Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	1		
Reactor Trip - Stabilization - Recovery / 1	**Group #**	1		
	K/A #	00007 EK2.02		
	Importance Rating	2.6		
Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects				

Question # 1

A manual reactor trip is initiated from the Control Room:

- Reactor Trip Breaker 'A' indicates green
- Reactor Trip Breaker 'B' indicates red

(1) What is the condition of the reactor?

And

(2) The Condenser Steam Dumps are in the ____(2)____ mode?

A. (1) Tripped
(2) Load Reject

B. (1) Tripped
(2) Plant Trip

C. (1) NOT Tripped
(2) Load Reject

D. (1) NOT Tripped
(2) Plant Trip

Answer: A

Explanation: *The RTBs are in series and when either one of the RTB indicates open, the reactor has been tripped as power to the rod control system has been removed.*

The permissive P-4, RX trip interlock, is based on the RTB and bypass breaker position. The B RTB has a 'b' contact that transfers the steam dumps to plant trip mode from the load reject when the RTB opens. While the B RTB is still closed, the steam dumps will be in Load Reject mode as the transfer has not occurred. The A RTB is an arming signal for the #3 solenoid of the steam dump.

Which RTB performs what function is often confused and makes either RTB and therefore Steam Dump mode a plausible distractor. The distractor of reactor trip not tripped is plausible if it is believed that the RTBs are in parallel, not series, to avoid a single failure criteria causing a reactor trip

A. Correct – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. OTO-SA-00001, EFSAS Verification and restoration, Rev 39, Attachment AQ
2. 7250D64 S010, SNUPPS Projects Functional Diagram Steam Dump Control, Rev 4

References to be provided to applicants during examination: None

Learning Objective:
T61.0110, Systems, LP #27, Reactor Protection, Objective A &D

A. STATE the function and EXPLAIN the design criteria of the Reactor Protection System (RPS).

D LIST all the RPS Permissive Signals, including setpoints, coincidence and function.

T61.0110, Systems, LP #20, Main Steam, Objective I: DISCUSS the four Steam Dump permissive interlocks and EXPLAIN the effects of each on system operation.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

ATTACHMENT AQ
(Page 1 of 1)
Protective Interlocks

AQ1. Protective Interlocks:

Permissive	Setpoint	Coincidence	Function
P-4 Rx Trip	Rx Trip & Bypass Breakers Open		Trips Main Turbine, Feedwater Isolation w/Lo Tavg (564F), Prevents Re-Actuation of SI After Reset, Transfers Steam Dumps from Load Reject to Plant Trip, Arms Steam Dumps
P-6 Source Range	IR>10-10 Amps	1 of 2 IR	Permits Block of SR Trip = HV
P-7 At Power	P-10 or P-13	1 of 2	Unblocks PZR Low Pressure, PZR High Level, Low Flow in > 1 Loop, RCP UV and RCP Underfrequency
P-8 3 Loop Flow	PR > 48%	2 of 4	Unblocks Low Flow in 1 Loop Trip
P-9 Turbine Trip- Rx Trip	PR > 50%	2 of 4	Unblocks Rx Trip on Turbine Trip
P-10 Nuclear At Power	PR > 10%	2 of 4	Feeds P-7, Blocks SR HV, Permits Block of IR Trip, IR Rod Stop and PR Low Setpoint Trip
P-11 Pressure SI	< 1970 PSIG	2 of 3	Permits Block of Lo PZR Press SI and Lo Stm Line Press SI/SLIS which Enables Steam Line Isolation on Hi Negative Pressure Rate
P-12 Low-Low Tavg	550°F	2 of 4	Block Steam Dump
P-13 Turbine at Power	P > 10% imp	1 of 2	Feed P-7
P-14 S/G Hi Level	Level > 91%	2 of 4 Level on 1 of 4 S/G	Trips MFPs Trips Main Turbine FWIS

-END-

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Pressurizer Vapor Space Accident / 3	Group #	1		
	K/A #	00008 G2.4.11		
	Importance Rating	4.0		
Knowledge of abnormal condition procedures.				

Question # 2

Reactor Power is 100%.

- Pressurizer pressure 2100 psig and slowly lowering.
- Containment radiation is 1 Rem/hr and rising.
- VCT level is 60% and lowering at 1% every 4 minutes.
- VCT makeup is out of service.

(1) What is the approximate RCS leak rate?

And

(2) Per the appropriate abnormal procedure, what is the MINIMUM RCS leak rate that requires a manual reactor trip?

A. (1) 5 gpm
(2) 25 gpm

B. (1) 5 gpm
(2) 50 gpm

C. (1) 15 gpm
(2) 25 gpm

D. (1) 15 gpm
(2) 50 gpm

Answer: B

Explanation:

With the indications given, a pressurizer vapor space accident is in progress. Using VCT level trends to approximate the RCS leak size is valid especially if VCT makeup is out of service. Per a note prior to step #5 in OTO-BB-00003, VCT level is 20 gal/% and PZR level is 60 gal/%. 20 gallon/% times 1%/4 minutes = 5 gallon / minute. The distractor of 15 gpm is if the candidate uses the PZR level thumbrule of 60 gal per inch. PZR level is 60 gallon/% times 1%/4 minutes =

15 gallon / minute.

Per OTO-BB-00003, step #5b states the a leak rate of 50 gpm requires a manual reactor trip since the reactor is critical. The distractor of 25 gpm is from the EAL Matrix specifically, SU6.1 for an identified leakage of 25 gpm. Additionally OTO-BB-00001, SG tube leak, has 25 gallons per day (gpd not gpm) as an action level criteria change therefore 25 gpm is plausible and a number that larger than the calculated leak rate in part #1 of the question).

A. Incorrect - the procedural requirement for the RCS leak rate is wrong
B. Correct
C. Incorrect – both are wrong
D. Incorrect – the calculated leak rate is wrong

Technical Reference(s):

1. OTO-BB-00003, RCS Excessive Leakage, Rev 22

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #12, OTO-BB-00003, Objective D,E,H:

D. Given a set of plant conditions or parameters indicating excessive Reactor Coolant leakage, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

E. DISCUSS the cautions and notes contained in OTO-BB-00003, Reactor Coolant System Excessive Leakage.

H. IDENTIFY the conditions that would require a Reactor Trip/Turbine Trip in OTO-BB-00003.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(10)

Comments:

k/a match as the indications present indicate a PZR vapor space leak and upon these indications the control room crew enter OTO-BB-00003, Excessive RCS leakage and knowledge of the abnormal operating procedure is tested by asking the manual trip requirement per step #5 of the OTO.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

VCT Level is 20 gal/% and PZR level is 60 gal/%.

5. DETERMINE If Plant Trip Is Required:

a. DETERMINE leak size and rate of change using any of the following:

- USE trends of VCT level and PZR level

  OR

- COMPARE charging and letdown flows

  OR

- UTILIZE **"GD SG17"** or "T4 SG17:
  - REL0112M (VCT Level)
  - REU0483M (PZR Level)
  - REL0485M (PRT Level)
  - RET0485M (PRT Temp)
  - REU0484M (RCS Tavg)
  - REU0482M (PZR Press)
  - REP0498M (RCS Press)
  - REP0499M (RCS Press)
  - REU0486M (RCS HL Temp)

b. Leak rate – LESS THAN 50 GPM

RESPONSE NOT OBTAINED:

b. PERFORM the following:

1) IF the reactor is critical, THEN PERFORM the following:

a) Manually TRIP the Reactor.

b) STABILIZE the plant using EOPs while continuing with this procedure.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Small Break LOCA / 3	**Group #**	1		
	K/A #	00009 EK1.02		
	Importance Rating	3.5		
Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Use of steam tables				

Question # 3

A Reactor trip and safety injection have occurred due to a small break Loss of Coolant Accident.

- ES-1.2, Post LOCA Cooldown and Depressurization, is in progress.
- Current RCS conditions are as indicated below:
    - RCPs are running
    - BB PI-455A, RCS Narrow Range Pressure 1700 psig
    - BB PI-456, RCS Narrow Range Pressure 1700 psig
    - BB PI-403, RCS Wide Range Pressure 1535 psig
    - Highest Core Exit Thermocouple 530°F
    - Highest RCS Hot Leg Temperature 510°F

On the Foldout Page of ES-1.2, RCS Subcooling is monitored to____(1)_______ and the current value of subcooling is ___(2)_____.

A. (1) Establish ECCS flow, if required
(2) 70°F

B. (1) Control depressurization to prevent voiding in the reactor vessel head
(2) 70°F

C. (1) Establish ECCS flow, if required
(2) 104°F

D. (1) Control depressurization to prevent voiding in the reactor vessel head
(2) 104°F

Answer: A

Explanation:

A. Correct. Per ES-1.2.Foldout Page criteria, RCS subcooling is monitored for SI reinitiation criteria. Current subcooling for given conditions is: saturated temperature for lowest pressure is 600°F – highest temperature of 530°F equals 70°F.

B. Incorrect. Correct subcooling but incorrect reason. The voiding concern is plausible as the procedure caution that it could occur; however there is no subcooling limits given for this concern.
C. Incorrect. Reason given is correct but subcooling is incorrect. Subcooling value is plausible if the non-conservative RCS temperature is used in the calculation.
D. Incorrect. Both subcooling and reason are incorrect. See A and B for explanations.

Technical Reference(s):
1. ASME Steam Tables, Compact Edition, Volume 83, 2006
2. ES-1.2, Post LOCA Cooldown and Depressurization, Rev 14

References to be provided to applicants during examination:
1. ASME Steam Tables, Compact Edition, Volume 83, 2006

Learning Objective: T61.003D, LP D-10, Obj E, Describe the criteria and the basis for information as stated on the ES-1.2, Post LOCA Cooldown and Depressurization, Foldout Page.

Question Source: Bank # __X L16613____
Modified Bank # ______
New _______

Question History: Last NRC Exam ______2013______

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(10)

Comments:

FOLDOUT PAGE FOR ES-1.2

1. **SI REINITIATION CRITERIA**
   IF either condition listed below occurs,
   THEN ESTABLISH ECCS flow as necessary:
   - RCS subcooling - LESS THAN 30°F [50°F]
     OR
   - PZR level - CANNOT BE MAINTAINED GREATER THAN 9% [29%]

2. **SECONDARY INTEGRITY CRITERIA**
   IF BOTH conditions listed below occur,
   THEN Go To E-2, Faulted Steam Generator Isolation, Step 1:
   - Any SG pressure is lowering in an uncontrolled manner OR has completely depressurized.
     AND
   - Affected SG has NOT been isolated using E-2, Faulted Steam Generator Isolation.

3. **E-3 TRANSITION CRITERIA**
   IF either condition listed below occurs,
   THEN ESTABLISH ECCS flow as necessary and Go To E-3, Steam Generator Tube Rupture, Step 1:
   - Any SG level rises in an uncontrolled manner.
     OR
   - Any SG has abnormal radiation.

4. **COLD LEG RECIRCULATION CRITERIA**
   IF RWST level lowers to less than 36%,
   THEN Go To ES-1.3, Transfer To Cold Leg Recirculation, Step 1.

5. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Large Break LOCA / 3	Group #	1		
	K/A #	000011 EK3.02		
	Importance Rating	3.5		
Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: Feedwater isolation				

Question # 4

Per E-0, Reactor Trip or Safety Injection, what is the PRIMARY reason for isolating Main Feedwater during a Large Break LOCA?

A. Minimize the possibility of a secondary side fault.

B. Allow identification of a ruptured SG when monitoring SG levels.

C. Prevent uncontrolled filling of any steam generator and the associated excessive RCS Cooldown.

D. Ensure auxiliary feedwater pump(s) autostart to provide feed to the steam generators for decay heat removal.

Answer: C

Explanation:

Per the E-0 Basis document for step #7, feedwater is isolated ***"****The main feedwater system is isolated on a FW Isolation signal to prevent uncontrolled filling of any steam generator and the associated excessive RCS cooldown which could aggravate the transient, especially if it were a steamline break.*

A. Incorrect – this is the basis/ discussed in E-1 Step #9 Check SG and RCS pressure, and is plausible as it may be believed that overfeeding may results in a SG fault.
B. Incorrect – Plausible if the candidate believes that main feedwater is isolated so that an operator can monitor SG level to detect primary to secondary (or secondary to primary if RCS pressure is lower than SG pressure due to the LOCA). Without main feedwater flow, any change to SG level would be due to tube ruptures since the rupture would not be masked by feedwater flow. Step #15 of E-0 directs the operator to "Check if SG tubes are intact" and the basis of this step was used to develop this distractor.
C. Correct – See explanation above.
D. Incorrect – E-0 step #8 checks AFW pumps are running and it is plausible that feedwater is isolated to receive the autostart of the AFW pumps (since step #7 verifies isolation and Step #8 verifies autostart of the standby ECCS system)

Technical Reference(s):

1. E-0, Reactor Trip or Safety Injection, Rev 16
2. BD-E-0, Basis Document for E-0, Rev 6
3. BD-E-1, Basis document for E-1, Rev 10

References to be provided to applicants during examination: None

Learning Objective: None

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(10)

Comments:

k/a match for a large break LOCA although the information and procedure steps are located in E-0 for convenience instead of E-1. With the conditions stated in the stem, the operator will be performing step #7 prior to a transition to E-1, Loss of reactor or secondary coolant at E-0 step #16. E-1 does not contain any step to verify feedwater isolation as those action are done in E-0 prior to a transition to E-1.

Revised Distractor B of "Minimize the possibility of primary to secondary leakage" per NRC comments. Also reword stem to be consistent with other questions and feedback (i.e match the stem wording of question #54 for example)

EOP STEP: 7 WOG ERG STEP: 5 (partial)

STEP:

CHECK Feedwater Isolation:

PURPOSE:

To ensure feedwater isolation has occurred.

BASIS:

The main feedwater system is isolated on a FW Isolation signal to prevent uncontrolled filling of any steam generator and the associated excessive RCS cooldown which could aggravate the transient, especially if it were a steamline break. Other plant specific valves that receive a FW Isolation, signal, should also be checked.

KNOWLEDGE:

N/A

DEVIATIONS:

The order of ERG steps 5 through 18 have been split between the main body and Attachment A to reduce the time that it takes for the operators to perform E—0 actions as described in the ERG Background Key Utility Decision Points Section and Appendix To Section 4.1 of the Step Description Tables. These steps have been resequenced in logical control board and priority order as allowed by the ERG Background Step Sequence Table for E—0. Consideration has been given to keep operators at different sections of the main control board to prevent interference during automatic action verification.

Split ERG High Level Step 5 into two separate High Level Steps to enhance procedure usage since the SG blowdown and sample valves are checked on a different panel than the feedwater isolation valves.

Added substeps to check main feedwater pumps tripped as allowed by the ERG intent and discussed in the Basis Section for this step. The feedwater chemical injection valves are not included because the manual isolation valves are maintained closed and locked during Modes 1 through 4.

REFERENCES:

Rev 2 DW—96—038

EOP STEP: 8 WOG ERG STEP: 7

STEP:

CHECK AFW Pumps:

PURPOSE:

To ensure AFW pumps are running.

BASIS:

The MD AFW pumps start automatically on an SI signal to provide feed to the SGs for decay heat removal. If SG levels drop below the appropriate setpoint, the turbine-driven AFW pump will also automatically start to supplement the MD AFW pumps.

KNOWLEDGE:

N/A

DEVIATIONS:

The order of ERG steps 5 through 18 have been split between the main body and Attachment A to reduce the time that it takes for the operators to perform E-0 actions as described in the ERG Background Key Utility Decision Points Section and Appendix To Section 4.1 of the Step Description Tables. These steps have been resequenced in logical control board and priority order as allowed by the ERG Background Step Sequence Table for E-0. Consideration has been given to keep operators at different sections of the main control board to prevent interference during automatic action verification.

Added plant specific actions for substep b. RNO to enhance procedure usage and assist the operator in meeting the ERG intent for starting the TD AFW pump.

REFERENCES:

Rev 2 DW-96-038

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of RHR System / 4	Group #	1		
	K/A #	00025 AK3.01		
	Importance Rating	3.1		
Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: Shift to alternate flowpath.				

Question # 5

The Plant is in MODE 5.

- RCS level is 80 inches.
- All CET's read 195ºF and are stable.
- All S/G Wide Range levels are between 88% and 90%.
- All RCP's are off.
- Train 'B' RHR is inoperable for repairs.
- Train 'A' RHR is in service.

Then the 'A' RHR pump trips on overcurrent and cannot be recovered. The crew has entered OTO-EJ-00001, Loss of RHR Flow.

RCS temperature is starting to rise.

What is the preferred method for heat removal under these conditions?

A. Charging Pump injecting flow through the normal charging line, spill through the Pressurizer PORVs.

B. Raise RCS level to ensure loops are filled, feed S/Gs as required, dump steam using the atmosphere steam dump valves.

C. One train of SI valves aligned for injection and a High-Head Safety Injection pump running, spill through the Pressurizer PORVs.

D. Start forced circulation in the RCS, establish Auxiliary feedwater flow and dump steam using the atmospheric steam dump valves.

Answer: B

Explanation:

Per OTO-EJ-00001, step 6.4.1 the control room crew will go to Attachment 1 for subsequent Operator Actions. RCS level of 80" is not reduced inventory (64" is reduced inventory). The operator would progress through, attachment 1 to step #5 and perform the RNO actions. All 4 SG are available as a heat since their level is greater than 86% WR. The operator will raise RCS level to ensure RCS loops are filled, feed the SG using AFW (or other items listed in step RNO5.a.3 and then dump steam per step a.4 to maintain RCS temperature.

A. Incorrect; This charging lineup is established for increasing RCS inventory on a sustained loss of RHR during reduced inventory conditions. The bleed path is the correct RCS bleed path if secondary heat sink cannot be established (i.e. at least two S/G available or if the plant is in mode 6 and they proceed to step 13 of attachment 1). Step #14 of Attachment 1 directs referring to Attachment 7 where this method of feed and bleed is accomplished.
B. Correct.
C. Incorrect; This is an alternate RCS feed and bleed cooling method if secondary heat sink can not be established (i.e. at least two S/G available) and temperature is INCREASING. With the CCP not available (Attachment #7 step #1 C actions and step #2 actions)
D. Incorrect; An RCP would not be started until after natural circulation has been established and RCS cold leg temperatures are greater than 275°F and S/G temperatures are within 10°F of RCS Tcold.

Technical Reference(s):
1. OTO-EJ-00001, Loss of RHR flow, Rev 32

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #62, Loss of RHR Flow, Objective E: Given a set of conditions, DISCUSS the required flowpath to stabilize the plant for a loss of RHR per OTO-EJ-00001, Loss of RHR Flow.

Question Source: Bank # _X L16484_____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2009_________

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(10)

Comments:
Reworded correct answer from bank question and reworded one distractor plus removed one item from stem.

6 SUBSEQUENT OPERATOR ACTIONS

CAUTION:	The standby RHR Pump should not be started unless the cause of the loss of flow is known and corrective action has taken.

NOTE:	If RHR is returned to service this procedure may be terminated at SM/CRS discretion.

6.1 If the RHR pump is cavitating, SECURE it.

6.2 IF in Mid-Loop or Reduced Inventory, THEN Go To OTO-EJ-00003, Loss Of RHR While Operating At Reduced Inventory Or Mid-Loop Conditions.

6.3 IF the cause of Loss of RHR Flow was a pipe break due to a heavy load drop in containment Go To **OTO-EJ-00002,** otherwise continue with the subsequent actions in Step 6.

6.4 IF AC Power, NB01 or NB02 is available, CONTINUE on, IF NOT, Go To Step 6.5.

6.4.1 IF RCS level is BETWEEN 64 inches and 94 inches (lower than 6" below flange but not in Reduced Inventory). Go To Attachment 1, Loss Of RHR NOT In Reduced Inventory With AC Power Available, for Subsequent Operator Actions.

6.4.2 IF RCS level is GREATER THAN 94 inches (Modes 4, 5 or 6), Go To Attachment 2, Loss Of RHR With Level Greater Than 94", for Subsequent Operator Actions.

6.5 For NB01 AND NB02 not available.

6.5.1 IF in MODE 4, Go To **ECA-0.0.** IF NOT, CONTINUE ON.

6.5.2 IF RCS level is GREATER THAN 64 inches (Mode 5 or Mode 6), Go To Attachment 3, Loss Of RHR NOT In Reduced Inventory With Loss Of NB01 and NB02, for Subsequent Operator Actions.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: If in Mode 6 and RCS temperature increases to GREATER THAN 140°, EALs should be referred to AND implemented.

