow to the RCS,
- Rapidly depressurize the S/Gs to depressurize the RCS,
- Start the RCPs and open all RCS vent paths to containment.

2.0 VERIFY ENTRY CONDITION FOR EOP FR-C.1

2.1 EOP F-0.2, RED Condition

The following criteria should be used to determine if five T/Cs have exceeded 700° F or 1200°F.

All T/Cs listed on PAMS 3/4 or SPDS may be used to determine an Inadequate Core Cooling Condition.

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**

****

CAUTION: RHR Pps should NOT be run on recirculation GREATER THAN 30 MINUTES WITHOUT CCW to the RHR Heat Exchangers.

****

1. **RESET SI**

2. **CHECK ECCS Valve Alignment - PROPER EMERGENCY ALIGNMENT:**

   - Activate the monitor lights for monitor light Box C by turning the Monitor Test Light Switch to ON

     Use White Status lights to verify ECCS valve alignment

     OR

   - Use Control Board mimic bus to verify ECCS valve alignment

RESPONSE NOT OBTAINED (Step 2): Manually or locally align valves with White Status Lights - ON.

- -

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
3. VERIFY ECCS Flow In All Trains:	
a. Charging Injection flow; FI-917 - GREATER THAN 100 GPM	a. Perform the following: 1) Verify both CCPs Running. 2) IF CCPs Running at LESS THAN minimum recirc flow 48 AMPS, THEN Verify ECCS valve alignment. (a) Check monitor light Box A and C. (b) Align valves with White Status Lights - ON. (c) Verify manual flowpath valves aligned. REFER TO OP K-10G1, SEALED COMPONENT CHECKLIST FOR ECCS MODES 1, 2, & 3. (N/A for valves inside containment.) IF Charging Injection FLOW GREATER THAN 100 GPM, THEN GO TO Step 3b (Next Page).

THIS STEP CONTINUED ON NEXT PAGE

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
3. VERIFY ECCS Flow In All Trains: (Continued)	
	(d) Establish alternate ECCS injection path by performing ONE of the following: 1. Recip Chg Pp a. Verify lineup • 8805A and B - OPEN • FCV-128 - OPEN • 8803A and B - OPEN • 8801A and B - OPEN b. Start Recip Chg Pp 2. Normal Charging a. 8805A and B - OPEN b. FCV-128 - OPEN c. HCV-142 - OPEN d. 8107 and 8108 - OPEN e. 8146 - OPEN f. Verify CCP - Running
b. RCS WR Pressure - LESS THAN 1650 PSIG	b. GO TO Step 4 (Page 6).

THIS STEP CONTINUED ON NEXT PAGE

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
3. VERIFY ECCS Flow In All Trains: (Continued)	
c. SI Pp Flow FI-918/922 - GREATER THAN 100 GPM	c. Perform the following: 1) Verify both SI Pps Running. 2) IF SI Pp Running LESS THAN minimum recirc flow 28 AMPS, THEN Check ECCS valve alignment. (a) Check Monitor Light Box A and C. (b) Align valves with White Status Lights - ON. (c) Verify manual flowpath valves aligned. REFER TO OP K-10G1, SEALED COMPONENT CHECKLIST FOR ECCS MODES 1, 2, & 3. (N/A for valves inside containment.) -
d. RCS WR Pressure - LESS THAN 300 PSIG	d. GO TO Step 4 (Next Page). -

THIS STEP CONTINUED ON NEXT PAGE

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. VERIFY ECCS Flow In All Trains: (Continued)	
e. RHR Pp Flow FI-970A/971A - GREATER THAN 100 GPM	e. Perform the following: 1) Verify both RHR Pps Running. 2) IF RHR Pps Running at LESS THAN minimum recirc flow 28 AMPS, THEN Check ECCS valve alignment. (a) Check Monitor Light Box A and C. (b) Align valves with White Status Lights - ON. (c) Verify manual flow path valves aligned. REFER TO OP K-10G1, SEALED COMPONENT CHECKLIST FOR ECCS MODES 1, 2, & 3. (N/A for valves inside containment.) -

****

CAUTION: A Very High Radiation Area will occur if LTDN or RCP Seal Return is initiated with core damage suspected or imminent.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. CHECK RCP Support Conditions - AVAILABLE: • REFER TO Appendix B • Continue with this instruction	Try to establish support conditions for running an RCP. REFER TO Appendix B. Continue with procedure while doing this step. -

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
5. CHECK Accumulator Isolation Valve Status:	
a. Restore power to Accum Disch Isol Vlvs 8808A 52-1F-46 52-1F-46R 8808B 52-1G-07 52-1G-07R 8808C 52-1H-14 52-1H-14R 8808D 52-1G-05 52-1G-05R	
b. Check Accum Disch Isol Valves - OPEN	b. Open Accum Disch Isol Valves unless Closed after Accumulator Discharge.
6. CHECK Core Exit T/Cs - LESS THAN 1200°F (PAMS 3/4)	GO TO Step 9 (Page 9).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. CHECK RVLIS Level Indication: (PAMS 1/2)	
a. Check all RCPs shutdown.	a. RETURN TO procedure and step in effect
b. Full Range GREATER THAN 32%	b. IF RVLIS Level Increasing, THEN RETURN TO Step 2 (Page 2). IF RVLIS Level NOT increasing, THEN • Verify ECCS valve alignment. REFER TO OP K-10G1, SEALED COMPONENT CHECKLIST FOR ECCS MODES 1, 2 & 3. (N/A for valves inside containment). AND • GO TO Step 8.
c. RETURN TO procedure and step in effect	
8. CHECK Core Exit T/Cs: (PAMS 3/4)	
a. Temperature - LESS THAN 700°F	a. IF Core Exit T/Cs Decreasing, THEN RETURN TO Step 2 (Page 2). IF NOT Decreasing, THEN GO TO Step 9 (Next Page).
b. RETURN TO procedure and step in effect	

SRO Question 92

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2__
	Group:	_____	_2___
	K/A:	_____	034 A1.02
	Importance Rating:		3.7

Proposed Question:

Refueling is in progress one week after the reactor was shutdown.

A spent fuel assembly is dropped while the operator is removing the assembly from the core.

For this accident, which of the following was verified prior to fuel movement to ensure that the 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 limits?

A. One train of Containment Ventilation Isolation instrumentation OPERABLE.

B. Two trains of Fuel Handling Building Ventilation System OPERABLE.

C. Refueling Cavity Concentration greater than limit in the COLR.

D. Refueling Cavity Level greater than 23 feet.

Proposed Answer:

D. Refueling Cavity Level greater than 23 feet.

Explanation:

A incorrect. CVI instrumentation is required during movement of "recently irradiated fuel" or fuel that has been in a critical reactor less than 100 hours ago.

B incorrect. Only applies during movement of recently irradiated fuel.

C incorrect. Ensures subcriticality.

D correct. per TS bases 3.9.7 - Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits

Technical Reference(s): TS bases 3.9.7, OP B-8

Proposed references to be provided to applicants during examination: None

Learning Objective: 6497 - State the responsibilities and duties of Refueling SRO

Question Source:
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
55.43.7 - Fuel handling facilities and procedures.

Comments:

K/A: 034 A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal (3.7)

NOTE: From ES-401 - A number of the generic K/As in Section 2 of the catalogs are specifically linked to one or more topics specified in 10 CFR 55.43(b), and all of the Category A2, AA2, and EA2 K/A statements are (or, in the case of NUREG-1123, should be) similarly linked. Consequently, the K/As for the SRO examination will be drawn from those K/A categories (denoted by Columns “A2” and “G” in the SRO-only section of the applicable examination outline) ***and from all K/A categories related to the fuel handling facilities, which are specifically identified for sampling in 10 CFR 55.43(b)(7).***

BACKGROUND	The FHBVS filters airborne radioactive particulates and radioactive iodine from the area of the fuel pool following a fuel handling accident. The FHBVS provides environmental control of temperature and humidity in the fuel pool area and for the AFW pump motors. The ventilation for the AFW pump motors is to provide cooling flow for EQ considerations, i.e., motor longevity. The ventilation is not required to function during an accident or for the few hours required to reach RHR conditions during a natural circulation cooldown. The FHBVS consists of two independent and redundant trains. Each train consists of, an exhaust prefilter, high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and an exhaust fan. A third non-vital exhaust fan is used for normal operation and has only a prefilter and a HEPA filter. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal or loss of the normal exhaust fan E-4. The FHBVS is a standby system, parts of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharge from the fuel handling building is isolated and the normal exhaust fan shuts down and the vital exhaust fans start and the stream of ventilation air discharges through the system filter trains. The prefilter removes any large particles in the air, to prevent excessive loading of the HEPA filter and charcoal adsorber. The FHBVS is discussed in the FSAR, Sections 9.4.4 and 15.5 (Refs. 1, and 2, respectively) because it may be used for normal, as well as post (fuel handling) accident, atmospheric cleanup functions.
APPLICABLE SAFETY ANALYSES	**<u>The FHBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident involving the handling of recently irradiated fuel. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged.</u>** The DBA analysis of the fuel handling accident assumes that only one train of the FHBVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident. Due to radioactive decay, the

(continued)

BASES

BACKGROUND

The limit on the boron concentrations of the filled portions of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity, that have direct access to the reactor vessel, during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. The refueling boron concentration is sufficient to maintain shutdown margin (SDM) with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in MODE 6 is sufficient to maintain $k_{eff} \leq 0.95$ with the most reactive rod control assembly completely removed from its fuel assembly.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the principle system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with refueling grade borated water from the liquid hold up tanks or the refueling water storage tank.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation cooling in the RCS and assist in maintaining the boron concentration uniformity in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

(continued)

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.
APPLICABLE SAFETY ANALYSIS	During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within allowable limits (Refs. 4, and 5). Refueling cavity water level satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.

(continued)

Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Not used				
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
	(a)	1 train	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Purge Radiation Gaseous and Particulate	1, 2, 3, 4	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	Per ODCM
	(a)	1	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	Per ODCM
4. Containment Isolation-SI	Refer to LCO 3.3.2, “ESFAS Instrumentation,” Functions 1 and 3, for all initiation functions and requirements.			

(a) During movement of recently irradiated fuel assemblies within containment.

OP B-8D attachment 9.1

1. Temporary nozzle dams removed, or use approved by PSRC. N/A [ ] 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) MECH ______/______ ______/______

   (CONT) MECH ______/______ ______/______

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, or the filter assembly flange isol valve is CAUTION tagged closed.

   e. RCS-8032 Leak Detection Return to RCDT.