NOTE: Level indicators BBLI53A and BBLI53B may indicate higher than actual as pressure in the vessel increases.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. MONITOR RCS LEVEL • BBLI53A and • BBLI53B	1.
a. RCS level stable or increasing	a. CLOSE the following to Establish Primary Integrity: (1) Letdown • BG HIS-460 • BG HIS-459 • BG HC-128 (2) Rx Vessel Head Vent • BBV0233 (RB-2047-A02B-O above Rx Head) (3) Pressurizer Vent • BBV0085 (RB-2081-D14N-O west side by PZR) (4) Any known drain/vent or leak paths

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. MONITOR CORE EXIT TEMPERATURES FOR INCREASES	2. NOTE: USE W.R. T-Hot for SGs without Nozzle Dams installed.
a. Core Exit Thermocouples available. • MONITOR Plant Computer Displays - GD SG1 and GD SG2	a. MONITOR Wide Range Hot Leg temperature indications. • BB TI-413A – Loop 1 • BB TI-423A – Loop 2 • Temperature Recorders BB TR-413 - Loop 1 BB TR-423 - Loop 2 BB TR-433 - Loop 3 BB TR-443 - Loop 4
3. DETERMINE THE TIME TO BOIL	
a. USE T-Boil calculation results located in the BOP Log for an estimation of RCS behavior.	
4. CHECK RCS STATUS	4.
a. Plant in Mode 5	a. Go To Step 13.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. CHECK STATUS OF RCS HEAT REMOVAL CAPABILITY	5.
	NOTE: For a SG to be available as a heat sink it must have at least 86% W.R. level and RCS Loops filled.
a. START the standby RHR Train in accordance with **OTN-EJ-00001.**	a. At least 2 SG(s) available for RCS heat removal. IF NOT, Go To Step 6. (1) To establish Primary Integrity: • CLOSE Letdown Valves • BG HIS-460 • BG HIS-459 • BG HC-128 • CLOSE Rx Vessel Head Vent BBV0233 (RB-2047-A02B-O above head) • CLOSE Pressurizer Vent BBV0085 (RB-2081-D14N-O west side by PZR) • CLOSE known drain/vent or leak paths.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. (continued)	5. (continued)

a. (continued)

NOTE: With the RCS intact, pressure will increase thus increasing the saturation temperature at which boiling will occur.

NOTE: Seal Injection should be maintained to all RCPs not backseated when RCS level is being changed.

(2) IF Normal Charging is available, RAISE RCS level into the Pressurizer (to >256" on Mid Loop Level indicator).

IF NOT, ESTABLISH charging through the Boron Injection Header as follows:

- OPEN the Boron Injection Header Inlet and Outlet Valves and the RCP seal injection valve for the CCP being used to raise level into the pressurizer.

Train	Boron Injection Header ISO VALVE INLET	Boron Injection Header ISO VALVE OUTLET	RCP SEAL INJ.
"A"	EMHV8803A	EMHV8801A	BGHV8357A
"B"	EMHV8803B	EMHV8801B	BGHV8357B

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. (continued)	5. (continued) a. (continued) (3) FEED SG(s) as required to maintain levels using: • Aux Feed pumps • Non Safety Aux Feed Pump, EOP Addendum 38 • Condensate pumps • S/U feed pump • Fire water EOP Addendum 32, Establishing Emergency Feedwater from Fire Water (4) Use Condenser Steam Dumps (if available) or S/G ASDs to maintain RCS temperature stable. (a) NOTIFY the Count Room Tech as soon as possible of an Atmospheric Steam Dump opening and provide him with opening and closing times. This information is utilized to track the release IF radioactivity is present within the Steam Generator. (5) SM reference EALs (6) DETERMINE the cause of loss of RHR and RESTORE at least one train of RHR to service in accordance with **OTN-EJ-00001.** (7) Return To General Operating Procedure

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. (continued)	5. (continued)
b. DETERMINE the cause of loss of RHR and RESTORE the other train of RHR to operable status in accordance with **OTN-EJ-00001.**	b. ENSURE two S/G'S are available with: (1) GREATER THAN 86% W.R. level AND (2) RCS loops are filled.
c. Ensure compliance with **T/S LCO 3.4.7 or** T/**S LCO 3.4.8** whichever is applicable.	
d. Return To General Operating Procedure.	
6. <u>INITIATE ACTIONS TO EVACUATE CTMT AND CTMT CLOSURE BASED ON PLANT CONDITIONS</u>	6.
a. SOUND the CTMT Evacuation Alarm and make the following announcement: "ATTENTION IN THE PLANT, a loss of RHR has occurred. All nonessential personnel evacuate CTMT". (*Repeat*)	a. NOTIFY the Security Shift Supervisor by radio or telephone to have CTMT evacuated.
b. Start all CTMT Coolers with SW/ESW flow in Slow Speed: • GN HIS-9 • GN HIS-17 • GN HIS-5 • GN HIS-13	
c. DIRECT CTMT Coordinator and pre-designated OTs to COMPLETE CTMT CLOSURE within 4 hours or prior to T-Boil, whichever is less.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. ESTABLISH PRIMARY INTEGRITY a. CLOSE the following: 1) Letdown Valves • BG HIS-460 • BG HIS-459 • BG HC-128 2) BBV0233, Rx Vessel Head Vent (RB-2047-A02B-O above Rx head) 3) BBV0085, Pressurizer Vent (RB-2081-D14N-O west side by PZR) 4) Known drain/vent or leak paths	

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

NOTE: With RCS Loops not filled, RCS cooling is expected to occur through reflux boiling.

NOTE: With the RCS bottled up, pressure will increase as level is increased thus increasing the saturation temperature at which boiling will occur.

NOTE: Seal Injection should be maintained to all RCPs not backseated when RCS level is being changed.

8. INCREASE RCS LEVEL

a. INCREASE Normal Charging to raise level into the Pressurizer (to >256" on Mid Loop Level indicators).

8.

a. With Normal Charging not available, CHARGE through the Boron Injection Header to raise RCS level into the Pressurizer as follows:

- OPEN the Boron Injection Header Inlet and Outlet valves and the RCP Seal Injection Valve for the CCP being used.

	Boron Injection Header ISO VALVE		RCP
Train	INLET	OUTLET	SEAL INJ.
"A"	EMHV8803A	EMHV8801A	BGHV8357A
"B"	EMHV8803B	EMHV8801B	BGHV8357B

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. <u>ESTABLISH HEAT SINK</u>	9.
a. FEED SG(s) as required to maintain levels greater than 7% NR (U-Tubes covered) using: • Aux Feed Pumps • Non Safety Aux Feed Pump, EOP Addendum 38 • Condensate Pumps • S/U Feed Pump	a. FILL available SGs with firewater in accordance with EOP Addendum 32, Establishing Emergency Feedwater from Fire Water, to maintain levels greater than 7% NR (U-Tubes covered).
b. USE condenser steam dumps to control RCS temperature.	b. USE SG ASDs to control RCS temperature (1) NOTIFY the Count Room Tech as soon as possible of an Atmospheric Steam Dump opening and provide him with opening and closing times. This information is utilized to track the release <u>IF</u> radioactivity is present within the Steam Generator.
10. <u>DETERMINE EALs</u>	
a. SM should reference and implement appropriate EALs.	
11. <u>CHECK STATUS OF RHR</u>	
a. DETERMINE the cause of loss of RHR and restore two trains of RHR to operable status with one in operation in accordance with **OTN-EJ-00001.**	
b. Ensure compliance with **T/S LCO 3.4.8**.	
12. <u>RETURN TO GENERAL OPERATING PROCEDURE</u>	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. <u>INITIATE ACTIONS TO PROTECT PERSONNEL IN CTMT AND INITIATE CTMT CLOSURE</u>	13.
a. SOUND the CTMT Evacuation Alarm and make the following announcement: "ATTENTION IN THE PLANT, a loss of RHR has occurred. All nonessential personnel evacuate CTMT". (*Repeat*)	a. NOTIFY the Security Shift Supervisor by radio or telephone to have CTMT evacuated.
b. Start all CTMT Coolers with SW/ESW flow in Slow Speed: • GN HIS-9 • GN HIS-17 • GN HIS-5 • GN HIS-13	
c. DIRECT CTMT Coordinator and pre-designated OTs to COMPLETE CTMT CLOSURE within 4 hours or prior to T-Boil, whichever is less.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

14. CHECK ON RHR STATUS

a. START the standby train of RHR in accordance with **OTN-EJ-00001.**

14.

a. ESTABLISH level and a heat sink.

> NOTE: Seal Injection should be maintained to all RCPs not backseated when RCS level is being changed.

(1) IF Normal Charging is available, RAISE RCS level to 94" (BBLI53A or B)

IF NOT, ESTABLISH charging through the Boron Injection Header as follows:

- OPEN the Boron Injection Header inlet and outlet valves and the RCP Seal Injection Valve for the CCP being used to raise level into the pressurizer.

	BORON INJECTION HEADER ISO VALVE		RCP
Train	INLET	OUTLET	SEAL INJ.
"A"	EMHV8803A	EMHV8801A	BGHV8357A
"B"	EMHV8803B	EMHV8801B	BGHV8357B

(2) FEED SG(s) without nozzle dams as required to establish and maintain levels greater than 7% NR (U-Tubes covered) using:

- Aux feed pumps
- Non Safety Aux Feed Pump, EOP Addendum 38
- Condensate pumps
- S/U feed pump
- Fire water EOP Addendum 32, Establishing Emergency Feedwater from Fire Water

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. (continued)	14. (continued) a. (continued) (3) MONITOR RCS temperature using Incore Thermocouples or W.R. T-Hot if Thermocouples aren't available. IF temperature increases OR T-Boil indicates that boiling is imminent, THEN Fill and Spill or Feed and Bleed is recommended for core cooling. (Refer To Attachment 7, Raising Loop Level) (4) Return To Step 13 of this Attachment.
15. <u>RESTORE RHR</u> a. DETERMINE the cause of loss of RHR and restore two trains of RHR to operable status with one in operation in accordance with **OTN-EJ-00001.** b. Refer To **T/S LCO 3.9.6.**	
16. <u>RETURN TO GENERAL OPERATING PROCEDURE</u>	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE: Step 1 is a continuous action step.	
1. CHECK PLANT STATUS	1.
a. Plant in Mode 5.	a. Go To Step 8.
2. MONITOR CORE EXIT TEMPERATURE	2.
a. Core Exit Thermocouples are available. • MONITOR Plant Computer Displays - GD SG1 and GD SG2	a. MONITOR W.R. HOT LEG Temperature indications for intact S/G. • BB TI-413A Loop 1 • BB TI-423A Loop 2 • Temperature Recorders BB TR-413 Loop 1 BB TR-423 Loop 2 BB TR-433 Loop 3 BB TR-443 Loop 4
3. CHECK RHR STATUS	3.
a. Standby train of RHR available for service.	a. Go To Step 4.
b. START the standby train of RHR in accordance with **OTN-EJ-00001.**	
c. DETERMINE the cause of loss of RHR and RESTORE the other train of RHR to operable status in accordance with **OTN-EJ-00001.**	c. ENSURE two S/G's are available with: (1) GREATER THAN 86% W.R. level AND (2) RCS Loops are filled
d. Ensure compliance with **T/S LCO 3.4.7 or T/S LCO 3.4.8** whichever is applicable.	
e. Return To General Operating procedure.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. ESTABLISH A HEAT SINK	4.

CAUTION: An inability to pressurize the RCS can severely degrade the effectiveness of natural circulation cooling. The adverse impact on natural circulation is due to flashing and steam voiding in the upper reaches of the Steam Generator U-tubes. Additionally, gasses coming out of solution may hamper natural circulation flow. The minimum RCS-SG ΔT that would ensure sufficient natural circulation flow is 28°C (50°F). This corresponds to a saturation pressure of approximately 40 psia (25 psig).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Close known vent/drain or leak paths.	
b. FEED SG(s) without nozzle dams as required to establish and maintain levels greater than 86% WR using: • Aux feed pumps • Non Safety Aux Feed Pump, EOP Addendum 38 • Condensate pumps • S/U feed pumps • Fire water EOP Addendum 32, Establishing Emergency Feedwater from Fire Water	
c. USE Condenser Steam Dumps to control RCS temperature.	c. USE SG ASDs to control RCS temperature. (1) NOTIFY the Count Room Tech as soon as possible of an Atmospheric Steam Dump opening and provide him with opening and closing times. This information is utilized to track the release <u>IF</u> radioactivity is present within the Steam Generator.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. <u>DETERMINE EALs</u>	
a. SM should reference and implement appropriate EALs.	
6. <u>CHECK STATUS OF RHR</u>	
a. DETERMINE the cause of loss of RHR and RESTORE at least one train of RHR to service in accordance with **OTN-EJ-00001.**	
b. ENSURE compliance with applicable T/Ss. • T/S LCO 3.4.7 • T/S LCO 3.4.8	b. Return To Step 1 until T/S compliance restored
7. <u>RETURN TO GENERAL OPERATING PROCEDURE</u>	
8. <u>CHECK PLANT STATUS</u>	8.
a. Plant in Mode 6.	a. Go To Step 10.
b. CHECK Rx Vessel water level GREATER THAN 23 feet above the flange • BB LI-462 reads GREATER THAN 23.1%	b. Go To Step 13.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. <u>CHECK RHR STATUS</u>	9.
a. CHECK standby train of RHR available.	a. Go To Step 15.
b. START the standby train of RHR in accordance with **OTN-EJ-00001**.	
c. REFERENCE **T/S LCO 3.9.5** and ensure the Diesel Generator is operable for the train in operation.	
d. DETERMINE the cause of loss of RHR and restore it to operable status if applicable.	
e. Return To General Operating Procedure.	
10. <u>MAINTAIN RCS TEMPERATURE</u>	10.
a. USE Condenser Steam Dumps to maintain RCS temperature stable.	a. USE SG ASDs to maintain RCS temperature stable. (1) NOTIFY the Count Room Tech as soon as possible of an Atmospheric Steam Dump opening and provide him with opening and closing times. This information is utilized to track the release <u>IF</u> radioactivity is present within the Steam Generator.
11. <u>ENSURE COMPLIANCE WITH T/S</u>	11. DETERMINE the cause of loss of RHR and RESTORE to operable status to ensure compliance with T/S. Return to Step 10.
a. Adequate RHR train operability for the present plant conditions. Reference **T/S LCO 3.4.6 and T/S LCO 3.5.3.**	
12. <u>RETURN TO GENERAL OPERATING PROCEDURE</u>	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. <u>CHECK RHR STATUS</u>	13.
a. CHECK the standby train of RHR available.	a. Go To Step 19.
b. START the standby train of RHR in accordance with **OTN-EJ-00001.**	
c. DETERMINE the cause of loss of RHR and RESTORE 2 trains of RHR to operable status to comply with **T/S LCO 3.9.6**.	
14. <u>RETURN TO GENERAL OPERATING PROCEDURE</u>	
15. <u>INITIATE ACTIONS TO PROTECT PERSONNEL IN CTMT AND INITIATE CTMT CLOSURE</u>	15.
a. SOUND the CTMT Evacuation alarm and make the following announcement: "ATTENTION IN THE PLANT, a loss of RHR has occurred. All nonessential personnel evacuate CTMT." (*Repeat*)	a. NOTIFY the Security Shift Supervisor by radio or telephone to have CTMT evacuated.
b. Start all CTMT Coolers with SW/ESW flow in Slow Speed: • GN HIS-9 • GN HIS-17 • GN HIS-5 • GN HIS-13	
c. DIRECT CTMT Coordinator and pre-designated OTs to COMPLETE CTMT CLOSURE within 4 hours or prior to T-Boil, whichever is less.	
16. <u>MONITOR REFUEL POOL LEVEL</u>	16.
a. ENSURE PZR Level BB LI-462 stable or increasing.	a. MONITOR sump levels and isolate any known drain paths.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. MAKE PREPARATION FOR ESTABLISHING LONG TERM COOLING	17.
a. DETERMINE the cause of loss of RHR and restore at least one train of RHR to operation.	a. LINEUP SFP Cleanup System to the Refuel Pool through a SFP Hx using **OTN-EC-00001,** Addendum 4, Refuel Pool Cleanup Operations.
b. Ensure compliance with **T/S LCO 3.9.5**.	b. Do Not proceed until T/S LCO 3.9.5 compliance restored.
18. RETURN TO GENERAL OPERATING PROCEDURE	
19. INITIATE ACTIONS TO PROTECT PERSONNEL IN CTMT AND INITIATE CTMT CLOSURE	19.
a. SOUND the CTMT Evacuation alarm and make the following announcement: "ATTENTION IN THE PLANT, a loss of RHR has occurred. All nonessential personnel evacuate CTMT." (*Repeat*)	a. NOTIFY the Security Shift Supervisor by radio or telephone to have CTMT evacuated.
b. Start all CTMT Coolers with SW/ESW flow in Slow Speed: • GN HIS-9 • GN HIS-17 • GN HIS-5 • GN HIS-13	
c. DIRECT CTMT Coordinator and pre-designated OTs to COMPLETE CTMT CLOSURE within 4 hours or prior to T-Boil, whichever is less.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20. <u>EVALUATE PLANT STATUS</u>	
a. ENSURE adequate Charging from the RWST based on: (1) Refuel Pool level and (2) Time after shutdown	a. RESTORE the following: (1) Any CCP incapable of injection per OSP-BG-00002, Verify One CCP Incapable of Injection Into RCS (2) SI pumps that are incapable of injection per OSP-EM-00002, Rendering SI Pumps Incapable of Injection. IF both CCPs and both SI pumps are incapable of Injection into the RCS Go To Attachment 7, Raising Loop Level.
b. DETERMINE the cause of loss of RHR and RESTORE two trains of RHR to operable status with one in operation.	c. Return to Step 20.a
c. Ensure compliance with **T/S LCO 3.9.6**.	
21. <u>RETURN TO GENERAL OPERATING PROCEDURE</u>	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. <u>INITIATE ACTIONS TO PROTECT PERSONNEL IN CTMT AND CTMT CLOSURE</u>	1.
a. SOUND the CTMT evacuation alarm and make the following announcement: "ATTENTION IN THE PLANT, a loss of RHR has occurred. All nonessential personnel evacuate CTMT." *(Repeat)*	a. NOTIFY the following personnel by radio or telephone. (1) Security Shift Supervisor for CTMT evacuation. <u>AND</u> (2) CTMT Coordinator and pre-designated OTs to COMPLETE CTMT CLOSURE within 4 hours or prior to T-Boil, whichever is less. (3) Go To Step 2.
b. DIRECT CTMT Coordinator and pre-designated OTs to COMPLETE CTMT CLOSURE within 4 hours or prior to T-Boil, whichever is less.	
<u>NOTE</u>: If in Mode 6 and temperature increases to GREATER THAN 140° the SM should Refer To and implement EALs.	
2. <u>DETERMINE EALs</u>	
a. SM should Refer To and implement EALs	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Step 3 is a continuous action step.

3. RESTORE POWER TO NB01 AND NB02

   a. Refer To Attachment 4 for restoration of power to NB01 and NB02.

4. DETERMINE TIME TO BOIL BASED ON EXISTING CONDITIONS

   a. USE T-Boil calculation results located in the BOP Log for an estimation of RCS behavior.

5. ESTABLISH PRIMARY INTEGRITY

   a. CLOSE the following valves:

      (1) Letdown Isolation Valves
         - BG HIS-460 and
         - BG HIS-459
         - BG HC-128

      (2) Rx Vessel Head Vent
         - BBV0233 (RB-2047-A02B-O above Rx Head)

      (3) Pressurizer Vent
         - BBV0085 (RB-2081-D14N-O west side by PZR)

      (4) RCP Seal Water Return Outer CTMT Isolation
         - BGHV8100 (South Pipe Pen Room, Pen 024)

      (5) Any known drain or vent paths

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. MONITOR CORE EXIT TEMPERATURES	
a. Core Exit Thermocouples • MONITOR Plant Computer Displays - GD SG1 and GD SG2	a. NOTE: Use of W.R. T-Hot should be used for SGs without nozzle dams installed. MONITOR Wide Range Hot Leg temperature indications • BB TI-413A - Loop 1 • BB TI-423A - Loop 2 • Temperature Recorders BB TR-413 - Loop 1 BB TR-423 - Loop 2 BB TR-433 - Loop 3 BB TR-443 - Loop 4

NOTE: Due to the limitations of EOP Addendum 32, Establishing Emergency Feedwater from Fire Water, only one SG may be filled at a time.

NOTE: DO NOT Align Firewater To SGs with nozzle dams installed.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. MAKE PREPARATIONS FOR ESTABLISHING A HEAT SINK	
a. ALIGN firewater to all available SGs in accordance with EOP Addendum 32, Establishing Emergency Feedwater from Fire Water.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION: If RCS pressure increases, BBLI53A and B will start to show inaccuracies, since they are vented to the pressurizer.	
NOTE: Step 8 is a continuous action step.	
8. MONITOR RCS LEVEL	8.
a. MONITOR RCS level to remain GREATER THAN a reduced inventory situation. (64")	a. IF level decreases to reduced inventory, Go To OTO-EJ-0003, Loss Of RHR While Operating At Reduced Inventory Or Mid-Loop Conditions.
NOTE: Step 9 is a continuous action step.	
9. MONITOR FOR BOILING IN THE RCS • RCS Temperature Approaching 200˚F OR • RCS Level Shows Significant Rise • BB LI-53A • BB LI-53B	9. Go To Step 13

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: RWST gravity feed to RCS must be established prior to the onset of core boiling to ensure RWST head pressure is sufficient to provide flow to RCS.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. <u>ESTABLISH RCS FEED AND BLEED</u>	10.

a. OPEN all available vents;
• Rx Vessel Head Vent
  • BBV0233 (RB-2047-A02B-O Above Rx Head)
• Pressurizer Vent
  • BBV0085 (RB-2081-D14N-O West side by PZR)
• Any other vent path available	

b. Locally ALIGN <u>One</u> of the following paths to reestablish RWST feed path:
• <u>“A” RHR Cold Leg Injection path</u>
  • OPEN the following:
    • BNHV8812A (A RHR P Rm)
    • EJHCV0606 (A RHR Hx Rm)
    • EJHV8809A (North Pipe Pen Rm, Pen 082)
  • CLOSE the following:
    • EJFCV0618 (A RHR Hx Rm)
    • EJHV8716A (A RHR Hx Rm)
• <u>“B” RHR Cold Leg Injection Path</u>
  • OPEN the following:
    • BNHV8812B (B RHR P Rm)
    • EJHCV0607 (B RHR Hx Rm)
    • EJHV8809B (South Pipe Pen Rm, Pen 027)
  • CLOSE the following:
    • EJFCV0619 (B RHR Hx Rm)
    • EJHV8716B (B RHR Hx Rm)
	b. Locally OPEN the following valves to establish feed path.
	
	<u>NOTE</u>: BN8717 RHR SPLY to RWST ISO is a non-frangible lock, which uses a PA-300 key. This key can be obtained from key issue tag 101 with SM permission.
	
	• BN8717 (A RHR Hx Rm)
	
	<u>AND</u>
	
	• EJHV8840 (South Piping Pen Room Pen 021)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. ESTABLISH A HEAT SINK	
a. FEED all available SGs as required with feedwater to facilitate natural circulation or reflux boiling. EOP Addendum 38, Non Safety Auxiliary Feewater Pump, to raise levels greater than: • 86% WR for Natural Circulation • 7% NR for Reflux Boiling	a. Feed all available SGs as required with firewater to facilitate natural circulation or reflux boiling. EOP Addendum 32, Establishing Emergency Feedwater from Fire Water, to raise levels greater than: • 86% WR for Natural Circulation • 7% NR for Reflux Boiling
b. OPEN available SG ASDs (1) NOTIFY the Count Room Tech as soon as possible of an ASD opening and provide opening and closing times. This information is utilized to track the release IF radioactivity is present within the Steam Generator.	
12. CHECK FEED AND BLEED STATUS	12.

a. (1) MONITOR RWST level to ensure level is decreasing
• BN LI-930
• BN LI-931
• BN LI-932
• BN LI-933
(2) Significant increase in BB LI-53A or B.
	a. Perform the following:
	• DIRECT pre-designated EO or CTMT Coordinator to locally OPEN the available accumulator's outlet valve.
	  • EPHV8808A
	  • EPHV8808B
	  • EPHV8808C
	  • EPHV8808D
	• After contents have injected into the RCS, VENT the respective accumulator when pressure decreases to approx. 10 psi. (REP0490A thru REP0497A)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. CHECK ON PLANT STATUS a. Continue to MONITOR RCS conditions until AC power is restored. b. CHECK status of AC power restoration. c. CHECK status of CTMT Closure. d. DISPATCH an Equipment Operator to verify running: • Security Diesel Generator • TSC Diesel Generator • EOF Diesel Generator	
14. CHECK AC POWER RESTORED	14. Return To Step 2, observe note prior to the step.
15. ENSURE COMPLIANCE WITH T/S a. Refer To **T/S:** **T/S LCO 3.4.7** **T/S LCO 3.4.8** **T/S LCO 3.9.6**	
16. RETURN TO GENERAL OPERATING PROCEDURE	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. <u>TRY TO RESTORE POWER TO EITHER NB01 OR NB02</u>	1.
<u>NOTE</u> Ensuring LSELS is energized will ensure automatic load shed and sequence.	<u>NOTE</u> It should take approximately 12 seconds for the NB Bus to reenergize AFTER LSELS REENERGIZATION.
a. ENSURE the LSELS systems are in operation for all operable Emergency Diesel Generators.	a. ENERGIZE all available LSELS trains, NF039A or NF039B, associated with operable diesel generator.
(1) NF039A	• DC power Supply SW 8N25-1 to <u>ON</u>
(a) DC Power Supply SW 8N25-1 <u>ON</u>	<u>AND</u>
(b) Output Relay/ATI DC Pwr SW 8N28-1 <u>ON</u>	• Output relay/ATI DC Pwr SW 8N28-1 <u>ON</u>
(2) NF039B	
(a) DC Power Supply SW 8N25-1 <u>ON</u>	
(b) Output Relay/ATI DC Pwr 8N28-1 <u>ON</u>	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION An ESW pump should be kept available to provide diesel generator cooling.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (continued)	1. (continued)
b. VERIFY NB01 OR NB02 energized.	b. INITIATE manual load shed for both NB buses by placing the following equipment in PULL TO LOCK/STOP. • CCP's • SI Pumps • RHR Pumps • Containment Spray Pumps • CCW Pumps • MD AFW Pumps • Containment Fan Coolers • Instrument Air Compressors • Fuel Pool Cooling Pumps • PK Battery Chargers
c. CHECK Diesel Generators NE01 OR NE02 at least one running, and available for loading.	c. START diesel generator NE01 OR NE02 from main control board. Momentarily PRESS the Diesel Generator start pushbutton KJ HS-8A for NE01 or KJ HS-108A for NE02. VERIFY that the Diesel Generator STARTS and is available for loading. OR START diesel generator NE01 OR NE02 locally. Use **EOP Addendum 21** for locally starting NE01 and NE02.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION The EMERG DG BKR will NOT AUTO CLOSE IF THE Normal or Alternate NB Bus Feeder Breakers are closed.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (continued)	1. (continued)
d. VERIFY NB01 OR NB02 energized from the Emergency Diesel Generator.	d. IF a Diesel Generator is running, ENSURE its NB Bus Normal and Alternate Breakers are OPEN. NOTE: It may be necessary to reset the anti-pump device by taking the DG supply breaker to the CLOSE or TRIP position. IF the NB Bus does not reenergize, THEN locally STOP the Diesel Generator(s). Use **OTN-NE-0001A,** STANDBY DIESEL GENERATION SYSTEM - TRAIN 'A', or **OTN-NE-0001B,** STANDBY DIESEL GENERATION SYSTEM - TRAIN 'B', as applicable. AND Energize NB01 OR NB02 using any available power supply. USE Attachment 5, Restoration Of Offsite Power if Offsite power is available. If Offsite power is not available Energize NB01 or NB02 using EOP Addendum 39, Alternate Emergency Power Supply.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (continued)	1. (continued)
e. CHECK NB01 AND NB02 at least one energized from any available AC source.	e. CONTINUE efforts to restore power to NB01 OR NB02 Via an Emergency Diesel or an offsite source. AND Locally CHECK Spent Fuel Pool level above the top of the Spent Fuel Pool Cooling discharge piping. (10 inch piping penetration, located on the Plant West wall of the Spent Fuel Pool.) IF NOT, THEN consult Engineering to initiate makeup to the spent fuel pool. USE Fire Water System via diesel fire pumps and the hose rack(s) on the 2047 elevation of the Fuel Building.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (continued)	1. (continued)
CAUTION The loads placed on the energized AC bus should not exceed the capacity of the source (520 amps for XNB01 / XNB02).	
f. MANUALLY load equipment onto Energized AC Emergency Bus as needed: • ESW Pump • Instrument Air Compressor • CCW Pump • CCP • CTMT Coolers	
g. PERFORM Attachment 6, Verification Of Equipment Loaded Onto AC Emergency Bus.	
h. RESTORE RHR in accordance with **OTN-EJ-00001.**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

1. This attachment restores offsite power assuming that all systems are available. Each step should be evaluated independently depending on the operability status of the components involved in the steps.
2. This attachment restores offsite power in the following sequence;
   Step 1. Re-energizes the Switchyard.
   Step 2. Startup XFMR.
   Step 3. XNB01.
   Step 4. XNB02.
   Step 5. Restores power to NB01 and NB02.
   Step 6. Restores power to PA01 and PA02
   Step 7. Restores electrical load centers and motor control centers.

CAUTION: Prior to closing any 4160V or higher breakers, ENSURE all relays are reset using the following guidance:

1. NOTIFY System Relay Service when a 4160 VAC or higher breaker trips with an electrical relay target or if a relay is in question. An apparent cause shall be determined prior to attempting to close the breaker.
2. Electricians should PERFORM a megger test if a 4160 VAC or higher voltage pump trips due to electrical fault (protective relay operation).
3. Relay flags/targets in the switchyard should not be reset until authorized by the TD or LD.
4. IF a tripped component is required for plant safety the SM has the authority to RESET the relay and close the breaker even though a cause has not been determined.
5. UE form #604 should be filled out for all protective relay devices activated on 4160V or higher breakers.
6. LOG each protective relay actuated and/or operated in the Control Room narrative log. Log specific relays by component ID.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. CHECK OFFSITE POWER AVAILABLE.	1.
a. OFFSITE POWER ANY LIGHTS LIT (1) CAL-BLAND-1 (2) MTGY-CAL-7 (3) MTGY-CAL-8 (4) CAL-LSCR-2	a. CONTACT the Power Supervisor and the Transmission Dispatcher for the status of restoring an OFFSITE power source. PERFORM this attachment when an offsite power source has been restored. Return To Step in Effect.
b. ENSURE all relays are reset for the appropriate switchyard breakers and call the Transmission Dispatcher.	b. RESET relays at Transmission Dispatchers request.
c. CLOSE the following breakers to re-energize available Switchyard Buses; • PCB-V85 (BUS B MTGY-CAL-7) • PCB-V81 (BUS A MTGY-CAL-7) • PCB-V45 (BUS B CAL-BLAND-1) • PCB-V71 (BUS A MTGY-CAL-8) • PCB-V75 (BUS B MTGY-CAL-8) • PCB-V51 (BUS A CAL-LSCR-2)	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. <u>CHECK IF TRANSFORMER XNB01 IS ENERGIZED</u> USING the Main Control Board electrical mimic CHECK that breakers are closed between the energized offsite source and transformer XNB01.	2. RE-ENERGIZE XNB01 AS FOLLOWS: <u>CAUTION</u> Do <u>NOT</u> re-energize XNB01 if a XNB01 XFMR Lockout (ANN 19A) is energized. a. IF a XNB01 XFMR Lockout (Ann 19A) is ENERGIZED Go To Step 3. <u>Panel RL015</u> b. ENSURE 4.16 KV Bus NB01 BKR NB0112 (NB HIS-2) - PTL c. ENSURE 4.16 KV Bus NB02 BKR NB0212 (NB HIS-5) - PTL <u>Switchyard</u> d. <u>IF</u> switchyard Bus A is energized, <u>THEN</u> (1) Ensure 13.8 KV BKR 52-3 is OPEN. (2) CLOSE 13.8 KV BKR 52-1 e. <u>IF</u> ONLY switchyard Bus B is energized, <u>THEN</u> (1) Ensure 13.8 KV BKR 52-1 is OPEN. (2) CLOSE 13.8 KV BKR 52-3

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. <u>CHECK THE STARTUP TRANSFORMER IS ENERGIZED</u> USING the Main Control Board electrical mimic CHECK that breakers are closed between the energized offsite source and the Startup Transformer.	3. RE-ENERGIZE THE STARTUP TRANSFORMER AS FOLLOWS: <u>CAUTION</u> Do <u>NOT</u> re-energize the Startup Transformer if a S/U XFMR Lockout (Ann 14A) is energized. a. IF a S/U XFMR Lockout (Ann 14A) is ENERGIZED Go To Step 5. <u>Panel RL016</u> b. ENSURE the Startup XFMR XMR01 Breaker PA0202 (PA HIS-8) - PTL c. ENSURE the Startup 13.8 KV Breaker PA0110 (PA HIS-6) - PTL <u>Panel RL014</u> d. When switchyard Bus A is re-energized, CLOSE PCB-V41, STARTUP XFMR 1 Bus A. e. When Callaway Bland 1 is re-energized, CLOSE PCBV43 and PCBV41 Bus TIE breakers.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. <u>CHECK IF TRANSFORMER XNB02 IS ENERGIZED</u> USING the Main Control Board electrical mimic CHECK that breakers are closed between the energized offsite source and transformer XNB02.	4. RE-ENERGIZE XNB02 AS FOLLOWS <u>CAUTION</u> Do <u>NOT</u> re-energize XNB02 if a XNB02 XFMR Lockout (ANN 22A) is energized. a. IF a XNB02 XFMR Lockout (Ann 22A) is ENERGIZED Go To Step 5. <u>Panel RL015</u> b. ENSURE 4.16 KV Bus NB01 BKR NB0109 (NB HIS-3) - PTL c. ENSURE 4.16 KV Bus NB02 BKR NB0209 (NB HIS-4) - PTL <u>Panel RL016</u> d. CLOSE XMR01 to XNB02 Breaker PA0201 (NB-HIS-1)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION IF a single offsite source is used to supply both NB01 and NB02, DO NOT exceed the following current limits.

(1) DO NOT exceed 520 AMPs on XNB01 XFMR as read on MSIIXNB011, ESF XFMR XNB01 AMMETER.

(2) DO NOT exceed 520 AMPs on XNB02 XFMR as read on NB II-8 XMR01 to XNB02 AMPs (Located on RL016).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. RESTORE NB01 AND NB02 TO EITHER NORMAL OR ALTERNATE POWER	5.

NOTE It is preferred to energize both NB buses even if only one off-site source is available to facilitate recovery actions.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. ESF Transformer(s) XNB01 AND XNB02 at least one Energized.	a. Go To Step 6 of this Attachment. CAUTION Do NOT re-energize an NB bus if its Bus Lockout Annunciator is energized. (ANNs 18A, 21A)
b. NB01 and NB02 both energized from Emergency Diesels.	b. RESTORE offsite power to de-energized NB Busses. Use **OTN-NB-0001A,** 4.16KV Vital (Class IE) Electrical System - Train 'A', or **OTN-NB-0001B,** 4.16 KV Vital (Class 1E) Electrical System - Train 'B'.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE It is NOT preferred to go to a single offsite source (XNB01 OR XNB02) supplying both NB buses if an Emergency Diesel is supplying an NB Bus.	
5. (continued)	(continued)
c. RESTORE offsite power to NB busses that are being supplied by Emergency Diesels. USE **OTN-NE-0001A,** Standby Diesel Generation System - Train 'A', or **OTN-NE-0001B,** Standby Diesel Generation System - Train 'B'.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. RE-ENERGIZE PA01 AND PA02 FROM THE STARTUP TRANSFORMER	6.
a. Startup XFMR Energized	a. PERFORM Step 6 when the Startup transformer has been re-energized. Go To Step 7.
Panel RL016	
CAUTION Do NOT re-energize a PA Bus if its Bus Lockout Annunciator is energized. (ANN 15A)	
b. ENSURE Unit Aux 13.8 KV Breaker PA0211 (PA HIS-13) - PTL	
c. ENSURE Unit Aux 13.8 KV Breaker PA0101 (PA HIS-1) - PTL	
d. POSITION 13.8 KV Source Select Switch (PA HS-7) to ST-UP.	
CAUTION An auto BKR closure will occur when either PA HIS-8 or PA HIS-6 are taken out of PTL.	
e. CLOSE Startup XFMR XMR01 Breaker PA0202 (PA HIS-8)	
f. CLOSE Startup 13.8 KV Breaker PA0110 (PA HIS-6)	
g. POSITION 13.8 KV Source Select Switch to OFF (PA HS-7).	
7. RE-ENERGIZE ELECTRICAL LOAD CENTERS AND MOTOR CONTROL CENTERS DE-ENERGIZED DUE TO LOSS OF OFFSITE POWER AS REQUIRED	
8. RETURN TO Attachment 4, Step 1e.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE After completing this Attachment, Return To Step 1h of Attachment 4.

1. IF NB01 is energized, THEN ensure the following equipment is energized. IF NB01 NOT energized, THEN perform Step 2 of this Attachment

    a. 480 Volts Busses NG01 and NG03

        (1) BKR NB0113 - CLOSED (NG HIS-1)

        (2) BKR NB0110 - CLOSED (NG HIS-2)

        (3) BKR NG0101 - CLOSED (NG HIS-9)

        (4) BKR NG0301 - CLOSED (NG HIS-11)

    b. Battery Chargers

        (1) NK21, BKR NG0103 - CLOSED

        (2) NK23, BKR NG0303 - CLOSED

    c. Instrumentation And Control

        (1) NK41, BKR NK0104 - CLOSED

        (2) NK51, BKR NK0105 - CLOSED

        (3) NK43, BKR NK0304 - CLOSED

    d. Emergency Lighting

        (1) Control Room Emergency Lighting, BKR NK5120 - CLOSED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. IF NB02 is energized, THEN ensure the following equipment is energized. IF NB02 is NOT energized, THEN Go To Step 1e of Attachment 4	
a. 480 Volt Busses NG02 and NG04	
(1) BKR NB0213 - CLOSED (NG HIS-8)	
(2) BKR NB0210 - CLOSED (NG HIS-7)	
(3) BKR NG0201 - CLOSED (NG HIS-12)	
(4) BKR NG0401 - CLOSED (NG HIS-14)	
b. Battery Chargers	
(1) NK24, BKR NG0203 - CLOSED	
(2) NK22, BKR NG0403 - CLOSED	
c. Instrumentation And Control	
(1) NK42, BKR NK0204 - CLOSED	
(2) NK44, BKR NK0404, - CLOSED	
(3) NK54, BKR NK0405 - CLOSED	
d. Communications	
(1) QF076, Emergency Supply, BKR PN0803 - CLOSED	
3. Go To Step 1h of Attachment 4	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE DO NOT fill SGs with nozzle dams installed.	
1. EVALUATE ALTERNATIVE COOLING PLANS	1.

a. RESTORE SGs to facilitate reflux boiling.
• FILL SGs to greater than 7% NR (U-Tubes covered) using:
  • Auxiliary Feed Pumps
  • Non Safety Auxiliary Feedwater Pump, EOP Addendum 38
  • Condensate Pumps
  • S/U Feed Pump
• OPEN respective SG ASDs
	a. RESTORE SGs to facilitate reflux boiling.
	NOTE: Due to the limitations of EOP Addendum 32, Establishing Emergency Feedwater from Fire Water, only one SG may be filled at a time.
	• FILL SGs to greater than 7% NR (U-Tubes covered) using EOP Addendum 32, Establishing Emergency Feedwater from Fire Water
	• OPEN respective SG ASDs

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (continued)	1. (continued)

NOTE: If the RCS is intact, at least 2 SGs are needed to maintain RCS pressure low enough for gravity feed of the RWST.

CAUTION: RWST gravity feed to the RCS must be established prior to the onset of core boiling to ensure RWST head pressure is sufficient to provide flow to the RCS.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. ESTABLISH RCS Feed And Bleed or Fill and Spill using RWST gravity feed to RCS. 1) IF the RCS is Intact, THEN ESTABLISH Feed and Bleed: a) ALIGN <u>One</u> of the following feed paths: • <u>"A" RHR Cold Leg Injection Path</u> • OPEN the following: • BNHV8812A (A RHR P Rm) • EJHCV0606 (A RHR Hx Rm) • EJHV8809A (North Pipe Pen Rm, Pen 082) • CLOSE the following: • EJFCV0618 (A RHR Hx Rm) • EJHV8716A (A RHR Hx Rm) • <u>"B" RHR Cold Leg Injection Path</u> • OPEN the following: • BNHV8812B (B RHR P Rm) • EJHCV0607 (B RHR Hx Rm) • EJHV8809B (South Pipe Pen Rm, Pen 027) • CLOSE the following: • EJFCV0619 (B RHR Hx Rm) • EJHV8716B (B RHR Hx Rm)	a) OPEN the following to gravity feed RWST to RCS: • BN8717 (A RHR Hx Rm) <u>AND</u> • EJHV8840 (South Pipe Pen Rm, Pen 021)
b) OPEN BBV0233, Rx Vessel Head Vent (RB-2047 A02B-O Above Rx Head)	b) OPEN other available Vent paths: • BBV0085, Pressurizer Vent (RB-2081-D14N-O West Side by PZR) • As determined by Engineering

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.b (continued)	1.b (continued)

CAUTION: Fill and Spill can rapidly deplete RWST inventory. Action should be taken to refill the RWST from any available source.

- If it is desired to maintain the RCS saturated, flow should be limited to approximately 175 gpm.
- If it is desired to maintain the RCS subcooled, flow should be limited to approximately 1000 gpm

CAUTION: All strategies involving Fill and spill will result in contaminated water exiting through open SG manways. Appropriate access control and personnel protective actions must be implemented.