5. Portable radiation monitor(s) are available for use on the manipulator crane and spent fuel pool bridge crane, and a continuous air monitor is located in Containment (140' el.) and the Fuel Handling Building. RP ______/______ ______/______

ITEM NO.	DESCRIPTION	PRIOR TO CORE ALT. COMPLETE WITHIN	RESPONSIBLE DEPARTMENT INIT./DATE	REVIEWED (SFM) INIT. HOUR/DATE
6.	Verify all OP L-0 items for Mode 6/ Core Alterations are complete.	2 hours	OP ___/___	____ ____/____
7.	ECG 42.5, Verify <u>refueling cavity</u> water level >23 feet above irradiated fuel assemblies within the reactor. (elev. 126 ft. 6 in.)	2 hours	OP ___/___	____ ____/____
8.	ECG 42.2, Establish direct communications between the control room and the refueling stations.	1 hour	OP ___/___	____ ____/____
9.	Test containment evacuation alarm.	1 hour	OP ___/___	____ ____/____
10.	SRO for fuel handling operations only <u>and</u> licensed RO required for continuous monitoring of count rate data during CORE ALTERATIONS.		OP ___/___	____ ____/____

RCCA Unlatching Commenced ______/______ ______
TIME DATE SFM

RCCA Unlatching Completed ______/______ ______
TIME DATE SFM

ITEM NO.	DESCRIPTION	RESPONSIBLE DEPT INITIALS/DATE	REVIEWED (SFM) INITIALS/DATE
	NOTE: Prior to core unloading, verify Items 1 through 5 and 11 through 25 are current.		
11.	Fuel accountability tracking software for control of fuel assembly and insert locations during refueling operations are available for use in the appropriate areas (See TS6.ID2).	RX ENG ______/______	______/______
12.	If handling recently irradiated fuel per TS 3.7.13 ensure FHB Ventilation is capable of supporting fuel movement by verifying the following are satisfied: N/A [ ]	OP ______/______	______/______
	a. Complete OP H-7:I, Attachment 9.2.		
	b. Active SFM clearance exists on the 140' partition wall locks and 115' roll-up door chains. CR NUMBER ____________________		
	c. The Refueling SRO or designee has possession of the keys to the roll-up doors to prevent them from being opened.		
13.	Verify an active SFM Admin Clearance exists on the source range speaker in containment.	OP ______/______	______/______
14.	All critical personnel participating in the core unloading have been adequately trained for their part in the fuel handling operations including nuclear engineering/operations fuel handlers.	RX ENG ______/______ OP ______/______	______/______ ______/______
15.	Core unloading sequence completed and issued by reactor engineering to determine: • 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.	RX ENG ______/______	______/______
16.	Verify that RCP D-220 Controls are in place (High Rad Areas locked for fuel transfer). Setting the controls too early may hinder maintenance activities.	RP ______/______	______/______
17.	Perform a spot check of the Manipulator Crane indexing after upper internals removal. Check a minimum of two widely separated core locations.	OP ______/______	______/______
18.	A test run of the fuel transfer system and manipulator crane should be made with the dummy assembly if possible.	OP ______/______	______/______

ITEM NO.	DESCRIPTION	PRIOR TO UNLOADING COMPLETE WITHIN	RESPONSIBLE DEPARTMENT INIT./DATE	REVIEWED (SFM) INIT. HOUR/DATE	RETEST FREQUENCY	NEXT DUE DATE	(SFM) RETEST COMPLETED INIT. HOUR/DATE
19.	Procedure MP I-4.4-2A, Spent Fuel Pool crane load cell calibration.	6 months	ELE ___/___	____ ____/____	6 mo.	____	____ ____/____
20.	LT 13-2, Spent Fuel Pool Temperature Channel TIC-651 Calibration.	31 days	I&C ___/___	____ ____/____	92 days	____	____ ____/____
21.	STP M-27, Fuel handling interlocks.	7 days	OP ___/___	____ ____/____	N/A		
22.	Verify acceptability of the SFP cooling system status per Attachment 9.4 of OP B-8DS1.	12 hours	OP ___/___	____ ____/____	12 hours		See OP B-8DS1
23.	TS 3.9.7, Verify <u>refueling cavity</u> water level >23 feet over the reactor vessel flange. (elev. 138 ft. 2 in.)	2 hours	OP ___/___	____ ____/____			
24.	ECG 42.2, Establish direct communications between the control room and the refueling stations.						1 hour OP ___/___
	a. Test containment evacuation alarm.						C. O. Logs
25.	SRO for fuel handling operations only <u>and</u> licensed operator required for continuous monitoring of count rate data during CORE ALTERATIONS. MODE 6	OP ___/___	____ ____/____	ALL TIME			

Core Unloading Commenced ______/______ ______ Core Unloading Completed ______/______ ___

SRO Question 93

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_2__
	Group:	_____	_2___
	K/A:	_____	075 A2.02
	Importance Rating:		2.7

Proposed Question:

GIVEN:
- The plant is at 40% power.
- Kelp loading causes condenser vacuum to slowly degrade
- While performing OP AP-7, Degraded Condenser, both Circulating Water pumps trip

Which of the following actions should be taken by the Shift Foreman?