2) IF the RCS is NOT Intact, THEN ESTABLISH Fill and Spill:

a) ALIGN <u>One</u> of the following feed paths:

- <u>"A" RHR Cold Leg Injection Path</u>
  - OPEN the following:
    - BNHV8812A (A RHR P Rm)
    - EJHCV0606 (A RHR Hx Rm)
    - EJHV8809A (North Pipe Pen Rm, Pen 082)
  - CLOSE the following:
    - EJFCV0618 (A RHR Hx Rm)
    - EJHV8716A (A RHR Hx Rm)
- <u>"B" RHR Cold Leg Injection Path</u>
  - OPEN the following:
    - BNHV8812B (B RHR P Rm)
    - EJHCV0607 (B RHR Hx Rm)
    - EJHV8809B (South Pipe Pen Rm, Pen 027)
  - CLOSE the following:
    - EJFCV0619 (B RHR Hx Rm)
    - EJHV8716B (B RHR Hx Rm)

RESPONSE NOT OBTAINED:

a) OPEN the following to gravity feed RWST to RCS:

- BN8717 (A RHR Hx Rm)

<u>AND</u>

- EJHV8840 (South Pipe Pen Rm, Pen 021)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (continued)	1. (continued)
c. ESTABLISH Feed and Bleed or Fill and Spill using a CCP or SI pump.	c.
1) CCP A or CCP B available	1) RESTORE CCP incapable of injection per OSP-BG-00002, Verify One CCP Incapable of Injection Into RCS. Continue with Step 1.c.2 to Inject with SI Pumps while CCP is being restored.
a) LINE up a charging flow path as follows:	
b) SHIFT the CCP Suction to RWST OPEN: BNLCV112D, BNLCV112E CLOSE: BGLCV112B, BGLCV112C	
c) ESTABLISH one of the following flow paths by OPENING: • Normal Charging via BGHV8146 or • Alternate Charging via BGHV8147 or • Boron Injection Header Flowpath for the respective CCP "A" Train: EMHV8803A, EMHV8801A "B" Train: EMHV8803B, EMHV8801B	
d) START an available CCP • BG HIS-1A CCP A • BG HIS-2A CCP B	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. c (continued)	1. c (continued)

CAUTION: RCS Level increase will be rapid when using an SI Pump

ACTION/EXPECTED RESPONSE

2) SI Pump A or SI Pump B available

a) CHECK S/G Nozzle Dams - NOT INSTALLED

b) ALIGN an SI Pump for Cold Leg Injection as follows:

c) OPEN the following valves for the available train:

"A" Train	"B" Train
EMHV8835	EMHV8835
EMHV8821A	EMHV8821B
EMHV8923A	EMHV8923B
BNHV8806A	BNHV8806B
EMHV8814A	EMHV8814B
BNHV8813	BNHV8813

d) CLOSE the following valves:

"A" Train	"B" Train
EMHV8802A	EMHV8802B

e) START an available SI Pump

- EM HIS-4 – SI Pump A
- EM HIS-5 – SI Pump B

3) IF Establishing Feed and Bleed THEN OPEN RCS Vent Paths

- BBV0233, Rx Vessel Head Vent (RB-2047 A02B-O Above Rx Head)
- BBV0085, Pzr Vent (RB-2081-D14N-O West Side by PZR)
- Pressurizer PORVs
- As determined by Engineering

RESPONSE NOT OBTAINED

2) RESTORE SI pumps per OSP-EM-00002, Rendering SI Pumps Incapable of Injection.

a) ALIGN an SI Pump for Hot Leg Injection as follows:

OPEN the following valves for the available train:

"A" Train	"B" Train
EMHV8802A	EMHV8802B
EMHV8923A	EMHV8923B
BNHV8806A	BNHV8806B
EMHV8814A	EMHV8814B
BNHV8813	BNHV8813

CLOSE the following valves:

"A" Train	"B" Train
EMHV8835	EMHV8835
EMHV8821A	EMHV8821B

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
d. ESTABLISH and MAINTAIN desired RCS temperature and level by CONTROLLING Feed and Bleed or Fill and Spill as necessary.	
2. <u>MONITOR CTMT RADIATION, TEMPERATURE, AND HUMIDITY</u> • Radiation • GTRE0031 • GTRE0032 • GTRE0059 • GTRE0060 • SDRE0039, Seal Table Area • SDRE0040, Personnel Hatch • SDRE0041, Manipulator Crane • SDRE0042, 2047 SE Area • Temperature • GN TI-60, CTMT Cooler A Inlet • GN TI-61, CTMT Cooler B Inlet • GN TI-62, CTMT Cooler C Inlet • GN TI-63, CTMT Cooler D Inlet • GN TR-63, CTMT Cooler B Recorder • Humidity • GN AI-27 • GN AI-28	
3. <u>CHECK RHR STATUS</u>	3.
a. ATTEMPT to restore at least one train of RHR to service.	
b. At least one train of RHR available for service	b. Return To Step 1 of this Attachment.
c. Go To Step in effect.	

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Component Cooling Water	Group #	1		
	K/A #	000026 AA2.06		
	Importance Rating	2.8		
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				

Question # 6

Reactor Power is 40%.

- At 0858, CCW has been lost to the 'A' RCP motor.
- At 0900, 'A' RCP Motor Parameters are as follows:
  - o Motor Bearing Temperatures are 190°F and rising at 1°F/min.
  - o Motor Stator Winding Temperatures are 302°F and rising at 3°F/min.

(1) The FIRST 'A' RCP motor component to reach a temperature limit will be the …...?

And

(2) What is the LATEST time the 'A' RCP MUST be secured?

A. (1) Bearings
(2) 0903

B. (1) Bearings
(2) 0905

C. (1) Stator Windings
(2) 0903

D. (1) Stator Windings
(2) 0905

Answer: C

Explanation:

Per OTO-EG-00001, if CCW flow is lost to the RCPs, then the crew is directed to OTO-BB-00002 Specifically Attachment C for a loss of CCW to the RCP. The limits in Attachment C are as follows:

1. *Motor Bearing Temperatures – less than 195F*

2. *Motor Stator Winding temperature – less than 311F*
3. *Time lost less than 10 minutes*

And if either one of these is exceeded, C1 RNO applies which directs securing the affected RCP.

With the values and trends given, bearing temperatures can rise for 5 minutes, (195-190F)/ 1F/min, before the criteria to secure the RCP is met. Motor winding temperatures can rise for 3 minutes, (311-302F) / 3F/min, before the criteria to secure the RCP is met. Neither time exceeds the 10 minutes from the time of CCW lost to the RCPs (i.e. 0908 is not the limiting time to secure the 'A' RCP). Therefore, the RCP must be secured by 0903 to prevent damage to Stator Winding temperature trends.

A. Incorrect - wrong component
B. Incorrect - both are wrong
C. Correct – see above explanation
D. Incorrect – wrong time

Technical Reference(s):.
1. OTO-EG-00001, CCW System Malfunction, Rev 14
2. OTO-BB-00002, RCP Off-Normal, Rev 32

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #17, Loss of CCW, Objective C & E
C. DETERMINE the affect that a CCW System malfunction has on a cooled system/component and SELECT the subsequent action to respond to the associated malfunction.

E. Given a set of plant conditions or parameters indicating a CCW System Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ______N/A______

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(3)

Comments:

K/A match as the only component cooled by CCW that has a procedurally driven time to secure/ trip prior to damage are the RCP pumps/motors. The loss of CCW off normal procedure directs the operator to the RCP off normal procedure to prevent duplication of procedure steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C
(Page 1 of 4)
CCW To RCP Abnormal

NOTES

- Redundant indications should be used for Bearing and Stator Winding problems. A rise in RCP vibrations and increase in other RCP Bearing or oil temperatures on the same RCP are indicative of a CCW induced motor bearing problem. A change in RCP current may also occur with a Stator Winding issue.
- RCP computer values may be obtained using Plant Computer display "32BBG10" or "GD SG8" through "GD SG11".

C1. **CHECK RCP Motor Parameters meet all of the following:**

- Motor Bearing Temperatures - LESS THAN 195°F ON ALL RCPs
- Motor Stator Winding Temperatures - LESS THAN 311°F ON ALL RCPs
- CCW lost to RCP motors - LESS THAN 10 MINUTES

RESPONSE NOT OBTAINED:

IF Reactor power is greater than or equal to 48% (P-8 lit),
THEN PERFORM the following:

a. TRIP the affected RCP per Attachment D, RCP AND Reactor Trip.

b. IF CCW Train Supplying the RCPs is greater than 105°F,
THEN Slowly REDUCE CCW temperature using OTN-EG-00001, Component Cooling Water:

- EG TI-31
- EG TI-32

IF Reactor power is less than 48% (P-8 extinguished),
THEN PERFORM the following:

a. IF Reactor power is greater than or equal to 10% (P-7) AND more than one RCP is affected
THEN TRIP the affected RCP(s) per Attachment D, RCP AND Reactor Trip.

(Step C1. continued on next page)

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Pressurizer Pressure Control System Malfunction / 3	Group #	1		
	K/A #	00027 A2.10		
	Importance Rating	3.3		
Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunction: PZR Heater energized/ de energized condition.				

Question # 7

Reactor Power is 100%.

The pressurizer pressure control system is in the following alignment:

- o Pressure control selected to PT-455/456.
- o Variable Heaters' handswitch is in the CLOSE position.
- o Backup Heater Group A's handswitch is in the CLOSE position.
- o Backup Heater Group B's handswitch is in the AUTO position.

Then, Pressurizer Pressure Channel, PT-455, begins to fail HIGH. It currently indicates 2310 psig and RISING.

All other Pressurizer Pressure channels are 2170 psig and LOWERING. (No operator action has been taken.)

(1) What is the status condition of pressurizer variable heaters?

And

(2) What is the status condition of pressurizer backup heaters?

A. (1) Variable heaters are energized at MINIMUM amperage.
   (2) All backup heaters are de-energized.

B. (1) Variable heaters are energized at MINIMUM amperage.
   (2) Backup heater group A is energized. Backup heater group B is de-energized.

C. (1) Variable heaters are energized at MAXIMUM amperage.
   (2) All backup heaters are de-energized.

D. (1) Variable heaters are energized at MAXIMUM amperage.
   (2) Backup heater group A is energized. Backup heater group B is de-energized.

Answer: B

Explanation:

Utilizing Attachment #1 of OTN-BB-00005, as the selected controlling PZR level channel goes up (regardless of actual RCS pressure) the Variable heaters will go to minimum voltage and the backup heater that are in Auto will deenergize. This means that the A Backup Heaters will be ON with B Backup Heaters deenergized. The variable heaters will be at MINIMUM amperage.

A. Incorrect – part 2 of the question is incorrect. Plausible if the student believes that controlling RCS pressure channel will solely drive the PZR Backup Heaters and does not apply the stem data correctly understanding that Backup Group A HS in Close means they will be energized regardless of the signal from the controlling channel (i.e 455, the "upper channel" directs PZR heater automatic operation regardless of handswitch position)
B. Correct – see above explanation
C. Incorrect – both are wrong, see other explanations
D. Incorrect – part 1 of the question is incorrect. Plausible if the student falsely believes that the different PZR heaters are controlled by different PZR Pressure Channels or does not understand the Selection on the Pressure control handswitch. Example: Candidate believes that BB PT 456 (the 'lower channel' selected on the control handswitch) controls the variable heater group, the variable heaters would be energized at maximum voltage as this pressure channel is below the value where Variable heaters should be at a MAXIMUM. Furthermore, 2 or the 3 pressure channels are at a pressure where the variable channels would be energized at a Maximum amperage making this a plausible distractor.

Technical Reference(s):
1. OTO-BB-00006, Pressurizer Pressure Control Malfunctions, Rev 20
2. OTN-BB-00005, Pressurizer and Pressurizer Pressure Control, Rev 14

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Procedures, LP #41, OTO-BB-00006, Objective C - Given a set of plant conditions or parameters indicating a Pressurizer Pressure Control Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # _X L16210_____
Modified Bank # ______
New _______

Question History: Last NRC Exam __2005__________

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(5)

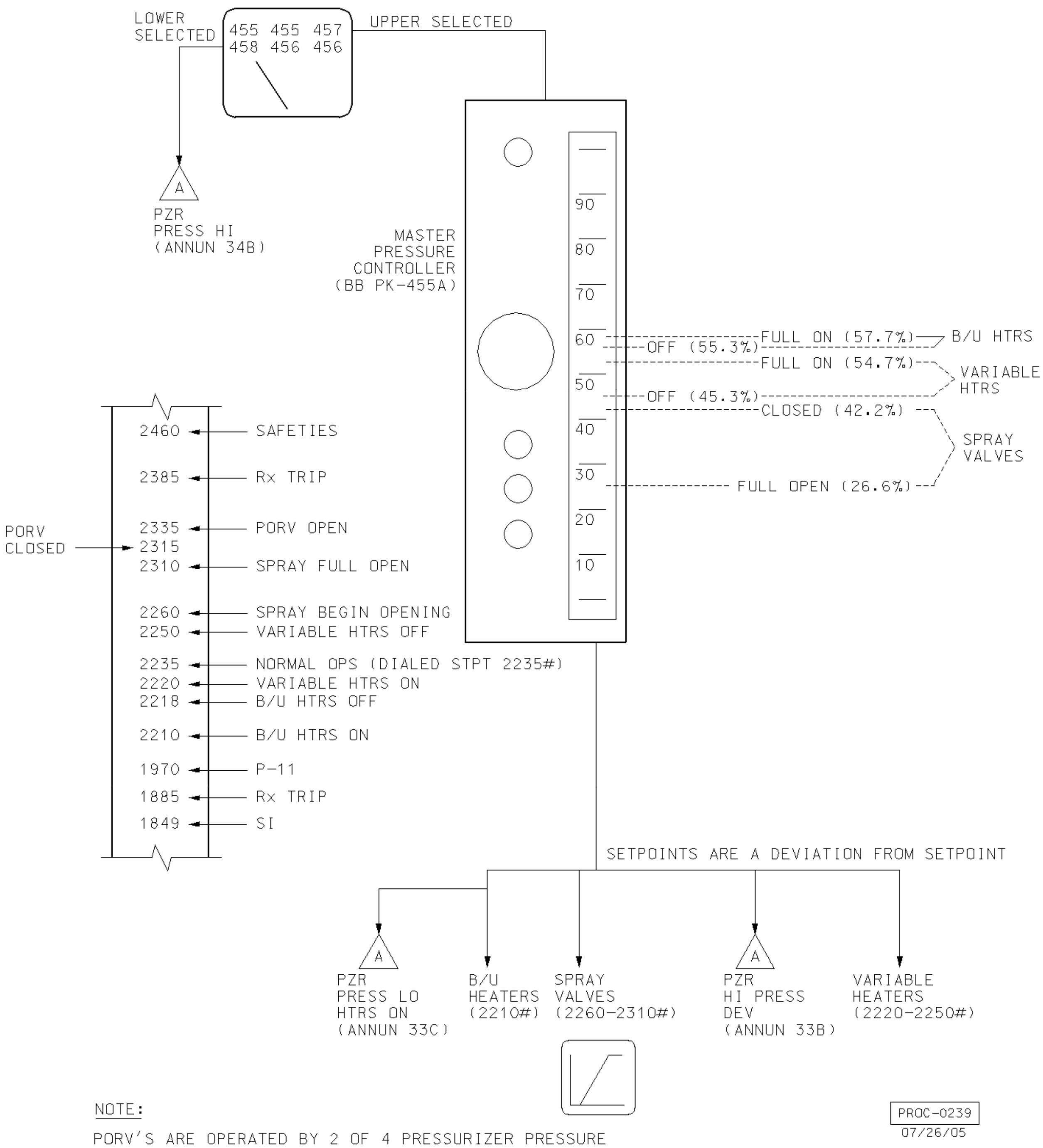
Comments:

Revised question per NRC comment. Made question a 2x2 format based on comment.

Attachment 1

Master Pressure Controller

Sheet 1 of 1



NOTE:

PORV'S ARE OPERATED BY 2 OF 4 PRESSURIZER PRESSURE NARROW RANGE INSTRUMENTS OR COMS SETPOINT IF ARMED.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
ATWS / 1	Group #	1		
	K/A #	00029 EK3.12		
	Importance Rating	4.4		
Knowledge of the reasons for the following responses as the apply to the ATWS: Actions contained in EOP for ATWS				

Question # 8

During an ATWS, Reactor Coolant Pumps should NOT be tripped when reactor power is GREATER THAN a MINIMUM of ______(1)_____ because the PRIMARY reason the RCPs are required is to ______(2)______.

A. (1) 10^{-8} amps (POAH)
(2) maintain core cooling

B. (1) 10^{-8} amps (POAH)
(2) maintain PZR Pressure control

C. (1) 5%
(2) maintain core cooling

D. (1) 5%
(2) maintain PZR Pressure control

Answer: C

Explanation:

Per the FR-S.1 basis document for Caution 1, "During an ATWS, RCP operation could be beneficial by temporarily cooling the core under voided RCS conditions. If reactor power is greater than 5%, the RCPs should not be tripped even if all normal running conditions are not satisfied. Manually tripping the RCPs during some ATWS events could result in reduced heat removal and a challenge to fuel integrity. An ATWS is not a design basis event; therefore the licensing requirement to trip the RCPs within a timely manner to remain within the small-break LOCA design basis is not applicable."

10-8 amps (POAH) is plausible because this is the point at which the fission process stops adding fission heat into the RCS (by definition) and the candidate may be believe that RCPs can not be secured until below this level during an AWTS situation.

"Maintain PZR Pressure control to prevent PZR PORV(s) from opening" is plausible as Step #4.c checks PZR Pressure is less than 2335 PSIG and if NOT the procedure directs Checking PZR PORVS and Block Valves open. The candidate may falsely believe that the PRIMARY reason RCPs are left in service to maintain normal PZR spray available such that PZR Pressure remains below the PORV lift setpoints to ensure RCS inventory is not discharged to the PRT during an ATWS situation.

A. Incorrect – wrong power level
B. Incorrect – both are wrong
C. Correct – See above explanation
D. Incorrect – wrong reason

Technical Reference(s):
1. FR-S.1, Response to Nuclear Power Generation / ATWS, Rev 10
2. BD- FR-S.1, Response to Nuclear Power Generation / ATWS basis document, Rev 4

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #29, FR-S.1, Objective K:

K. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of:
1. FR-S.1, Response to Nuclear Power Generation/ATWS.

Question Source: Bank # _____
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(10)

Comments:

Replaced question per NRC comments

BD-FR-S.1

RESPONSE TO NUCLEAR POWER GENERATION/ATWS

Revision 004

EOP STEP: 1-CAUTION 1

WOG ERG STEP: 1-CAUTION 1

STEP:

RCPs should NOT be tripped with reactor power greater than 5%.

PURPOSE:

To inform the operator that the RCPs should not be tripped even if all normal running conditions are not satisfied.

BASIS:

During an ATWS, RCP operation could be beneficial by temporarily cooling the core under voided RCS conditions. If reactor power is greater than 5%, the RCPs should not be tripped even if all normal running conditions are not satisfied. Manually tripping the RCPs during some ATWS events could result in reduced heat removal and a challenge to fuel integrity. An ATWS is not a design basis event; therefore the licensing requirement to trip the RCPs within a timely manner to remain within the small-break LOCA design basis is not applicable.

KNOWLEDGE:

This caution is applicable during the performance of the Immediate Action Steps and should be known by the operator without availability of the written procedure.

DEVIATIONS:

N/A

REFERENCES:

Rev 2 DW-99-060

COMN 3196 (ATWS response procedures based on WOG FRGs)

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Steam Gen. Tube Rupture / 3	Group #	1		
	K/A #	00038 EK3.01		
	Importance Rating	4.1		
Knowledge of the reasons for the following responses as the apply to the SGTR: Equalizing pressure on primary and secondary sides of ruptured S/G				

Question # 9

Per procedure E-3, Steam Generator Tube Rupture, what is the PRIMARY reason for reducing RCS pressure to match ruptured SG pressure?

A. To minimize the probability of a Pressurized Thermal Shock event.

B. To restore RCS inventory and stop break flow prior to stopping ECCS pumps.

C. To eliminate concern for SG overfill and damage to secondary side steam piping.

D. To ensure there will be NO release of radioactivity through the SG Atmospheric Dump valves for the duration of the SGTR.

Answer: B

Explanation:
Step #16 of E-3 directs the depressurization of the RCS to minimize break flow and refill the PZR. Per the basis document, the purpose of the step is to "To lower RCS pressure to stop primary-to-secondary leakage and establish an indicated pressurizer level."

A. Incorrect – this is not the primary concern as PTS is only a concern if ruptured SG pressure is low. (aka a ruptured and faulted SG that is not isolated).
B. Correct
C. Incorrect – This is not the primary concern and may not be able to stop overfill if the tube rupture is large enough but plausible as secondary side damage may occur.
D. Incorrect – this is the basis of step #3 (isolating flow from ruptured SG) in which the ASD controllers are adjusted upward to 1160 psig. Plausible as this an outcome of equalizing pressure but not the primary reason and is also wrong it indications that there is "no release". This release would be minimized.

Technical Reference(s):
1. E-3 Steam Generator Tube Rupture, Rev 17
2. BD-E-3, Steam Generator Tube Rupture basis document, Rev 8

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #17 E-3; Objective A and J:

A: EXPLAIN the Purpose and Major Action Categories of E-3, Steam Generator Tube Rupture

J: STATE the conditions necessary to secure RCS depressurization to minimize break flow and refill the Pressurizer.

Question Source: Bank # __X L16296____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2009 Audit Exam_________

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(10)

Comments:

Revised question per NRC comments – reworded stem and removed "when RCS cooldown is commenced" on the distractor for PTS. Reordered distractor and answer based on length after NRC comments incorporated.

EOP STEP: 16 WOG ERG STEP: 16

STEP:

DEPRESSURIZE RCS To Minimize Break Flow And Refill PZR:

PURPOSE:

To lower RCS pressure to stop primary-to-secondary leakage and establish an indicated pressurizer level.

BASIS:

After the cooldown is completed, ECCS flow will pressurize the RCS to an equilibrium condition where break flow equals ECCS flow. The equilibrium pressure will be somewhere between the ruptured steam generator pressure and the shutoff head of the ECCS pumps and rises with SI capacity, as shown in the E-3 ERG Background Document Figure 26. A major objective of the E-3 procedure is to bring the plant from point A to point B where primary-to-secondary leakage will be stopped. However the path one takes is important.

The illustrated curve in the E-3 ERG Background Document represents equilibrium conditions where ECCS flow and break flow are equal. Hence, for points on the curve, reactor coolant inventory is constant. To the left of this curve RCS pressure is greater than equilibrium so that break flow exceeds ECCS flow. Therefore, in this region coolant inventory is lowering. Conversely, to the right of the curve, ECCS flow exceeds break flow so that coolant inventory is rising. The ideal path from point A to point B should raise coolant inventory and restore pressurizer level. Hence, the ideal path (see E-3 ERG Background Document Figure 27) requires a depressurization of the RCS.

In some cases, pressurizer level may approach the upper tap (top of the indicating range) before RCS pressure is reduced to the ruptured steam generator pressure. This may be a symptom of a smaller tube failure, voiding in the upper head during natural circulation conditions, injection of the SI accumulators, or ineffectiveness of the depressurization method. In that case, the preferred path from point A to point B is demonstrated in the E-3 Background Document Figure 28. Depressurization of the RCS is terminated on high pressurizer level to prevent filling the pressurizer and loss of pressurizer pressure control. Following SI termination, pressurizer level lowers which further reduces RCS pressure to equilibrium with the ruptured steam generator. In some cases, such as a small tube failure in a high pressure SI plant, the pressurizer may be sufficiently full such that no depressurization of the RCS is necessary prior to SI termination.

(Continued on next page)

EOP STEP: 16 WOG ERG STEP: 16

On the other hand, for multiple tube failures or reduced SI capacity for a smaller tube failure, it may be necessary to lower RCS pressure below that of the ruptured steam generator pressure in order to restore pressurizer level. This path is shown in E-3 ERG Background Document Figure 29. In that case reverse flow, i.e., secondary to primary leakage, will supplement ECCS flow to restore pressurizer level. If pressure continued to be reduced to saturation, voiding in the primary system may result in an unreliable pressurizer level indication and delay SI termination. To avoid this, depressurization of the RCS is terminated if minimum RCS subcooling is reached.

With PZR spray (normal and auxiliary) stopped, both pressurizer pressure and level should rise toward equilibrium conditions. If level continues to rise without a corresponding rise in pressure, leakage from the spray valves should be suspected. If this persists until filling of the pressurizer is imminent, appropriate measures to stop the leakage, such as stopping RCPs as necessary to terminate spray flow or isolating the auxiliary spray line should be performed. It may be necessary to stop two (or more) RCPs to terminate spray flow, depending on which spray valve is failed open and the existing pressurizer level. Spray effectiveness with different combinations of RCPs running will vary with plant design as discussed in the Plant-Specific Information Section of the E-3 ERG Background Document. Depressurization of the RCS due to leakage from the spray valves will stop once the pressurizer fills with water. Therefore, this condition should not prevent or delay termination of ECCS flow in subsequent steps when all the necessary criteria are satisfied.

The preferred means of RCS depressurization is normal PZR spray since this does not result in a loss of reactor coolant. If normal spray is not available, an alternative means of depressurizing the RCS, such as a pressurizer PORV or auxiliary spray must be used. However, the use of a PORV will result in an additional loss of reactor coolant which may rupture the PRT and lead to abnormal containment conditions. On the other hand auxiliary spray may cause excessive thermal stresses in the spray nozzle and may not be sufficient to rapidly lower RCS pressure. For these reasons, it is used only if normal spray and all pressurizer PORVs are unavailable.

(Continued on next page)

EOP STEP: 16 WOG ERG STEP: 16

KNOWLEDGE:

Maximum spray flow should be established to lower primary system pressure as rapidly as possible. The operator should be familiar with how rapidly pressure will lower with full spray to avoid overshooting the termination criteria. In addition, if pressure does not lower or lowers only slowly, the operator should proceed to the next step to select an alternative means of depressurizing the RCS to expedite recovery.

Voiding in the upper head region is not expected to occur if the reactor coolant pumps are running even with full spray flow. However, if the RCS is depressurized concurrently with the cooldown some voiding may occur. In that case, pressurizer level will rise rapidly as water is displaced from the upper head into the pressurizer.

If a subsequent SGTR is diagnosed by the operator while the RCS depressurization is in progress, although it does not impact the pressure in the newest ruptured steam generator, for the sake of simplicity it should be stopped and the plant stabilized by the operator until the newest ruptured steam generator is isolated.

High PZR water level with any combination of RCPs operating will raise spray effectiveness.

DEVIATIONS:

Added plant specific RNO substep b. to enhance procedure usage and meet the ERG intent by transitioning to the next high level step if normal PZR spray is not available (or effective).

Added plant specific means for RNO substep c.1) as required by the ERG.

Added plant specific actions for RNO substep c.2) to enhance procedure usage and assist the operator in meeting the ERG intent for isolating the auxiliary spray line.

Deviated from PWROG guidance due to Simulation validation not supporting PWROG recommendations. Did not add Steps as requested in DW-04-009 and DW-10-017 which established an initial depressurization, then allowed ECCS termination. However, when the subsequent depressurization was completed to allow RCS pressure to drop below ruptured steam generator pressure, the subcooling margin of 30°F was lost. This complicates the scenario by requiring operators to transition to ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Required. This is an undesirable condition for the given conditions.

(Continued on next page)

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Steam Line Rupture - Excessive Heat Transfer / 4	Group #	1		
	K/A #	000040 W/E12 EA1.3		
	Importance Rating	3.4		
Ability to operate and / or monitor the following as they apply to the (Uncontrolled Depressurization of all Steam Generators): Desired operating results during abnormal and emergency situations.				

Question # 10

ECA-2.1, “Uncontrolled Depressurization of All Steam Generators,” is being performed. Efforts to isolate the leak have been unsuccessful.

(1) Per procedure ECA-2.1, how will the operator minimize the RCS Cooldown Rate?

And

(2) Per procedure ECA-2.1, what will the reactor operator monitor to verify the desired operating results have been achieved?

A. (1) Terminate SI flow and establish normal charging
(2) Core Exit Thermocouples

B. (1) Terminate SI flow and establish normal charging
(2) RCS Hot Leg temperatures

C. (1) Lower auxiliary feed flow to 27,500 lbm/hr to each S/G
(2) Core Exit Thermocouples

D. (1) Lower auxiliary feed flow to 27,500 lbm/hr to each S/G
(2) RCS Hot Leg temperatures

Answer: D

Explanation:

Per the stem of the question, step #1 of ECA-2.1 has been unsuccessful and the operators will be performing step #2. Step #2 RNO directs the operator to lower AFW flow to 27500 lbm/hr to each SG. The operator is then directed to step 2.c to check RCS hot leg temperatures are stable or lowering. This is the desired response that a cooldown is still in progress but not violating the

100F/hr limits and it is achieved by controlling AFW flow and monitoring RCS hot leg temperatures.

The distractor of terminating SI flow is from step #10-15 of the ECA-2.1 and is plausible if the operator believes that the action to secure the injection of cold ECCS water into the RCS will minimize the cooldown vice minimizing the S/G depressurization.

The distractor of Core Exit Thermocouples is plausible as CETs trends are used in several EOP procedures to check and see if Core Exit thermocouples are "stable or lowering". Example ECA 1.3 Step #12 b, ECA-3.2 step #20, and E-3 step #6c through step #6.e during a RCS Cooldown as the temperature to stop the cooldown.

A. Incorrect – both are wrong
B. Incorrect - wrong action
C. Incorrect - wrong parameter used to monitor
D. Correct

Technical Reference(s):

1. ECA-2.1, Uncontrolled Depressurization of all S/Gs, Rev 13
2. BD-ECA-2.1, Uncontrolled Depressurization of all S/Gs basis document, Rev 5

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #16; ECA-2.1, Objective I: OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ECA-2.1, Uncontrolled Depressurization of all Steam Generators.

Question Source: Bank # ____
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR: 55.41(b)(10)

Comments:

Revised stem of part 1 and replaced 'RCS Pressure' with 'Core Exit Thermocouples' for part 2 of question per NRC Comments.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

A minimum feed flow of 27,500 lbm/Hr must be maintained to each SG with a narrow range level less than 7% [25%].

NOTE

Shutdown margin should be monitored during RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**CONTROL Feed Flow To Minimize RCS Cooldown:**	
	a. CHECK cooldown rate in RCS cold legs — LESS THAN 100°F/HR	a. PERFORM the following: 1) LOWER feed flow to 27,500 lbm/Hr to each SG. 2) Go To Step 2.c.
	b. CHECK narrow range level in all SGs — LESS THAN 52%	b. CONTROL feed flow to maintain narrow range level less than 52% in all SGs.
	c. CHECK RCS hot leg temperatures — STABLE OR LOWERING	c. CONTROL feed flow or DUMP steam to stabilize RCS hot leg temperatures.

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	1		
Loss of Main Feedwater / 4	**Group #**	1		
	K/A #	00054 AA2.02		
	Importance Rating	4.1		
Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Differentiation between loss of all MFW and trip of one MFW pump				

Question # 11

Reactor Power is 100%.

A spurious "A" Train Safety Injection occurs.

What is the status of the Main Feedwater Pumps?

A. BOTH Main Feedwater Pumps are tripped.

B. BOTH Main Feedwater Pumps are running at maximum speed.

C. BOTH Main Feedwater Pumps are running at minimum speed.

D. "A" Main Feedwater Pump is tripped; "B" Main Feedwater Pump is running.

Answer: A

Explanation:
A SI signal (either train) will cause a FWIS and that will trip both Main Feedwater Pumps (MFP) and close Feedwater isolation valves creating a loss of main feedwater. A SI signal is also a direct trip of both MFPs. A common misconception is that these may be train specific i.e. A SI trips A MFP and one of the reasons that the distractors are plausible.

The signals that cause both MFP turbines to trip are:

- *FWIS*
- *Level Hi-Hi in any SG water level (2 of the 4 narrow range level detectors at 91%).*
- *All condensate pumps trip*
- *Safety Injection Signal*
- *High feed pump discharge pressure at 1900 psig,*

Note: a SIS will cause a FWIS

The signals that cause only the affected (single) MFP turbine to trip are:

- *Turbine Overspeed*
- *Exhaust Vacuum Low*

- *Low Lube oil pressure to respective feedwater Turbine bearings*
- *Low Lube oil pressure to respective feedwater Pump bearings*
- *Thrust Bearing Wear*

A. Correct – a SI signal from either train will trip both MFPs.
B. Incorrect – Plausible if the candidate correctly processes that the reactor trips due to the spurious SI signal but incorrectly believes that the MFPs will be available to restore SG levels after the reactor trip and then subsequent turbine trip.
C. Incorrect - Plausible if the candidate believes that the FWIS are closed due to these signals but the MFPs are not tripped by these signals. In this condition, the flowpath to the SGs would be blocked and MFPs would be running at minimum recirculating back to the condenser.
D. Incorrect – Plausible if the candidate believes that these signals are only train specific as discussed above

Technical Reference(s):
1. OTA-RK-00026, Addendum 120A, Main Feedwater Pump A Trip, Rev 5,
2. OTO-SA-00001, EFSAS Verification and Restoration, Rev 39

References to be provided to applicants during examination: None

Learning Objective: T61.0110.6 / Systems, LP #23 – Main Feedwater, Objective C & D:

C. STATE the conditions, including the setpoints and coincidences, that will cause a FWIS and EXPLAIN the system response to the signal.

D. LIST the conditions that will trip either a single main feed pump or both main feed pumps and EXPLAIN the system response for each case.

Question Source: Bank # __X L16231____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2005________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(4)

Comments:

k/a match as the question provides the operator with a set of plant conditions and makes the operator determine that a loss of all MFW has occurred (i.e both main feedwater pumps trip). The ability to differentiate between a loss of one of both main feedwater pumps is examined based on a misconception that the MFW pump trip is train specific i.e. A SI only gets A MFW pump.

Main Feedwater Pump A Trip			120A
Initiating Device: 1. See below	**Setpoint:** 1. See below	**Reset:** 1. See below	**MFP A TRIP**

1.0 AUTOMATIC ACTIONS:

1.1. MFP A trips

1.2. AEHV0016, MFP A DISCH ISO HV, closes.

2.0 IMMEDIATE ACTIONS:

2.1. None

3.0 OPERATOR ACTIONS:

NOTE:
If this was the only Main Feed Pump in operation or if both Main Feed Pumps tripped, an AFAS (MD) will occur.

Both MFPs will trip if:

- Hi Hi SG Level 91 % (2/4 NR on 1/4 SGs)
- SIS
- Condensate Pump Trip (All)
- Hi Discharge Press (1900 psig 2/3)
- FWIS

3.1. ENSURE the following:

- FC HIS-18, MFP TURB A, TRIP light LIT
- AE ZL-16, MFP A DISCHARGE ISOLATION VALVE, CLOSED
- FC ZL-5A, MFP A HIGH PRESSURE STOP VALVE, CLOSED light LIT
- FC ZL-9A, MFP A LOW PRESSURE STOP VALVE, CLOSED light LIT

3.2. IF the Main Generator is synchronized to the grid, Go To OTO-AE-00001, Feedwater System Malfunction.

OPERATOR ACTIONS (Cont'd):

3.3. IF the Main Generator is NOT synchronized to the grid, PERFORM the following:

3.3.1. DEPRESS Main Turbine, CLOSE VALVES pushbutton.

3.3.2. STABILIZE power at less than 2% by performing any combination of the following:

- ENSURE AB PK-507, STEAM HDR PRESS CTRL set to 1092 psig (7.28 pot setting).
- INSERT Control Rods to stabilize RCS temperature at no load Tavg.
- BORATE the RCS to reduce RCS temperature to no load Tavg.
- TERMINATE CHEST/SHELL WARMING per OTN-AC-00001, Main Turbine and Generator Systems.

3.3.3. IF the Plant cannot be stabilized, TRIP the Reactor and Go To E-0, Reactor Trip or Safety Injection.

3.3.4. IF Aux Feed is the only available source of feed to the Steam Generators, SHUT DOWN the Reactor per OTG-ZZ-00005, Plant Shutdown 20% Power to Hot Standby.

3.4. COMPLETE shutdown of MFP A per OTN-AE-00001, Feedwater System.

Initiating Devices	Actuation Type	Setpoint	Reset
FCPSL0022A FCPSL0022B	MFP A discharge valve closure and MFP A trip alarm annunicator	2/2 at 75 psig	83 psig
FCPSL0070A FCPSL0070B FCPSL0070C	MFP Pump Bearing Oil Pressure Trip	2/3 at 4 psig	7 psig
FCPSL0062A FCPSL0062B FCPSL0062C	MFP Turb Bearing Oil Pressure Trip	2/3 at 4 psig	7 psig
FCPS0077A FCPS0077B	Exhaust Case Vacuum Low Trip	2/2 14.32 inHg 15.32 inHg	20.82 inHg 21.82 inHg
FCZE0394A FCZE0398A	MFP Turbine Thrust Bearing Wear	2/2 Both +30 mills OR Both -40 mills	
SSPU-1 SSPU-2	Overspeed	(Elec 5880 RPM) (Mech 6098 to 6222 RPM)	

4.0 SUPPLEMENTAL INFORMATION:

4.1. M-22FC03, Auxiliary Turbines S.G.F.P. Turbine "A" P&ID

4.2. J-22FC06, Control Logic Diagram Auxiliary Turbines SGFP Pressure Alarms, Turbine Trips

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Off-site Power	Group #	1		
	K/A #	000056 AA1.08		
	Importance Rating	2.5		
Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power: HVAC chill water pump and unit				

Question # 12

A Loss of Off-Site Power has occurred.

The Reactor Operator should observe the Control Room A/C units (SGK04A/B) starting at ___________ seconds after the bus is reenergized by the emergency diesel generators.

A. 10

B. 15

C. 25

D. 30

Answer: D

Explanation:
With a shutdown Sequencer actuation (bus undervoltage and EDG start and closure back onto the bus after a bus load shed have occurred), the Control Room A/C unit will start at 30 seconds.

A. Incorrect – this is when the backup CCW pump would start if the normal CCW pump fails to start at the 5 second time.
B. Incorrect – this is when the Containment spray pumps would start.
C. Incorrect – this is when the ESW pumps start
D. Correct

Technical Reference(s):
1. E-22NF01, Load Shedding and Emergency Load Sequencing Logic, Rev 8

References to be provided to applicants during examination: None

Learning Objective: T61.0110, systems, LP #60, Secondary Ventilation Systems, Objective D: DESCRIBE the purpose and operation of the following Control Building Ventilation System components:

4. Control Room Air Conditioning (A/C) System
    a. Control Room Filtration System Absorber Unit
    b. Filtration Fan
    c. Control Room A/C Unit

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

k/a match as the control room A/C units are the "HVAC chill water pump and unit" in the K/A regardless of wither it has a chill water pump or is Freon based refrigerant cycle.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Station Blackout / 6	Group #	1		
	K/A #	00055 EA1.02		
	Importance Rating	4.3		
Ability to operate and monitor the following as they apply to a Station Blackout: Manual ED/G start				

Question # 13

A Station Blackout has occurred.

- Offsite power is unavailable.
- 4.16 kV Bus, NB02, has a bus lockout.
- The 'A' EDG, NE01, will not start from the control room.
- OTs are attempting to locally start the 'A' EDG, NE01, iaw EOP Addendum 21, Local Start of Emergency DGs.

(1) The Secondary OT will FIRST attempt to locally start NE01 by …?

And

(2) When NE01 is started, the operator should observe field flashing at a MAXIMUM of what engine speed?

A. (1) Breaking the glass on DG NE01 emergency start pop-out button KJ-HS-1D, (NE107)
(2) 300 RPM

B. (1) Breaking the glass on DG NE01 emergency start pop-out button KJ-HS-1D, (NE107)
(2) 471 RPM

C. (1) Placing the master transfer switch in LOC/MAN and press and hold local start pushbutton, KJ-HS-1C
(2) 300 RPM

D. (1) Placing the master transfer switch in LOC/MAN and press and hold local start pushbutton, KJ-HS-1C
(2) 471 RPM

Answer: B

Explanation:

ECA-0.0 Step #7 RNO action is directs EOP Addendum 21. Per EOP 21, Attachment A step A.3 "Break glass on DG NE01 Emergency Start Pop-out button KJ-HS-1D (NE107)" and is the first action performed. The second method (if the first is not successful) is to "Place the Master transfer switch in LOC/MAN and press and hold local start pushbutton KJ-HS-1C"

Per OSP-NE-00001A page 16 "At approximately 471 rpm, the following windows will alarm, then reset when the generator field is flashed:

- *MCB Annunciator 20B, DG NE01 UV OR UF*
- *KJ121 Annunciator 7B, GENERATOR UNDER VOLTAGE*
- *KJ121 Annunciator 7C, GENERATOR UNDER-EXCITED*

300 rpm is plausible if the candidate falsely believes that this situation will be simulator to a slow start as described in 6.1.10 of OSP-NE-0001A. Per OSP-NE-00001, the operator starts NE01 in step 6.1.9 and the engine ramps up to 300 rpms at which point the operator is required to record several reading and start times before ramping the speed up to 510 rpm which makes it plausible distractor.