A. Go to AP-29, Main Turbine Malfunction.

B. Direct the operator to trip the reactor and go to E-0, Reactor Trip or Safety Injection while AP-7 is implemented.

C. Perform the actions of AP-7 while AP-29, Main Turbine Malfunction, is implemented.

D. Direct the operator to trip the reactor and perform the actions of AP-7 while E-0, Reactor Trip or Safety Injection is implemented.

Proposed Answer:

B. Direct the operator to trip the reactor and go to E-0, Reactor Trip or Safety Injection while AP-7 is implemented.

Explanation:

A incorrect. This is the action if the turbine is tripped below P-9 due to degrading vacuum while there is still circ water in operation.

B correct. The SFM goes to E-0 while other actions are implemented.

C incorrect. AP-29 is not addressed.

D incorrect. The SFM action is to go to E-0, unlike other AP's which has the SFM remain in the AP while E-0 is implemented.

Technical Reference(s): OP AP-7, Degraded Condenser, rev 34.

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: 075 A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of Circulating Water pumps (2.7)

SECTION A: LOSS OF CONDENSER VACUUM (Continued)

ACTION / EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**

****

CAUTION 1: If both CWPs are lost, Attachment 6.1 should be implemented after EOP E-0 is performed.

CAUTION 2: If it appears that continuation of deliberate operator actions to stabilize the plant ON LINE will not be successful, manually trip the reactor and go to EOP E-0.

****

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. CHECK Condenser • Condenser Pressure LESS THAN maximum allowed per Attachment 6.2	INITIATE a Turbine Trip, a. If Rx Trips, GO TO EOP E-0. b. If < P-9, Turbine Trip without Rx Trip: 1) IMPLEMENT the remainder of this procedure. 2) GO TO OP AP-29.
2. STABILIZE Condenser Pressure: a. REDUCE Unit load as necessary to maintain Condenser pressure within the limitations of Attachment 6.2	a. IF Operating limitations are exceeded, THEN INITIATE a Turbine Trip, 1) If Rx Trips, GO TO EOP E-0. 2) If < P-9, Turbine Trip without Rx Trip: a) IMPLEMENT the remainder of this procedure. b) GO TO OP AP-29.
3. MAKE PA announcement for loss of condenser vacuum	
4. Determine MFW Pp Parameters • Condenser Pressure LESS THAN 10" Hg ABS	a. Trip the MFW Pps. b. GO TO OP AP-15, section B.

SECTION A: LOSS OF CONDENSER VACUUM (Continued)

ACTION / EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
5. CHECK Condenser Parameters - NORMAL:	
a. CHECK SJAE After Condenser Vent Flow (FR-81) - NORMAL	a. Perform the following: • Investigate cause of air inleakage. • Valve in additional Air Ejectors per OP C-6:I. • Secure nitrogen injection to the condenser
Check condenser pressure – STABLE OR DECREASING	b. Place the Vacuum Pump on the affected Unit per OP C-6:I, Attachment 9.1
6. CHECK Circulating Water System Operation - NORMAL:	
a. VERIFY at least 1 CWP running	a. TRIP the Reactor AND GO TO EOP E-0 AND IMPLEMENT the following: 1) Close the MSIVs. 2) Adjust 10% Steam Dump controllers as needed to maintain S/G pressure LESS THAN OR EQUAL TO 1005 PSIG (8.38 turns). 3) Perform Attachment 6.1, Hot Condenser Cooldown Instructions.
b. VERIFY CWP intake gates, discharge gates and CWP discharge valves are OPEN as required	

THIS STEP CONTINUED ON NEXT PAGE

SRO QUESTION 94

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	__3__
	Group:	_____	__1__
	K/A:		G2.1.20
	Imp. Rating:	_____	__4.2_

Proposed Question:

Given the following conditions:

- Unit 1 is at full power when Safety Injection actuates.
- The crew is performing E-0, Reactor Trip Or Safety Injection.
- The BOPCO has been directed to perform Appendix E, ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS.

The SFM determines that a transition to E-1, Loss Of Reactor Or Secondary Coolant, is necessary. The BOPCO reports only the first 6 steps of Appendix E are complete.

Which of the following actions is required?

A. Wait for the completion of all actions in Appendix E prior to transitioning to E-1.

B. Wait for the completion of actions through step 8 in Appendix E prior to transitioning to E-1.

C. Transition to E-1 and direct the BOPCO to continue Appendix E actions.

D. Transition to E-1 and direct the BOPCO to discontinue the actions in Appendix E.

Proposed Answer:

C. Transition to E-1 and direct the BOPCO to continue Appendix E actions.

Explanation:

A incorrect. The transition to E-1 should be made when the SFM determines it is required.

B incorrect. No wait is required.

C is correct. From LPE0: • If App E is not completed by the transition out of E-0, the SFM shall be notified of progress and the SFM shall determine how to proceed

• In most cases all of App E is expected to be done prior to transition out of E-0, but ESF verifications through step 8 (as a minimum) should be done (not set aside for other tasks) at that point

D incorrect. Appendix E should be completed.