A. Incorrect – see above explanation
B. Correct – see above explanation
C. Incorrect – see above explanation
D. Incorrect – see above explanation

Technical Reference(s):

1. ECA-0.0, Loss of all AC Power, Rev 22
2. EOP Addendum 21, Local Start of Emergency DGs, Rev 2
3. OSP-NE-0001A, Standby Diesel Generator A Periodic Tests, Rev 62

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #22, ECA-0.0, Objective I: DESCRIBE in order the methods used to locally start a Diesel Generator per ECA 0.0. (EOP Addendum 21)

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam __N/A____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. DETERMINE And CORRECT Cause Of DG Failure(s):

a. USE applicable Annunciator Response Procedure(s):

- OTA-KJ-00121, Diesel Generator NE01 Control Panel
- OTA-KJ-00122, Diesel Generator NE02 Control Panel

b. CHECK all of the following Engine Shutdown Relay actuation annunciators for affected DG(s) - CLEAR

- 2A, Lube Oil Pressure Low
- 4C, Jacket Water Temperature High
- 6C, Engine Overspeed
- 6D, Engine Shutdown Trouble
- 6F, Crankcase Pressure High
- 7D, Generator Protection Relay

RESPONSE NOT OBTAINED:

b. PERFORM the following for affected DG(s):

1) EVALUATE reason for Engine Shutdown Relay actuation.

2) CONSULT with Control Room prior to resetting any relays to prevent potential damage to DG.

3) WHEN directed, THEN AUTO START DG as follows:

a) ENSURE Differential OC Lockout Relay 186-1 is RESET:

- Panel NE107 (NE01)
- Panel NE106 (NE02)

b) ENSURE Mechanical Overspeed device [silver knob] is RESET [PULLED OUT]. (Plant South of DG)

c) ENSURE DG Master Transfer Switch is in AUTO.

d) RESET Engine Shutdown Relay:

- KJ-HS-12 (NE107)
- KJ-HS-112 (NE106)

e) **Go To Step 2.**

(Step 1. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Step 1.	(continued from previous page)	
	c. CHECK annunciator 6A, Diesel Start Failure for affected DG(s) - CLEAR	c. PERFORM the following for affected DG(s): 1) EVALUATE reason for Diesel Start Failure. 2) CONSULT with Control Room prior to resetting any relays to prevent potential damage to DG. 3) WHEN directed, THEN AUTO START DG as follows: a) ENSURE DG Master Transfer Switch is in AUTO. b) RESET Engine Shutdown Relay to auto start DG: • KJ-HS-12 (NE107) • KJ-HS-112 (NE106)
2.	**CHECK Affected DG(s) - RUNNING**	Locally START affected DG(s) using the following as necessary: • Attachment A, Local Start Of DG NE01. • Attachment B, Local Start Of DG NE02.
3.	**NOTIFY Control Room Of DG(s) Status**	
4.	**CONTINUE Efforts To Restore Both DGs As Necessary**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 1 of 3)
Local Start Of DG NE01

A1.	**NOTIFY Control Room Of Intent To Energize NB01 By Locally Starting DG NE01**	
A2.	**PERFORM Walkdown Of DG NE01 To Ensure NO Obvious Adverse Conditions Exist Prior To Starting DG**	
A3.	**BREAK Glass On DG NE01 Emergency Start Pop-out Button KJ-HS-1D (NE107)**	
A4.	**CHECK If DG NE01 - STILL STOPPED**	IF DG NE01 has started, THEN Go To Step A12.
A5.	**PLACE Master Transfer Switch KJ-HS-9 in LOC/MAN**	
A6.	**PRESS And HOLD Local Start Pushbutton KJ-HS-1C**	
A7.	**CHECK If DG NE01 - STILL STOPPED**	IF DG NE01 has started, THEN PERFORM the following: a. PLACE Master Transfer Switch KJ-HS-9 in AUTO. b. Go To Step A12.

6.1.8. ENSURE the diesel room Operations Technician(s) are prepared to perform the following actions when the diesel starts:

- MONITOR for obvious air leakage in air start system
- IF leakage is identified in the air start system, NOTIFY Control Room. [Ref: 8.2.2]
- PREPARE to record starting air tank pressure immediately after diesel start.

CAUTION

During DG start which causes high demand for instrument air, excess flow check valves KJV0742A and KJV0743A can seat causing loss of related instrumentation such as KJPI0026. These valves are reset by Step 6.1.14.

The lube oil low-pressure alarm is bypassed below 471 rpm. The diesel shall be shut down if oil pressure is less than 70 psig.

- At reduced speed, periodically MONITOR KJPI0026, DG A MN L-O STR OUT PRESS IND.
- STOP the diesel IF KJPI0026 is less than 70 psig.

NOTE

During the starting sequence, the following KJ121 alarms may alarm, then reset when the diesel is at rated voltage and frequency:

- Annunciator 7D, GENERATOR PROTECTIVE RELAY
- Annunciator 7C, GENERATOR UNDER-EXCITED
- Annunciator 7A, GENERATOR UNDER FREQ

6.1.9. At RL015, using KJ HS-8A, DG NE01 START-RESET/STOP, START Diesel Generator A.

6.1.10. Using NE SI-1, DG FREQ/SPD:

a. CHECK NE01 starts

b. CHECK NE01 accelerates to approximately 300 rpm

6.1.11. RECORD diesel start time on Attachment 6, MCB A Diesel Data Sheet

6.1.12. RECORD diesel start time on Control Room Log.

6.1.13. Immediately after diesel start, RECORD the following starting air pressures, on Attachment 7, Local A Diesel Data Sheet. [Ref: 8.2.2]

- KJPI0003A, DG STARTING AIR TK A PRESS IND
- KJPI0003B, DG STARTING AIR TK B PRESS IND

6.1.14. At East side of DG A Room, below Fuel Filters, PERFORM the following to equalize air pressure:

a. OPEN the following valves approximately 1/4 turn to equalize pressure:

- KJV0740A, DG A STARTING AIR TO PRESS INST'N KJV0742A BYP ISO
- KJV0741A, DG A STARTING AIR TO PRESS INST'N KJV0743A BYP ISO

b. WHEN pressure is equalized, CLOSE the following valves:

- KJV0740A, DG A STARTING AIR TO PRESS INST'N KJV0742A BYP ISO
- KJV0741A, DG A STARTING AIR TO PRESS INST'N KJV0743A BYP ISO

NOTE

The fuel transfer pump start may occur several minutes after diesel start. The following step is a Continuous Action, and its completion may be accomplished at any time following the diesel start.

6.1.15. WHEN PJE01A, EMERGENCY FUEL OIL SYS STORAGE TANK A F.O. XFR PUMP A, starts, CHECK the following:

a. PJE01A starts at greater than or equal to 3.5 feet day tank standpipe level, using one of the following level indicators:

- JE LI-12A, DG A DAY TANK FUEL LEV
- JEL0012B, EMERG F.O. DAY TK A FUEL LEV IND B

b. Annunciator 89E, DG NE01 STND PIPE LEV LO, did NOT alarm prior to pump start.

c. RECORD PJE01A started as SAT/UNSAT on Attachment 7, Local A Diesel Data Sheet.

NOTE

CGM01A, DG VENT SPLY FAN A, will start in AUTO when ambient room air temperature rises to 100°F.

6.1.16. WHILE continuing with the procedure, PERFORM the following:

a. CHECK CGM01A, DG VENT SPLY FAN A, is running locally.

b. RECORD CGM01A runs on Attachment 7, Local A Diesel Data Sheet.

6.1.17. DIRECT Operations Technician to:

- MONITOR operating parameters using Attachment 3, Diesel Generator A Normal Operating Parameter Log, until stable.
- RECORD readings hourly on Attachment 3, Diesel Generator A Normal Operating Parameter Log, until diesel is shutdown. [Ref: 8.2.3]

6.1.18. Momentarily PLACE NE HS-5, DG NE01 UNIT PARALLEL, in PARALLEL.

6.1.19. At NE107, CHECK the white PARALLEL OPERATION light is on.

6.1.20. At RL015, CHECK Annunciator 22B, VOLTAGE CONTROL FREEZE, is in alarm.

6.1.21. CHECK ONE of the following:

- Computer Point NBQ0003, NB03 CAPACITOR BANK, indicates FREEZE.
- Capacitor Bank NB03 is out of service.

6.1.22. CHECK ONE of the following:

- Computer Point NBX0001, XMFR XNB01, indicates FREEZE.
- Transformer XNB01 LTC is in manual.

NOTE

It will take approximately two (2) minutes to ramp speed to the rated 514 rpm.

At approximately 471 rpm, the following windows will alarm, then reset when the generator field is flashed:

- MCB Annunciator 20B, DG NE01 UV OR UF
- KJ121 Annunciator 7B, GENERATOR UNDER VOLTAGE
- KJ121 Annunciator 7C, GENERATOR UNDER-EXCITED

6.1.23. Using KJ HS-7A, DG NE01 GOV, RAISE speed UNTIL meter NE SI-1, DG FREQ/SPD, indicates 500 to 530 rpm.

NOTE

When the field is flashed, annunciator 7E, GENERATOR FIELD GROUND, may come in but should clear quickly upon completion of the field flash.

CAUTION

Placing KJHS0073 in RATED/AUTO FLASH below 500 RPM may cause a VOLTS/HERTZ trip.

6.1.24. At panel NE107, PLACE KJHS0073, DG NE01 START SPD/AUTO FLASH SW, in RATED/AUTO FLASH.

6.1.25. IF necessary, ADJUST voltage and frequency as follows:

- Using KJ HS-7A, DG NE01 GOV, ENSURE 58.8 to 61.2 Hz indicated by NE SI-1, DG FREQ/SPD.
- Using NE HS-13A, DG NE01 AUTO VOLT REG, ENSURE 3740 to 4320 VAC indicated by NE EI-1, DG VOLTS.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Instrument Air / 8	Group #	1		
	K/A #	00065 G2.1.28		
	Importance Rating	4.1		
Knowledge of the purpose and function of major system components and controls.				

Question # 14

Reactor Power is 100%.

- A loss of air is occurring.
- Air Header pressure is currently 115 psig and lowering.

(1) What is the HIGHEST air header pressure that KAPV0011, Service Air Header pressure control valve, should respond to this event?

And

(2) What is the reason for this action?

A. (1) 105 psig.
(2) CLOSES to isolate a potential leak in the Service Air Header.

B. (1) 105 psig.
(2) OPENS to assist in maintaining Instrument Air Header pressure.

C. (1) 110 psig.
(2) CLOSES to isolate a potential leak in the Service Air Header.

D. (1) 110 psig.
(2) OPENS to assist in maintaining Instrument Air Header pressure.

Answer: C

Explanation:

Per OTO-KA-00001 step #B.4, shows that @110 psig the KAPV00111 closes. The distractor of 105 psig is when the instrument air dryer pressure is low and the imnlet and ouitlet valves on the standby air dryer train fail open.

Per OTN-KA-00001 step 3.2 “The Service Air System is automatically isolated at 110 psig by closure of KAPV0011, COMPRESS AIR SYS SERV AIR SPLY PRESS STRL VLV.”

The distractor of to maintain instrument air header pressure is wrong as this is the purpose of the flow restricting orifices which are installed in both the service and instrument air headers. These orifices limit flow to 525 SCFM which is within the capacity of running air compressor (all 3 should be running at 115 PSIG). Therefore "maintaining instrument air header pressure" is plausible as it may be believed the purpose is to prevent a leak in the service building header from drawing down the instrument air header pressure.

A. Incorrect – wrong pressure. See above explanation
B. Incorrect – both are wrong. See above explanation
C. Correct – See above explanation
D. Incorrect – wrong reason. See above explanation

Technical Reference(s):

1. OTO-KA-00001, Partial or Total Loss of Instrument Air, Rev 23
2. M-22Ka01, P&ID Compressed Air System, Rev 35
3. OTN-KA-00001, Compresses Air System, Rev 26

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #14 Service and Instrument Air, Objective B and I:

B. DESCRIBE the purpose and operation of the following Service and Instrument Air components: 6. Pressure Control Valve (KA-PV-11)

I. EXPLAIN the precautions, limitations and bases for KA-PV-11 associated with OTN-KA-00001, "Compressed Air System."

Question Source: Bank # __X__L16394__
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2007_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

4) If KA PI-40 indicates low pressure (115 psig) and all three air compressors are not running a problem might exist with the instrument air dryers:

Instrument Air Pressure (psig)	**Automatic Actuation**
105	Instrument Air Dryer press low. Inlet-Outlet valves on Standby air dryer train fail open.
110	KAPV0011, Compressor Air System Service Air Supply Pressure Control Valve closes
112	Compressor Air Pressure Low

COMPRESSED AIR SYSTEM

1.0 PURPOSE

1.1. To establish proper alignment of the plants instrument and service air systems.

1.2. To provide operating instructions for the air compressors and dryer trains.

2.0 SCOPE

This procedure applies to the entire Instrument and Service Air Systems, with the following exceptions:

- During hydrogen purging, instructions for the operation of the following valves is provided by OTN-GS-00001, Containment Hydrogen Control System:
  - KAFV0029, RX BLD INST AIR SPLY FLOW CTRL VLV
  - KAHV0030, H2 CTRL SYS M/U AIR HV
- In MODES 5 or 6, instrument air to containment is maintained by OTS-KA-00001, Instrument Air Alternate Containment Air Supply During P-30 Outages.

3.0 PRECAUTIONS AND LIMITATIONS

3.1. Changes made in the status of the compressed/instrument air system have the potential to affect air operated components throughout the plant.

3.2. The Service Air System is automatically isolated at 110 psig by closure of KAPV0011, COMPRESS AIR SYS SERV AIR SPLY PRESS STRL VLV.

4.0 PREREQUISITES

None

-END OF SECTION-

CONTINUOUS USE

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
LOCA Outside Containment / 3	Group #	1		
	K/A #	W/E 4 EK2.1		
	Importance Rating	3.5		
Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.				

Question # 15

Operators are performing actions of ECA-1.2, LOCA Outside Containment.

(1) What action MUST be done in order to CLOSE EM HV-8835, SI Pumps to Cold Leg Injection Valve?

And

(2) ECA-1.2 step #3, "Check if Break is Isolated", directs the operator to monitor for a rise in ____(2)______ to verify the LOCA is isolated?

A. (1) Place the Power Lockout Switch in the NON ISO position
(2) PZR Level

B. (1) Place the Power Lockout Switch in the NON ISO position
(2) RCS Pressure

C. (1) Open EM HV8802A&B, SI Pump Discharge to Hot Leg Injection Valves
(2) PZR Level

D. (1) Open EM HV8802A&B, SI Pump Discharge to Hot Leg Injection Valves
(2) RCS Pressure

Answer: B

Explanation:

Per ECA-1.2, step #2b, to reposition this valve the power needs to be returned to the valve operator by placing the power Lockout switch to the Non ISO (not isolated) position. If left in the ISO position (normal lineup), no motive power will be available for the valve actuator. EM HV 8835 receives an open signal on an SI and should be open when the operators start to perform this step.

Per ECA-1.2, step #2c, operators are monitoring for a pressure rise. "Sequentially CLOSE and OPEN the following valves (or sets of valves) one at a time and MONITOR for an RCS pressure *rise: PZR level rising is plausible because if the LOCA has been isolated and ECCS flow is still injecting PZR could return on scale or rise if already onscale but the procedure ECA-2.1 observable response will be RCS pressure rising up due to ECCS pump head.*

The distractor of EMHV 8802A&B is plausible because there are in parallel flow paths to the RCS. One flow path is to the RCS hot leg and one to the RCS cold leg. With the data given in the stem, i.e the is a LOCA accident in progress, it may be falsely believed that the operator is required to establish an ECCS flowpath into the RCS hot legs before the ECCS flowpath to the RCS cold legs are isolated. Specifically, ECCS flow into the RCS must be available prior to closing a valve i.e. 8802A &B must be open before EM HV 8835 is closed as there is a LOCA in progress.

A. Incorrect – ECA-1.2 directs PZR Pressure not PZR level. See above explanation
B. Correct – See above explanation
C. Incorrect – both are wrong. See above explanation
D. Incorrect – the action is wrong. See above explanation

Technical Reference(s):
1. ECA-1.2, LOCA Outside Containment, Rev 7
2. M-22EM01, P&ID High Pressure Coolant Injection System, Rev 38

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #17 Safety Injection, Objective B:
DESCRIBE the purpose and operation of the following SI System components:
8. SI to Cold Leg Isolation Valve (EM-HV-8835)

T61.003D, Emergency Operations, LP #14, ECA-1.2, Objective F: OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ECA-1.2.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Not resetting SI (done in step #2.a) is also a correct answer for part a)

k/a/ match as this safety system component that is manipulated during the performance of ECA-1.2 and the candidate must understand the features and interlocks (or lack there off) of this component. Not all ECCS valves have this Power ISO / NON ISO Lockout switch feature.

Revised question and explanation per NRC comments about plausibility of distractors

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

2. TRY To Identify And Isolate Break:

a. RESET SI If Necessary:

- SB HS-42A
- SB HS-43A

b. PLACE the following Power Lockout switches in NON ISO position:

- EJ HIS-8809AA
- EJ HIS-8809BA
- EM HIS-8835A

c. Sequentially CLOSE and OPEN the following valves (or sets of valves) one at a time and MONITOR for an RCS pressure rise:

- RHR To Accumulator Injection Loops 1 And 2:
  - EJ HIS-8809A
- RHR To Accumulator Injection Loops 3 And 4:
  - EJ HIS-8809B
- SI Pumps To Cold Leg Injection:
  - EM HIS-8835
- Boron Injection Header Outlet:
  - EM HIS-8801A

    AND

  - EM HIS-8801B

d. MAINTAIN affected valve(s) - CLOSED

e. PLACE the following Power Lockout switches in ISO position:

- EJ HIS-8809AA
- EJ HIS-8809BA
- EM HIS-8835A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**CHECK If Break Is Isolated:**	
	a. RCS pressure - RISING	a. Go To ECA-1.1, Loss Of Emergency Coolant Recirculation, Step 1.
	b. Go To E-1, Loss Of Reactor Or Secondary Coolant, Step 1	

-END-

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Emergency Coolant Recirc. / 4	**Group #**	1		
	K/A #	W/E 11 K1.3		
	Importance Rating	3.6		
Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation): Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Emergency Coolant Recirculation).				

Question # 16

What are the actions required in ECA-1.1, Loss of Emergency Coolant Recirculation, when Annunciator 47A, RWST EMPTY, is received? (RWST level is verified to be less than 5%.)

A. Stop ONE Containment Spray pump taking suction from the RWST, and throttle SI and RHR flow in accordance with decay heat removal requirements.

B. Stop ONE Containment Spray pump taking suction from the RWST, align normal charging, and initiate secondary depressurization to facilitate SI Accumulator injection.

C. Stop ALL pumps taking suction from the RWST, and throttle SI and RHR flow in accordance with decay heat removal requirements.

D. Stop ALL pumps taking suction from the RWST, align normal charging, and initiate secondary depressurization to facilitate SI Accumulator injection.

Answer: D

Explanation: *ECA-1.1 step #6 is a continuous action step that directs the operator to check if RWST is greater than 6%, if not the RNO action sends the operator to step #30. This step stops pumps taking suction from the RWST and places the switches in pull to lock. Steps 31-33 align the VCT and depressurize the secondary side to lower RCS pressure to facilitate SI accumulator injection.*

Step #7 Determine Containment Spray Requirements (Suction from RWST) part b would determine the number of Spray pumps and cooler fans that are required to be running. For a RWST level less than 6%, no pumps and 0-4 fans in slow speed can be running.

Step #14 establishes only one train of ECCS flow and Step #17b RNO (also a continuous action step) establishes minimum SI flow to remove decay heat. These procedure steps make " and throttle SI and RHR flow in accordance with decay heat removal requirements." plausible if the candidate cannot properly apply the procedure to the given question stem conditions.

A. Incorrect – See above explanation.
B. Incorrect – See above explanation.
C. Incorrect - See above explanation.
D. Correct – See above explanation

Technical Reference(s):

1. ECA-1.1, Loss of Emergency Coolant Recirculation, Rev 11
2. OTA-RK-00018, Addendum 47A, RWST Empty, Rev 1

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #13, ECA-1.1, Objective C: DESCRIBE the requirements and basis for the Continuous Action Steps of ECA-1.1, Loss of Emergency Coolant Recirculation.

Question Source: Bank # __X L16516____
Modified Bank # ______
New _______

Question History: Last NRC Exam _N/A___2011 Audit Exam________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised per NRC comments and changed.

RWST Empty	47A
Initiating Device: 1. BNLB0930D 2. BNLB0931D 3. BNLB0932D 4. BNLB0933D — **Setpoint:** 1. 5.65% 2. 5.65% 3. 5.65% 4. 5.65% — **Reset:** 1. 6.65% 2. 6.65% 3. 6.65% 4. 6.65%	RWST EMPTY

Initiating Device:	Setpoint:	Reset:
1. BNLB0930D	1. 5.65%	1. 6.65%
2. BNLB0931D	2. 5.65%	2. 6.65%
3. BNLB0932D	3. 5.65%	3. 6.65%
4. BNLB0933D	4. 5.65%	4. 6.65%

1.0 AUTOMATIC ACTIONS:

1.1. None

2.0 IMMEDIATE ACTIONS:

2.1. CHECK the following:

- BN LI-930, RWST LEV
- BN LI-931, RWST LEV
- BN LI-932, RWST LEV
- BN LI-933, RWST LEV
- BN LR-930, RWST LEV

2.2. IF an instrument failure is evident, Go To OTO-BN-00001, RWST Level Channel Malfunction.

> **CAUTION:**
> At 5.65% level, the NPSH available to the following pumps is marginal. The top of the horizontal RWST discharge pipe is at a tank level of approximately 2.3%.

2.3. STOP and PLACE in PULL TOLOCK any ECCS Pump still taking a suction from the RWST using the associated Control Switch:

- BG HIS-1A, CCP A
- BG HIS-2A, CCP B
- EM HIS-4, SI PUMP A
- EM HIS-5, SI PUMP B
- EJ HIS-1, RHR PUMP A
- EJ HIS-2, RHR PUMP B
- EN HIS-3, CTMT SPRAY PUMP A
- EN HIS-9, CTMT SPRAY PUMP B

3.0 OPERATOR ACTIONS:

3.1. IF required, CONTINUE with the actions from OTA-RK-00018 Addendum 47D, RWST Level High or Low.

3.2. Return To the procedure and step in effect.

4.0 SUPPLEMENTAL INFORMATION:

4.1. Drawings:

- M-22BN01, P&ID, Borated Refueling Water Storage System
- M-22BG01, P&ID, Chemical and Volume Control System
- M-22EJ01, P&ID, Residual Heat Removal System
- M-22EM01, P&ID, High Pressure Coolant Injection System
- M-22EN01, P&ID, Containment Spray System
- E-23BN07, Schematic Diagram, Miscellaneous Instruments

4.2. TDB-001, Tank Data Book, TBN01, Refueling Water Storage Tank

4.3. Computer Points:

RWST Level Computer Points	
BNL0930D, RWST LEVEL	REL0930A, RWST CH 1 LEVEL
REU0511, RWST LEVEL	REL0931A, RWST CH 2 LEVEL
	REL0932A, RWST CH 3 LEVEL
	REL0933A, RWST CH 4 LEVEL

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	Group #	1		
	K/A #	W/E 05 EA2.1		
	Importance Rating	3.4		
Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Facility conditions and selection of appropriate procedures				

Question # 17

The crew is responding to a Loss of Coolant Accident.

The crew has transitioned from E-0 and is performing the appropriate procedure.

Current plant conditions are:

- RCS pressure — 1200 psig and lowering
- CTMT pressure — 30 psig and stable
- Containment Spray Pumps — OFF
- RCS subcooling — 10°F
- Core Exit TC's — 730°F and stable
- Steam Gen press — 900 psig and stable
- Steam Gen levels — 10% NR all generators
- AFW status — 265,000 lbm/hr maximum feed flow
- RCP status — OFF
- RVLIS Pump OFF Indication — 70%

What is the correct action to take?

A. Immediately transition to FR-C.1, Response to Inadequate Core Cooling.

B. Immediately transition to FR-Z.1, Response to High Containment Pressure.

C. Immediately transition to FR-H.1, Response to Loss of Secondary Heat Sink.

D. Immediately transition to ES-1.2, Post LOCA Cooldown and Depressurization.

Answer: C

Explanation: *With the given conditions and the fact that adverse containment values exist, a heat sink does not exist (SG NR level needs to be 25% or AFW flow at 285,000lbm/hr) and a transition to FR-H.1 is required. A transition out of E-0 has been made so RED path CSF are required to be addressed.*

The decision to stay in the current procedure is plausible if the candidate does not recognize that adverse containment values have resulted in a loss of heat sink and makes the decision not to perform a yellow path CSF (which is at the discretion of shift management).

With the containment readings an orange path on containment exists and the candidate may believe that this must be addressed prior to heat sink. But the heat sink red path takes priority over this orange path to FR-Z.1. The action to verify containment isolations is a major purpose of the FR procedure and basically step #1 RNO actions.

With the RCS Subcooling less than 50F and no RCP running and Core Exit TC greater than 706F, the candidate must know that a RVLIS pumps off indication less than 42% will result in a red path on core cooling. This distractor is plausible as core cooling is a higher priority than secondary heat sink but with RVLIS pumps off @70% only a orange path exists and the red path on heat sink takes priority

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Correct – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):

1. CSF-1, Critical Safety Function Status Trees, Rev 10
2. ODP-ZZ-00025, EOP/OTO User's guide, Rev 27, Section 4.23 and 4.24

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #26, FR-H series, Objective B: B. DESCRIBE the Symptoms and/or Entry Conditions for:
1. FR-H.1, Response To Loss Of Secondary Heat Sink.

Question Source: Bank # _X R13482_____
Modified Bank # ______
New _______

Question History: Last NRC Exam _____N/A_______

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

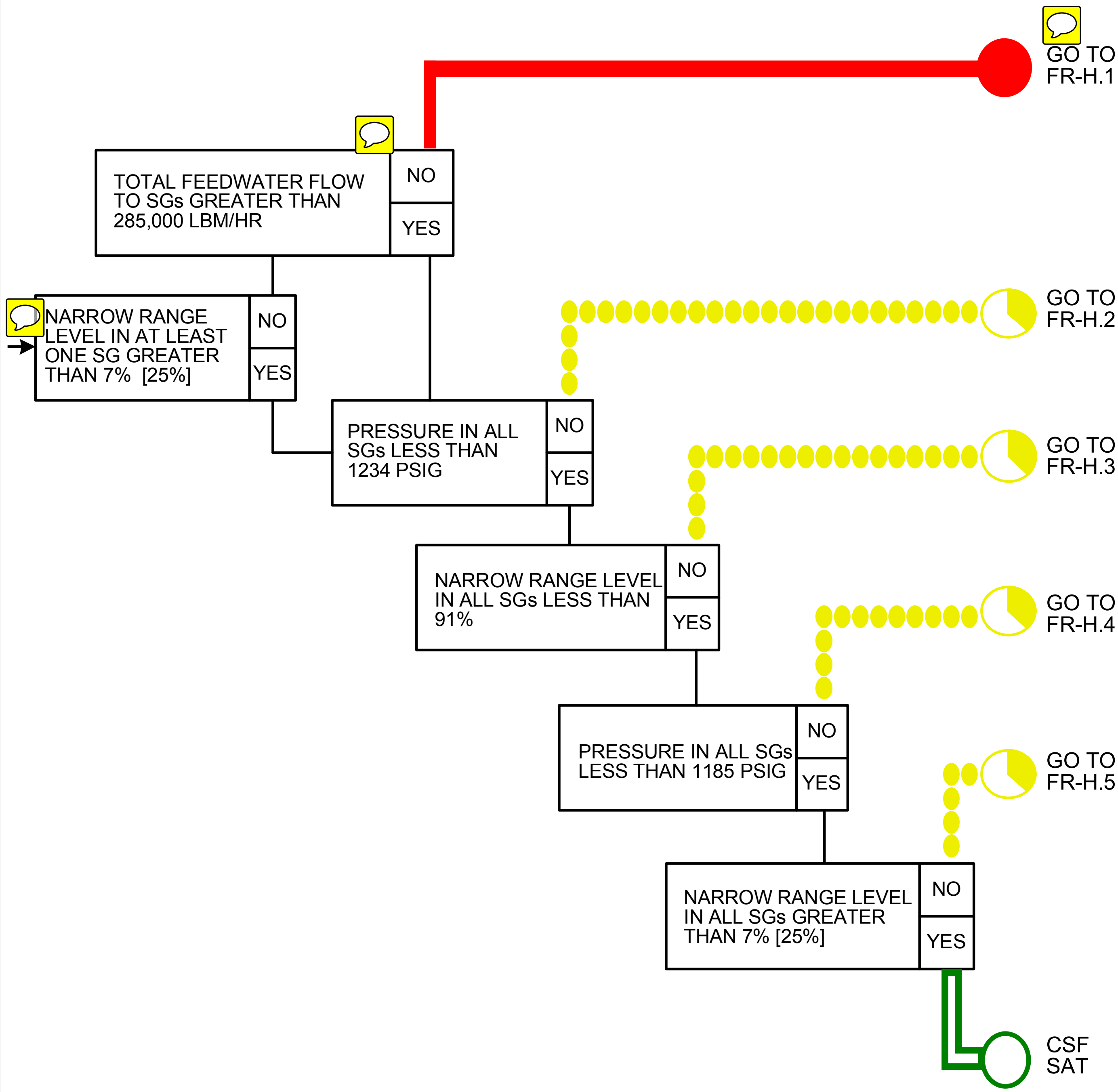
10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised all choices per NRC comments. Question is at the RO level of knowledge as per ES 401 Attachment 2 page 7 states that SRO-only knowledge should not be claimed for questions that can be answered solely using fundamental knowledge of: plant parameters that require direct entry to major EOPs; e.g., major Westinghouse EOPs are E0, E1, E2, E3, ECA-0.0, and Red/Orange Functional Restoration Procedures and major General Electric EOPs are Reactor Vessel Control, Primary Containment Control, Secondary Containment Control, and Radioactive Release Control. The above FR procedures listed in the question are RED path procedure per CSF-1 and therefore are not SRO making the RO level knowledge.

Figure 3
Heat Sink



Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Generator Voltage and Electric Grid Disturbances	Group #	1		
	K/A #	000077 G2.1.19		
	Importance Rating	3.9		
Ability to use plant computers to evaluate system or component status.				

Question # 18

Reactor Power is 100%.

- Annunciator 134D, Switchyard Voltage High/Low, is LIT.
- The following plant process computer parameters are present:
  - MSE345AM, 345 kV BUS A 1 MIN AVG VOLT, 340kV
  - MSE345BM, 345 kV BUS B 1 MIN AVG VOLT, 341kV
- The Transmission Operation Supervisor reports a Category 8 Alarm is present and the predicted contingency voltage is the same as the plant process computers values.
- Operators are performing the below step of OSP-NB-00001, Attachment 4; Switchyard Voltage Requirements – Main Generator On Line (Mode 1)

NB01 and NB02 Powered from separate ESF transformers (Dual Source) or from the same ESF transformer (Single Source)				
Configuration	Required Voltage (kV)		Predicted Contingency Voltage	Initials
	Dual Source	Single Source		
LTC(s) AUTO & Cap Banks	372.6 to 329.8	372.6 to 332.9		
LTC(s) MAN* & Cap Banks	372.6 to 335.7	372.6 to 344.3		
LTC(s) AUTO & No Cap	372.6 to 341.2	372.6 to 344.3		
LTC(s) MAN* & No Cap	372.6 to 347.1	(Note 2)		

(1) Two Off-Site AC electrical sources shall be Operable in …?

And

(2) Using the above procedure step, what is the status of the Off-Site Sources?

A. (1) MODES 1-3 ONLY
(2) Operable

B. (1) MODES 1-3 ONLY
(2) Inoperable

C. (1) MODES 1-4
(2) Operable

D. (1) MODES 1-4
   (2) Inoperable

Answer: C

Explanation: *With annunciator 134D LIT, it provides direct to monitor Switchyard Bus voltages using process computer points amnd then directs you to OSP-NB-00001 if the values are low. If the computer point for NB03 is in alarm, then per this NOTE in OSP "Capacitor Banks are in service if no trouble alarm is actuated for their respective ESF transformer [(Window 19D, XNB01 XFMR/VOLT CTRL TRBL, and 22D, XNB02 XFMR/VOLT CTRL TRBL) and Computer points NBU0003A, NB03 CAPACITOR BANK PLC TROUBLE, and NBU0004A, NB04 CAPACITOR BANK PLC TROUBLE]". Therefore, with no information in the stem about these computer point or annunciators the NB03 and NB04 CAP banks are in service.*

When the PPC bus voltage values are compared to the steps table for Dual Source with the CAP banks in service, both Switchyard bus voltages are above the required minimum of 329.8kV. 340 and 341kv was chosen such that it is between the values of LTC in AUTO and CAP bank or LTC in AUTO and no CAP Banks for the dual source. Furthermore, the given voltages are between the single and dual source values for when the LTC are in Man with CAP Banks in service. Inoperable is a plausible distractor if the table is used incorrectly with the voltages given.

Per Technical Specification 3.8.1, AC sources Operating – the mode of Applicability is Modes 1-4.

A. Incorrect – wrong mode of applicability
B. Incorrect – both are wrong – see above explanations
C. Correct – see above explanation
D. Incorrect – AC offsite source are operable

Technical Reference(s):
1. OTA-RK-00026, Addendum 134D, Rev 2
2. OSP-NE-00003, Technical Specification Actions – A.C. Sources, Rev 29
3. OSP-NB-00001, Class 1E Electrical Source Verification, Rev 39
4. Technical Specification 3.8.1, AC Source Operating, Amendment #199

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #6, Objective B and C:

B: DESCRIBE the purpose and operation of the following Safeguards Power System components and subsystems:

1. ESF 'Load Tap Changing' Transformers, XNB01 and XNB02
2. Capacitor Banks, NB03 and NB04

C. IDENTIFY the Safeguards Power System Main Control Board (MCB) controls and indications and DESCRIBE how each is used to predict, monitor or control changes in the Safeguards Power System.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question History: Last NRC Exam __N/A__________

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s); and

c. Load Shedder and Emergency Load Sequencer (LSELS) for Train A and Train B.

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

ACTIONS

-- NOTE --

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	A.2	------------ NOTE ----------- In MODES 1, 2, and 3, the turbine driven auxiliary feedwater pump is considered a required redundant feature. -------------------------------	
			(continued)

Switchyard Voltage High Low			134D
Initiating Device: 1. MSEI0084	**Setpoint:** 1. HI ≥ 362.25 kV LO ≤ 331.8 kV	**Reset:** 1. HI < 361.35 kV LO > 332.7 kV	**SWYD VOLTAGE HILO**

1.0 AUTOMATIC ACTIONS:

1.1. None

2.0 IMMEDIATE ACTIONS:

2.1. None

3.0 OPERATOR ACTIONS:

NOTE:

MSEI0084 is only a valid comparison for Bus B if MTGY - CAL7 line is energized AND V85 and both of its disconnects are closed. [Ref: 4.2.]

MSEI0084 is only a valid comparison for Bus A if MTGY - CAL7 line is energized AND V81 and its disconnect are closed. [Ref: 4.2.]

If the computer points below are unavailable, MSER0001, BUS A & BUS B VOLTAGE RECORDER, may be used instead.

3.1. CHECK switchyard voltage by comparing the following:

- MSEI0084, MTGY-CAL7 L-L VOLTAGE
- Computer Points:
  - MSE345AM, 345 kV BUS A 1 MIN AVG VOLT
  - MSE345BM, 345 kV BUS B 1 MIN AVG VOLT

3.2. CONTACT the Power Supply Supervisor and CHECK the System condition.

3.3. IF the Main Generator is on line, REDUCE KV/VARS to clear the alarm.

3.4. IF voltage is low, Refer To OSP-NB-00001, Class 1E Electrical Source Verification, to determine OPERABILITY of Off-Site AC Sources.

OPERATOR ACTIONS (Cont’d):

> **NOTE:**
> Operation with high transformer loading near the main generator capability curves and at voltages above 362.25 kV can lead to increased transformer operating temperatures and reduced life.

3.5. IF voltage cannot be adjusted below 362.25 kV:

- CONTACT Systems Engineering for guidance
- MONITOR transformer temperatures are within limits

3.6. INFORM the Transmission Operations Supervisor of sustained voltage swings of ± 10% or greater.

3.7. IF problems in meeting the issued voltage schedule occur, PERFORM the following:

- CONTACT Transmission Operations to request an exception or variance.
- LOG the expected duration AND reason for variance in Auto Log.

4.0 SUPPLEMENTAL INFORMATION:

4.1. E-23MS01, Switchyard Miscellaneous Circuits

4.2. CAR 200002551, WPA Tagging of V85 results in unexpected loss of switchyard voltage.

4.3. CAR 200601731, Provide New Generator MVAR Limits.

4.4. The low voltage setpoint is based on the normal lineup where NB01 and NB02 are powered from separate ESF transformers.

4.5. AUE-ADM-2223, Distubance Reporting

4.6. AUE-ADM-2234, Maintaining Network Voltage Schedules

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Inoperable/Stuck Control Rod / 1	**Group #**	2		
	K/A #	00005 AK1.03		
	Importance Rating	3.2		
Knowledge of the operational implications of the following concepts as they apply to Inoperable / Stuck Control Rod: Xenon transient				

Question # 19

(REFERENCE PROVIDED)

Reactor Power is 100%.

- Shutdown Bank A Rod D-14, drops to the bottom of the core.
- 15 minutes later and during recovery of Rod D-14, it becomes stuck at position 010 and will not withdrawal or insert.

(1) Shutdown Margin must be verified to be within the limits of the COLR within a MAXIMUM of ……?

And

(2) Due to the Xenon transient, the Reactor Operator should expect SE NI-42B, Power Range Nuclear Instrument 42B, readings to start to slowly __(2)___ over the next hour. (Assume Turbine Load, Reactor Power, and Tave remain constant.)

A. (1) 30 minutes
(2) lower

B. (1) 30 minutes
(2) rise

C. (1) 1 hour
(2) lower

D. (1) 1 hour
(2) rise

Answer: C

Explanation: *Per Technical Specification 3.1.5, Required Action A.1.1, SDM must be verified within 1 hour. Procedurally, the operator will performed OTO-SF-00001 and at step #11 perform* *the RNO which directs the operator to Attachment A. Step A.4 also provides direct to check that SDM is within the limits of the COLR within 1 hour. 30 minutes is plausible as it a less than 1 hour required action so it has to be known from memory but wrong as it applies to AFD not within limits and the action is to lower power to less than 50%. No information is provided on AFD (either in or out of spec) and therefore 30 minutes is wrong.*

With the dropped rod and failed recovery due to it being stuck, power is suppressed in the area near PR SE NI-42B, this lowers the Xenon burnout by absorption but the production from iodine decay is still present and Xenon concentration starts to rise which would lower the PR 42B reading due to less neutrons leaking from the core (more are being absorbed by Xenon in this area of the core). Rise is plausible if the student does not understand the relation in neutron leakage and a higher poison concentration or answers the question based on xenon concentration instead of PR reading the Reactor Operator would observe.

A. Incorrect – wrong time
B. Incorrect – both are wrong
C. Correct – see above explanation
D. Incorrect – wrong direction

Technical Reference(s):
1. Curve Book, Figure 8-7, RCS LOOP with Control Rods and Excore Neutron Detector Locations, Rev. 000
2. Technical Specifications, 3.1.5, Shutdown Bank Insertion Limits

References to be provided to applicants during examination:
1. Curve Book, Figure 8-7, RCS LOOP with Control Rods and Excore Neutron Detector Locations, Rev. 000

Learning Objective:
1. T61.GFES, Reactor Operational Physics, LP #44, Objective 22: Explain reactor response to a control rod insertion.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ____N/A________

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

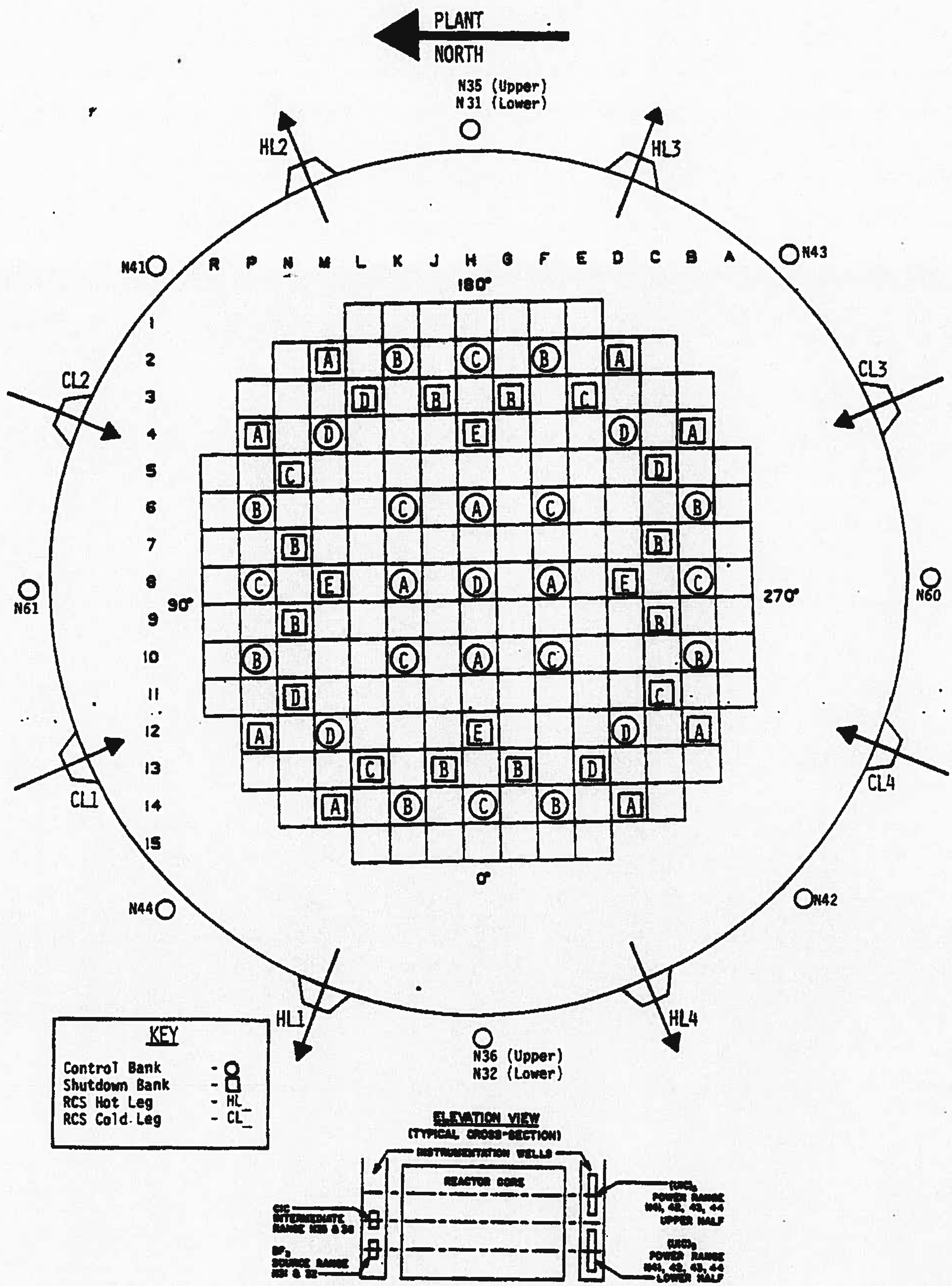
10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

FIGURE 8-7 Rev. 000

RCS LOOP ORIENTATION WITH CONTROL ROD AND EXCORE NEUTRON DETECTOR LOCATIONS



R. Affolter 1-3-30-86

Superintendent, Engineering Date

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Condenser Vacuum	Group #	2		
	K/A #	00051 G2.4.11		
	Importance Rating	4.0		
Knowledge of abnormal condition procedures				

Question # 20

Reactor Power is 25%.