• *The expectation is to complete App E, even if some steps remain after transition from E-0 (SFM call if an extreme resource problem arises)*

Technical Reference(s): Lesson LPE0.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3705 - Explain the responsibilities of the control room staff positions during emergencies

Question Source:
Bank

Question History: Last NRC Exam Beaver Valley 5/2005

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

10 CFR Part 55 Content:
55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: G2.1.20 – Ability to execute procedure steps.

Appendix E Actions, Continued

E-0 "Verify Auto Actions" Associated with App E (w/Basis)

Obj 10, 11

Review EOP E-0, Appendix E, emphasizing the following items.
- Action/Expected Response and Response Not Obtained basis
- Procedure transitions
- Other discussion items in the list below

Item	Basis and Discussion

Implement Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status	• App E is assigned in the A/ER column of E-0 just after completing the immediate action steps
	• App E is done concurrently with the body of E-0 so that timely diagnostics can be accomplished
	  • Also facilitates more timely SI termination in the event of a very small LOCA or spurious SI (which minimizes Pzr overfill and excessive Pzr PORV lifts)
	• If App E is not completed by the transition out of E-0, the SFM shall be notified of progress and the SFM shall determine how to proceed
	  • In most cases all of App E is expected to be done prior to transition out of E-0, but ESF verifications through step 8 (as a minimum) should be done (not set aside for other tasks) at that point
	  • The expectation is to complete App E, even if some steps remain after transition from E-0 (SFM call if an extreme resource problem arises)
	• Expectation is to notify the SFM prior to taking major actions in the RNO column (unless previously discussed).

Continued on next page

SRO QUESTION 95

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_3__
	Group:	_____	_2__
	K/A:	_____	G2.2.19
	Importance Rating:		3.1

Proposed Question:

Which of the following describes the type of work classified as “Emergent Work”?

A. Work added to the daily schedule after scope freeze.

B. Work assigned to the operations support team (OST).

C. Immediate maintenance to prevent or mitigate the release of radioactive material.

D. Work performed to restore a plant SSC with identified potential degradation that does not affect the SSC's design function or performance.

Proposed Answer:

A. Work added to the daily schedule after scope freeze.

Explanation:
A correct, per procedure definition on page 6. “Work added to the daily schedule after scope freeze. Emergent work does not include OST work.”

B incorrect, this is OST work.

C incorrect, this is Emergency Maintenance.

D incorrect, this is Elective Maintenance.

Technical Reference(s): AD7.DC8 – Work Control

Proposed references to be provided to applicants during examination: None

Learning Objective: 7945 - State the definition of emergent work

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.3 - Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments:

K/A: G2.2.19 – Knowledge of maintenance work order requirements. (3.1)

NOTE: knowledge of the work process is a function of the Work Control Lead (WCL), an SRO position.

Degraded Condition or Degraded

A condition of a system, structure, or component in which there has been a loss of quality or functional capability. Examples include:

- The physical or assembled condition of a system, structure, or component does not match its originally installed condition. This condition can include parts that are:
  - burned
  - cracked
  - corroded
  - damaged/broken
  - deformed (e.g., twisted, bulged)
  - missing or improperly installed
- The performance of a system, structure, or component does not meet established acceptance criteria or comply with design requirements.

Elective Maintenance(EM)

The classification used for work performed to restore a plant SSC with identified potential or actual degradation that is not significant and does not affect the SSC's design function or performance. Examples include the following:

- Minor leaks that are controlled and that do not justify immediate action to repair. For example:
  - Body to bonnet leakage that is collected by a drip bag and directed to a drain.
  - A pin hole leak in the air admission valve (SV) to a diesel generator.

- Minor SSC degradation identified by predictive, periodic, or planned preventive maintenance that must be corrected to maintain the long-term equipment reliability but that is not expected to result in failure prior to the SSC's next scheduled maintenance period. For example:
  - Vibration of a charging pump is found in the alert range during a monthly STP.
  - Grease is leaking from the valve operator of a condensate demin bypass valve.

- Other minor SSC deficiencies that do not affect plant operation, nuclear safety, plant reliability, or the ability of operators to properly respond to normal, off-normal, or accident transients or conditions. Examples include: a) damaged or broken local indication gauges that are only informational and are not required by operators to control systems for normal or emergency response; b) indications of internal valve leakage that do not affect system operation or the ability to provide maintenance isolation. For example:
  - A service cooling water pump local discharge pressure gauge has wide fluctuations.
  - One of two turbine building sump pumps will not run.
  - RCP seal injection backpressure control valve leaks by when shut.

Emergency Maintenance

Immediate maintenance to prevent or mitigate the release of radioactive material, hazards to personnel, or extensive equipment damage.

Emergent Work

Work added to the daily schedule after scope freeze. Emergent work does not include OST work. Refer to AD7.ID4 for additional requirements or information.

Maintenance

Activities performed to maintain or restore systems, structures, or components, including activities that implement design changes.

In addition to activities traditionally associated with maintaining and restoring equipment or implementing design changes, maintenance includes all supporting functions, such as planning, scheduling, procurement, testing, etc., required to perform these activities. Examples of maintenance activities include:

- Activities that do not permanently alter the design, performance requirements, operation or control of a system, structure, or component, such as:
  - Calibration
  - Housekeeping
  - Troubleshooting
  - Refurbishment
  - Identical replacements
  - Installing design changes
  - Postmodification testing
  - Postmaintenance testing
- Temporary modifications and configuration changes that directly relate to and are necessary to support the maintenance, such as:
  - Lifting leads
  - Jumpering terminals
  - Removing barriers
  - Erecting scaffolds
  - Placing ancillary structures
  - Installing temporary power supplies
  - Placing lead shielding on pipes and equipment
  - Using temporary blocks, bypasses, and supports

Minor Maintenance

Work that can be performed safely without detailed work instructions and without overall plant coordination or scheduling.

Modification

Any change to the plant's physical configuration or operation and maintenance practices that is implemented using the modification control process.

Other Maintenance (OM)

The classification used for work that does not directly correct material condition deficiencies on plant SSCs. Examples include:

- Support activities, such as, shielding and insulation activities associated with other work.
- Painting, housekeeping, and general plant upkeep.
- Proactive actions to enhance plant equipment performance (that is, plant betterment activities) that are not directly related to correcting existing deficiencies. This classification includes implementation of design changes that are not required to restore an existing degraded component and proactive replacement of obsolete equipment or replacements based on services life expectations.
- Preoutage work required to support planned or forced outage activities.
- Work required to rebuild components for return-to-stock and reuse.
- Nonplant equipment maintenance. This category includes items such as work on auxiliary equipment used in the power block, for example mobile cranes and M&TE.

OST Work

- Work assigned to the operations support team (OST).

SRO Question 96

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_3__
	Group:	_____	_2__
	K/A:	_____	G2.2.23
	Importance Rating:		3.8

Proposed Question:

Unit 1 is in MODE 5.

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

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

A. Tracking is not done.

B. Using an Info TS sheet.

C. Using an Active TS sheet.

D. Documenting the inoperability each night in the Shift Manager Logs.

Proposed Answer:

A. Tracking is not done.

Explanation:

Only A is correct. From OP1.DC17, step 2.1.2 - During refueling outages (Modes 5 or 6) or at other times as approved by the operations manager, tracking of Info TSs will not be done. Elimination of Info TSs reduces unnecessary work tracked by other methods such as Mode transition checklist and clarifies the important TSs for the plant status.

Technical Reference(s): OP1.DC17, step 2.1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 3503 - Discuss administration requirements of Tech Spec status sheets

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.2 - Facility operating limitations in the technical specifications and their bases.

Comments:

K/A: G2.2.23 - Ability to track limiting conditions for operations. (3.8)

TITLE: Control of Equipment Required by the Plant Technical Specifications or Other Designated Programs

10/06/05
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED
SPONSORING ORGANIZATION: OPERATIONS
REVIEW LEVEL: "A"

1. SCOPE

1.1 This procedure describes the methods for control and tracking of equipment inoperability as required by the Technical Specifications (TSs) or Equipment Control Guidelines (ECGs). In addition, risk significant equipment (designated per AD7.DC6, On-Line Maintenance Risk Management) unavailability shall also be tracked using this procedure.

1.2 Normally, Technical Specification Tracking sheets should only be used for the above items. However, other items (such as NPDES Permit required instrumentation) may be tracked if desired by the shift foreman (SFM).

2. DISCUSSION

2.1 When equipment required to comply with a TS or ECG LCO, including computer based equipment such as the PPC, EARS, SPDS and main annunciator system, is not able to meet the definition of operability, then tracking equipment status is normally performed using a TS sheet generated in PIMS.

2.1.1 TSs are considered either active or info. If the equipment inoperability causes entry into an action statement, then the TS sheet is considered active. Normally (Modes 1-4) both active and Info TS sheets are tracked for reference and status control purposes.

2.1.2 During refueling outages (Modes 5 or 6) or at other times as approved by the operations manager, tracking of Info TSs will not be done. Elimination of Info TSs reduces unnecessary work tracked by other methods such as Mode transition checklist and clarifies the important TSs for the plant status.

2.1.3 Operations provides a Tech Spec sheet usage guide which describes how to use the PIMS module.

2.2 All items which are risk significant as defined by AD7.DC6, "On-Line Maintenance Risk Management," are to be tracked per this procedure in Modes 1-4 if the equipment is risk significant for the current Mode.

2.3 Other items may also be tracked using a TS tracking sheet as desired by operations management.

SRO Question 97

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_3__
	Group:	_____	_2__
	K/A:	_____	G2.3.4
	Importance Rating:		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

Proposed references to be provided to applicants during examination: none

Learning Objective: 7954 - State the emergency dose limits

Question Source: Bank # NRC

Question History: Last NRC Exam DCPP 2005

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

10 CFR Part 55 Content:

55.43 43.4 - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments: K/A: G2.3.4 - Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Question Worksheet

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.3.4	
	Importance		3.1

<u>Proposed Question</u>:

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

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.

SRO Question 98

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_3__
	Group:	_____	_3__
	K/A:	_____	G2.3.6
	Importance Rating:		3.1

Proposed Question:

The crew is preparing to discharge the Laundry and Hot Shower tank to the outfall.

What Chemistry and SFM signatures are required for the discharge to proceed?

A. The SFM must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.

B. The SFM must sign the Discharge Checklist. Only Chemistry must sign the Discharge Permit.

C. Chemistry must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.

D. Both the SFM and Chemistry must sign the Discharge Checklist and the Discharge Permit.

Proposed Answer:

A. The SFM must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.

Explanation:
A correct. The SFM must sign the Discharge Checklist. Both the SFM and Chemistry must sign the Discharge Permit.

B incorrect. Both sign the Discharge Permit.

C incorrect. The SFM signs the Discharge Checklist.

D incorrect. Only the SFM must sign the Discharge Checklist.

Technical Reference(s): OP G-1:II, Discharge of Liquid Radwaste, rev. 32, page 1 and attachment 9.1.

Proposed references to be provided to applicants during examination: None

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

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.43.4 - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

K/A: 2.3.6 – Knowledge of the requirements for reviewing and approving release permits. (3.1)

TITLE: **Liquid Radwaste System - Discharge of Liquid Radwaste**

11/15/06
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure is intended to provide a method of safely discharging water from the Liquid Radwaste System. The primary goal is to prevent an inadvertent or unmonitored release. This procedure should be performed in conjunction with CAP A-5 and CAP A-11.

1.2 This procedure is important to safety and to environmental quality.

2. DISCUSSION

2.1 Since this procedure is important to safety and to environmental quality, all documentation for each discharge must be completed entirely and accurately. Under no circumstances shall any discharge be initiated without the appropriate attachments being completed and reviewed by the shift foreman (SFM).

2.2 All discharges must be authorized and approved by the chemistry foreman and the shift foreman. Authorization and approval is documented by signatures on the Authorization for Discharge, Attachment 9.1 and Attachment 9.2 through 9.9 as applicable.

3. RESPONSIBILITIES

3.1 It is the chemistry foreman's responsibility to generate an Authorization for Discharge Form (69-9325) based on analysis of the tank to be discharged.

3.2 It is the SFM's responsibility to issue, review and approve Attachment 9.1 and Attachment 9.2 through 9.9 as applicable prior to issuing the Authorization for Discharge. It is also the SFM's responsibility to ensure all documentation is properly filled out and that all requirements are met prior to starting a discharge.

3.3 It is the operators responsibility to perform Attachment 9.1 and Attachment 9.2 through 9.9 as applicable in its entirety prior to returning it to the SFM for approval. It is the operators responsibility to date strike and document the discharge number on FR-20 at the Auxiliary Control Board. The operator shall log the discharge (batch) number and tank being discharged in the aux senior log. It is also the operators responsibility to monitor the discharge and notify the SFM of any abnormalities which may occur during the discharge. The operator shall return all appropriate paper work to the SFM for review and if satisfactory it will be routed to the chemistry foreman.

TITLE: Liquid Radwaste Discharge Checklist

VERIFICATION FOR ALL TANKS TO BE DISCHARGED

In Service CCW HX to be used for Liquid Radwaste Discharge:
(circle one)
1-1 1-2 2-1 2-2

MANUAL VALVE LINEUP VERIFICATION

VALVE	REQUIRED POSITION	VALVE POS.	INIT	VERIF INIT
LWS-0-436 FLTR DISCH TO UNIT 1 OVERBOARD	Open if 1-1 or 1-2 CCW hxer aligned for discharge Otherwise closed			
LWS-0-439 FLTR DISCH TO UNIT 2 OVERBOARD	Open if 2-1 or 2-2 CCW hxer aligned for discharge. Otherwise closed			
SW-1-67 FLTR DISCH TO 1-1 ASW OVERBOARD	Open if 1-1 CCW hxer aligned for discharge. Otherwise closed			
SW-1-68 FLTR DISCH TO 1-2 ASW OVERBOARD	Open if 1-2 CCW hxer aligned for discharge. Otherwise closed			
SW-2-67 FLTR DISCH TO 2-1 ASW OVERBOARD	Open if 2-1 CCW hxer aligned for discharge. Otherwise closed			
SW-2-68 FLTR DISCH TO 2-2 ASW OVERBOARD	Open if 2-2 CCW hxer aligned for discharge. Otherwise closed			
Sealed Valve Checklist OP K-10L Complete				
CCW Hxer Aligned for Discharge Verified to be In Service by Checking Hxer D/P (VB-1)				

Checklist Performed By ______________ ______/______
(Signature) (Date/Time)

Checklist Verified By ______________ ______/______
(Signature) (Date/Time)

(Staple this checklist to the discharge permit) ________
Init

Liquid radwaste discharge checklist complete and satisfactory ______________ ______________
(SFM Signature) (Date/Time)

Comments ____
____
____

SRO Question 99

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_3__
	Group:	_____	_4__
	K/A:	_____	G2.4.4
	Importance Rating:		4.3

Proposed Question:

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

The operator reports the following:

- Pressurizer pressure is 2250 psig and slowly increasing
- Pressurizer spray valves are closed
- Letdown Orifice Valves are closed
- Charging flow is decreasing
- Pressurizer Level 54% and increasing
- Reactor Cavity Sump levels are 0%

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

A. AP-5, Malfunction of Eagle 21 Protection or Control Channel

B. AP-9, Loss of Instrument Air

C. AP-13, Malfunction of Reactor Pressure Control System

D. AP-18, Letdown Line Failure

Proposed Answer:

B. AP-9, Loss of Instrument Air

Explanation:

A incorrect. If a level channel had failed the spray valves would still operate. If a pressure channel failed, letdown would not be isolated.

B correct. Spray valves and letdown orifices would close on loss of air to containment. Loss of letdown causes pressurizer level to increase, which will cause charging to decrease.

C incorrect. System is responding to increasing pressurizer level.

D incorrect, conditions do not reflect those of a break in letdown.

Technical Reference(s): AP-9

Proposed references to be provided to applicants during examination: None

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures

Question Source:
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: G2.4.4 – Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (4.3)

APPENDIX A (Continued)

INSTRUMENT AIR HEADER TABLES
U-1 AUXILIARY BUILDING/CONTAINMENT

ISOLATION VALVE: AIR-I-1-221 (Inst. Air Receiver inlet)
Location: At Instrument Air Receiver GW areas.

PRINCIPAL AREAS SERVED:
1. U-1 Containment
2. U-1 Aux. Bldg., 100' elev., GE and in U-1 100' GW area.

PRECAUTIONS: 1. *Normally isolated by:
- MS-1-902 (sealed valve), 100' GE area, 140° and
- N_2-1-34 (100' pen GE area, 140°)

MAJOR COMPONENTS SERVED - AUX BLDG	FAILED CONDITION
HCV-637 and 638, RHR Hx Outlets	Fail Open
HCV-670, RHR Hx Bypass	Fail Open
8152, Letdown Hx Inlet	Fail Closed (N_2 Backup)
Containment LRW Iso's	Fail Closed

MAJOR COMPONENTS SERVED - CONTAINMENT	FAILED CONDITION
PCV-474, PORV	Fail Closed
PCV-455C and 456, PORV's	Fail Closed (N_2 B/U Bottles)
PCV-455A and 455B, Pzr Sprays	Fail Closed
LCV-459 and 460, Regen. Hx Letdown Iso's	Fail Closed (N_2 B/U from Stm.Gen. N_2)*
8149A, B, and C, Letdown Orifice Iso's	Fail Closed (N_2 B/U from Stm.Gen. N_2)*
8146 and 8147, Charging to Cold Legs	Fail Open (N_2 B/U Bottles)
8145, Charging to Aux. Spray	Fail Closed (N2 B/U Bottles)
8141A, B, C, and D, RCP Seal Leakoffs	Fail Open
8142, RCP #1 Seal Bypass	Fail Closed
RCS Hot Leg Sample Iso's	Fail Closed (Backup Air Bottles)
Steam Gen Blowdown and Blowdown Samples	Fail Closed
Manipulator Crane	Fails gripper engaged
FCV-584, Inst Air to Containment	Fail Closed
8166 and 8167 Excess Letdown Hx inlet	Fail Closed
HCV-123 Excess Letdown Hx Outlet	Fail Closed
8143 Excess Letdown to VCT or Seal Water Hx	Fail to VCT position

SRO Question 100

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier:	_____	_3__
	Group:	_____	_4__
	K/A:	_____	G2.4.5
	Importance Rating:		3.6

Proposed Question:

GIVEN:
- Reactor trip and safety injection from 100% power due to a Main Steamline break upstream of the 1-1 MSIV.
- The operator reports increasing radiation on steamline 1-3.
- PK11-18, MAIN STM LINE HI RAD is ON
- NO RED or MAGENTA paths exist on the CSFSTs.

The proper procedure flowpath for this event will be E-0 to….

NOTE:
- E-2, Faulted Steam Generator Isolation
- E-3, Steam Generator Tube Rupture
- ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired

A. E-2 to E-3.

B. E-2 to E-3 to ECA-3.1.

C. E-3 to E-2.

D. E-3, foldout page transition to E-2 and then back to E-3.

Proposed Answer:

A. E-2 to E-3

Explanation:
A correct. E-0 diagnostic steps are E-2, E-3, E-1. Crew would first go to E-2.

B incorrect, although there is a faulted SG, it is not the ruptured SG, so a transition to ECA-3.1 is not appropriate.

C incorrect, E-3 is performed after the faulted SG is isolated in E-2.

D incorrect, although there is a transition to E-2 from the E-3 foldout page, this would mean an improper transition to E-3 was made from E-0.

Technical Reference(s): E-0, rev30, page 9.

Proposed references to be provided to applicants during examination: None

Learning Objective: 6827 - State the entry conditions for emergency procedures

Question Source:
Bank – DCPP B-0076 - modified

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:

55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A: G2.4.5 - Knowledge of the organization of the operating procedures network for normal, abnormal and emergency evolutions. (3.6)

DCPP Bank Question:
A reactor trip and SI occurred from 100% power due to a Main Steamline break upstream of the 1-1 MSIV. NO RED or MAGENTA paths exist on the CSFSTs. Select the proper procedural progression from the choices below:

- E-0 "Reactor Trip or Safety Injection"
- E-1 "Loss of Reactor or Secondary Coolant"
- E-1.1 "SI Termination"
- E-1.2 "Post-LOCA Cooldown and Depressurization"
- E-2 "Faulted Steam Generator Isolation"

A. E-0, to E-2, to E-1, to E-1.2

B. E-0, to E-2, to E-1, to E-1.1

C. E-0, to E-1, to E-2, to E-1.1

D. E-0, to E-2, to E-1.1

Answer: D

Question Type: Multiple Choice

Topic: EOP Steam Line Break procedural sequence

System ID: 32956

User ID: **B-0076**

Status: Active

Always select on test: No

Authorized for practice: No

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.