- A startup is in progress.
- Annunciator 116B, Cond A Vac Lo is LIT.
- The crew has entered OTO-AD-00001, Loss of Condenser Vacuum.
- Condenser backpressure is 8 inches HgA and stable.

What action is required per the abnormal procedure?

A. Trip the turbine and go to OTO-AC-00001, Turbine Trip.

B. Secure from the load increase and immediately start reducing load.

C. Manually TRIP the Reactor and Go To E-0, Reactor Trip Or Safety Injection.

D. Stabilize the plant at the current power level, initiate action to restore condenser vacuum per Attachment A.

Answer: C

Explanation:

Per the continuous action step #1 if main condenser backpressure is greater than 7.5 inches HgA the RNO applies. The RNO directs a reactor trip if reactor power is greater than 10%. This action is of higher importance among the other actions and continuous actions of the procedure.

A. Incorrect – The turbine trip setpoint is 8.5 inches HgA. This action to only trip the turbine is plausible because of the note discussing the P-9 interlock at 50% power and that below P-9 a turbine trip does not cause a reactor trip. If the candidate does not apply the continuous action step #1 correctly and using the RNO action as greater or less than 50%, not 10%, they would arrive at tripping the turbine and transitioning to OTO-AC-00001.
B. Incorrect – This is the action in step #7 which is also a continuous action statement but is not the correct as backpressure has past the trip requirement setpoint. This continuous action is used to maintain backpressure somewhere below 7.5 inches HgA by reducing turbine load
C. Correct – see above explanation
D. Incorrect – This is step #2 of the OTO and per the note prior to the step Attachment A is a

diagnostic process which "contains actions which may be performed for slow moving events". This is not a slow moving event as the trip requirement setpoint has already been reached. Furthermore step #2 is not a continuous action step.

Technical Reference(s):

1. OTO-AD-00001, Loss of Condenser Vacuum, Rev 32

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #7, OTO-AD-00001; Objective C & D:

C. DESCRIBE Continuous Action Step(s) including the required Response Not Obtained actions.

D. Given a set of plant conditions or parameters indicating a Loss of Condenser vacuum, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # _X L16735_____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2009________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

When Reactor Power is less than 50% (P-9), a Turbine trip will not result in an automatic reactor trip.

# 1.	**CHECK Main Condenser Backpressure - LESS THAN 7.5 INCHES HGA**	IF Reactor power is greater than or equal to 10%, THEN PERFORM the following: a. Manually TRIP the Reactor. b. Go To E-0, Reactor Trip Or Safety Injection. IF Reactor power is less than 10%, THEN PERFORM the following: a. Manually TRIP the Main Turbine. b. Go To OTO-AC-00001, Turbine Trip.

NOTE

Attachment A, Diagnostic Actions, contains actions which may be performed for slow moving events.

2.	**Refer To Attachment A, Diagnostic Actions, As Time Permits To Perform Actions**	
# 3.	**CHECK Main Condenser Backpressure - GREATER THAN 4.0 INCHES HGA** a. Obtain permission from the SM/CRS b. PLACE DA HS-113, CIRC WTR PUMP TURB SETBACK, to ENABLE	
4.	**CHECK Main Condenser Backpressure - DETERIORATING OR STABLE**	Go To Step 17.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. PLACE Rod Control In AUTO:

- SE HS-9

6. MANAGE Reactivity:

a. PERFORM Reactivity Management Brief:

- DISCUSS Amount And Rate of Turbine Load reduction
- DETERMINE amount of boric acid needed

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- If using the Load Limit Potentiometer, unloading at greater than 50 MWe per minute may arm the Main Condenser Steam Dumps.
- If changing load reduction methods to the Load Limiter ENSURE Load Set is restored to AT SET LOAD prior to using the Load Limiter.

NOTE

Steps 7 and 8 may be performed concurrently while continuing with this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
# 7.	**REDUCE Turbine Load At Less Than Or Equal To 5% Per Minute Using Any Of The Following:**	REDUCE Turbine load at less than or equal to 5% per minute using the Standby Load Set Potentiometer.

- REDUCE Turbine load using the %/Min Loading Rate:
  a. SLOWLY LOWER Load using the DECREASE LOAD pushbutton until all of the following are met:
     - Load Limit Limiting Light - EXTINGUISHED
     - Decrease Loading Rate "OFF" Light - LIT
     - Loading Rate Limit %/MIN "1/2" Light - LIT
  b. ROTATE Load Limit Set potentiometer fully clockwise
  c. SELECT Decrease Loading Rate - ON
  d. SET Loading Rate Limit %/Min to desired value
  e. LOWER load set MW toward desired load using the DECREASE LOAD pushbutton

  OR

- REDUCE Turbine load using the Load Limit Potentiometer

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Accidental Gaseous Radwaste Release	**Group #**	2		
	K/A #	000060 AA1.02		
	Importance Rating	2.9		
Ability to operate and / or monitor the following as they apply to the Accidental Gaseous Radwaste: Ventilation System				

Question # 21

An Accidental Gaseous Radwaste release is occurring from the decay tanks.

If GH RE-10B, Radwaste Building Exhaust Fans Discharge Header Radiation Monitor, HI HI Radiation Alarm is received, the Reactor Operator will verify that the ……..?

A. Control Room Ventilation ISOLATES

B. Waste Gas Compressors (SHA02A & B) TRIP

C. Radwaste Building Supply Unit (SGH01) TRIPS

D. Gas Decay Tanks to RW HVAC Discharge Valve (HA HCV-14) ISOLATES

Answer: D

Explanation:
If a PRM Hi Hi alarm is received, Annunciator 61A will alarm and this directs the operator to OTA-SP-RM011. Per Attachment 1 of the OTA, when a HI HI on GH-RE10B isolates Gas Decay Tanks.

A. Incorrect – plausible as it may be believed that a HI HI on GHRE10B will cause CRVIS to occur. This may be believe due to the importance of establishing a Control Room Isolation boundary to protect the CR staff and maintain dose under 10 CFR 20. Additionally an exhaust hi hi radiation in the fuel building causes a FBIS and then CRVIS and this may be incorrectly applied to the Radwaste building exhaust radiation monitors.
B. Incorrect – plausible as these are a supply to the decay tanks and if they have a leak / release it would be prudent to stop supplying compressed air to the tanks that may be ruptured but wrong as this compressor trips on its supply pressure from the VCT.
C. Incorrect – but plausible as several ventilation systems design have a trip of the supply unit if a high rad conditions exist to prevent a positive pressure from developing and establishing a dp to allow a radiation release to the environment.
D. Correct – see above explanation

Technical Reference(s):

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

1. OTA-RK-00020, ADD 61A, PRM HI HI, Rev 0
2. OTA-SP-RM011, RM-11 Control Panel, Rev 40

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #16 Radwaste, Objective #B: DESCRIBE the purpose and operation of the following Gaseous Radwaste System components:

1. Decay Tanks
2. Waste Gas Tank Sampling / Monitoring
3. Waste Gas release and isolation
4. Radiation Monitoring

Question Source: Bank # __L 4706____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Attachment 1 (Cont'd.)

Sheet 2 of 3

Display STATUS

CHANNEL NUMBER	MONITOR	ATT #	PROCESS	CONTROL FUNCTION	
				RED	YELLOW
381	FC-RE-385	8	TD AFW	Alarm	Alarm, alert
925	GE-RE-92	9	Cond. Air Removal	Alarm	Alarm, alert
271	GG-RE-27	10	Fuel Bldg.	Alarm	Alarm, alert
272	GG-RE-27	10	Fuel Bldg.	Alarm	Alarm, alert
273	GG-RE-27	10	Fuel Bldg.	FBVIS-CRVIS	Alarm, alert
281	GG-RE-28	10	Fuel Bldg.	Alarm	Alarm, alert
282	GG-RE-28	10	Fuel Bldg.	Alarm	Alarm, alert
283	GG-RE-28	10	Fuel Bldg.	FBVIS-CRVIS	Alarm, alert
101	GH-RE-10A	11	Radwaste	Isolates Gas Decay Tank	Alarm, alert
102	GH-RE-10A	11	Radwaste Ventilation	Isolates Gas Decay Tank	Alarm, alert
105	GH-RE-10B	11	Radwaste Ventilation	Does not alarm	Does not alarm
108	GH-RE-10B	11	Radwaste Ventilation	Does not alarm	Does not alarm
109	GH-RE-10B	11	Radwaste Ventilation	Does not alarm	Does not alarm
103	GH-RE-10B	11	Radwaste Ventilation	Isolates Gas Decay Tanks	Isolates GH-RE-10A
224	GH-RE22	N/A	Radwaste Ventilation	Retired MP 93-2031A	
235	GH-RE-23	12	Gas Decay TK Ventilation	Alarm	Alarm, alert
041	GK-RE-04	13	Control Room	Alarm	Alarm, alert
042	GK-RE-04	13	Control Room	Alarm	Alarm, alert
043	GK-RE-04	13	Control Room	CRVIS	Alarm, alert
051	GK-RE-05	13	Control Room	Alarm	Alarm, alert
052	GK-RE-05	13	Control Room	Alarm	Alarm, alert
053	GK-RE-05	13	Control Room	CRVIS	Alarm, alert
414	GK-RE-41	14	Access Cntl	Alarm	Alarm, alert
604	GL-RE-60	15	Aux Bldg Exh	Alarm	Alarm, alert
211	GT-RE-21A	16	Unit Vent	Alarm	Alarm, alert
212	GT-RE-21A	16	Unit Vent	Alarm	Alarm, alert

 CONTINUOUS USE

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
High Reactor Coolant Activity / 9	Group #	2		
	K/A #	00076 AK2.01		
	Importance Rating	2.6		
Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors				

Question # 22

With the unit at power, which of the following describes the radiation monitor(s) that will provide direct confirmation of High RCS Activity in accordance with OTO-BB-00005, RCS High Activity?

A. SJ-RE-01, CVCS Letdown Monitor

B. GT-RE-59, CTMT High Range Area Monitor

C. GT-RE-31, Containment Atmosphere Monitor

D. GT-RE-21B, Unit Vent Air Exhaust Radiation Monitor

Answer: A

Explanation: *Per OTO-BB-00005, SJ RE-01 is a symptom or entry condition. OTA-SP-RM011 Attachment 19, provides direction for a high GT-RE-59 alarm. This alarm is either indicative of Excess RCS leakage or a fuel handling accident but not high RCS activity with an intact RCS boundary. OTA-SP-RM011 Attachment 18, provides direction for a high GT-RE-31 alarm. GT-RE-31 contains a gas, iodine, and particulate channel and is used during mostly during containment ventilation activities such as purge operations (backup to GT-RE 22 and 33) and serves similar functions that GT-RE-59 provides: Excess RCS leakage or a fuel handling accident. OTA-SP-RM011 Attachment 16 provides direction for a high GT-RE-21B. This rad monitor detects high rad in the containment ventilation exhaust as it leaves the unit vent and is a indication of an elevated release due to a LOCA in containment.*

A. Correct – Per OTO-BB-00005, section B, this is the only radiation monitor that is a symptom or entry condition for this procedure.
B. Incorrect – Plausible as this alarm would be present during LOCA or excessive RCS leakage conditions and is also used in the Fission Product Barrier Matrix for EAL determinations.
C. Incorrect – Plausible as this alarm would be present during LOCA or excessive RCS leakage conditions
D. Incorrect – Plausible as this alarm would be present during LOCA or excessive RCS leakage conditions with containment ventilation elevating the release out the unit vent.

Technical Reference(s):
1. OTO-BB-00005, RCS High Activity, Rev 14.
2. OTA-SP-RM011, Rad Monitor Control Panel, RM-11

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off normal Operations, LP #14, OTO-BB-00005, Objective C: DESCRIBE symptoms or entry conditions for OTO-BB-00005, RCS High Activity.

Question Source: Bank # ___X L16535___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2007________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(11)

Comments:

Revised per NRC comments about 1 implausible distractor, reordered distractors.

A. PURPOSE

This procedure provides instructions for responding to high activity in the Reactor Coolant System.

B. SYMPTOMS OR ENTRY CONDITIONS

1) Dose equivalent Iodine-131 activity level:

- Greater than 60 microcuries per gram

  OR

- Rising trend

2) Dose equivalent Xenon-133 activity level:

- Greater than 225 microcuries per gram

  OR

- Rising trend

3) Any of the following Radiation Monitors in alarm:

- SJ RE-01, CVCS Letdown Monitor

C. REFERENCES

1) Implementing:

   a. ODP-ZZ-00001 Addendum 13, Shift Manager Communications

   b. EIP-ZZ-00101, Classification Of Emergencies

   c. APA-ZZ-00345, Fuel Reliability Program

   d. CDP-ZZ-00800, Callaway Resin Monitoring Program

   e. CTP-ZZ-02590, Primary To Secondary Leakrate Determination

   f. OTN-BG-00001, Chemical and Volume Control System

2) Developmental:

   a. None

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
SI Termination	Group #	2		
	K/A #	W/E02 EA2.2		
	Importance Rating	3.5		
Ability to determine and interpret the following as they apply to the (SI Termination): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.				

Question # 23

The Reactor Power was at 100% when a transient occurred.

- The crew has entered the EOP network and has transitioned to ES-1.1, SI Termination.
- The Reactor Operator is checking if Letdown can be established per step #14 and reports the following plant status:
    - o The MSIVs have just closed due to containment pressure
    - o RCS Subcooling is 45°F and stable
    - o Pressurizer Level is 20% and stable
    - o RCS Pressure is 1800 psig and slowly lowering

Per ES-1.1, the control room crew should ….?

A. Re-establish ECCS flow as necessary and go to E-1, Loss of Reactor or Secondary Coolant.

B. Re-align the Boron Injection Header and go to ES-1.2, Post LOCA Cooldown And Depressurization.

C. Reinitiate Safety Injection and go to E-0, Reactor Trip or Safety Injection.

D. Continue in ES-1.1 and when RCS subcooling lowers to 30°F go to E-1, Loss of Reactor or Secondary Coolant.

Answer: A

Explanation: *Per ES-1.1 and the stem, the crew will be performing step 14. The foldout page action applies since SI has been terminated and the RCS subcooling is less than the adverse containment value of 50°F. Adverse containment values apply as the stem indicates that MSIVs have close due to containment pressure which happens above the adverse containment setpoint of 3.5 psig in containment.*

Step #1 of the foldout page states:
IF either condition listed below occurs after SI termination, THEN ESTABLISH ECCS flow as necessary and Go To E-1, Loss Of Reactor Or Secondary Coolant, Step 1:

- *RCS subcooling - LESS THAN 30°F [50°F]*

  OR
- *PZR level - CANNOT BE MAINTAINED GREATER THAN 9% [29%]*

A. Correct – see above explanation
B. Incorrect – Plausible as this is step 8 RNO of ES-1.1 and the stems indicates PZR level is slowly lowering but wrong as the candidate has already moved pass this step of the procedure. The candidate may falsely believe / remember that this is a continuous action step also which adds to its plausibility.
C. Incorrect – Plausible as this is the correct concept to reestablish ECCS flow but the return to E-0 is incorrect per ES-1.1 foldout page as explained above.
D. Incorrect – but plausible as this would be the correct action and values if they candidate does not apply / recognize that adverse containment values are present and an immediate foldout page action and transition to E-1 is required based on subcooling.

Technical Reference(s):
1. ES-1.1, SI Termination, Rev 12

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #9, SI Termination, Objective E and I

E. DESCRIBE the Criteria and the Basis for information as stated on the ES-1.1, SI Termination, Foldout Page.

I. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ES-1.1.

Question Source: Bank # __X R16150____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

k/a match as the operators are performing the SI termination procedure and must apply the given plant information and determine the correct course of action and maintain the plant with the facility license (establishing core cooling and maintain core inventory when necessary)

RO level knowledge as it is testing foldout page criteria of ES-1.1.

FOLDOUT PAGE FOR ES—1.1

1. **SI REINITIATION CRITERIA**
   IF either condition listed below occurs after SI termination,
   THEN ESTABLISH ECCS flow as necessary and Go To E—1, Loss Of Reactor Or Secondary Coolant, Step 1:
   - RCS subcooling — LESS THAN 30°F [50°F]

     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 9% [29%]

2. **SECONDARY INTEGRITY CRITERIA**
   IF BOTH conditions listed below occur,
   THEN Go To E—2, Faulted Steam Generator Isolation, Step 1:
   - Any SG pressure is lowering in an uncontrolled manner OR has completely depressurized.

     AND
   - Affected SG has NOT been isolated using E—2, Faulted Steam Generator Isolation.

3. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
W/E15 Containment Flooding	Group #	2		
	K/A #	W/E 15 EA1.2		
	Importance Rating	2.7		
Ability to operate and / or monitor the following as they apply to Containment Flooding: Operating behavior characteristics of the facility				

Question # 24

A LOCA is in progress with the following plant parameters:

- Containment CSF is ORANGE due to containment sump level.
- The crew has transitioned to FR-Z.2, Response to Containment Flooding.
- RWST level is 60% and lowering slowly.
- The Fire Water Jockey Pump, PKC1003, is running.
- CCW Surge Tank Level is 35% and lowering slowly.
- Containment pressure peaked at 22 psig.
- Essential Service Water Pressures are 145 psig on each train.

Based on the above indications, the reactor operator should secure and/or isolate __________ as necessary to limit containment flooding?

A. Containment Spray

B. Fire Protection Water

C. Essential Service Water

D. Component Cooling Water

Answer: D

Explanation:

Step #1 of FR-Z.2 directs the operator to try to identify the source of the leak and isolate the leak to limit containment flooding. Step #1a, list all of the external sources of water to inside containment:

☐ *Essential Service Water*
☐ *Component Cooling Water*
☐ *Reactor Makeup Water*
☐ *Fire Protection Water*

Note: Containment Spray was used as a distractor as its operating characteristics are more plausible during a LOCA and containment flooding than the reactor makeup water system which is of low volumetric flow. Containment spray normal suction source is the RWST and with this volume of water and a structurally compromised containment spray system containment flooding is plausible.

Per OTA-RK-00020 Annunciator 54A and 55A and ESW low pressure alarms are 115 psig. With ESW pressure above these values and the pressure given in the stem, the ESW system is physically intact and not the source of the leaking inside containment.

With the containment parameters present and RWST level trend, it can be determined that the containment spray system is functioning correctly (still in a standby alignment) and structurally intact. CSAS and a CISB occur at 27 psig. CISB would isolated CCW to containment so a value lower than 27 psig was chosen as the peak. Containment Spray is plausible as the candidate must know the system auto initiation parameters and determine if it is a source of the flooding based on what is given.

Per OTN-KC-00001, step 4.2.2.normal conditions are when the jockey pump is maintaining fire header pressure. IF Fire header pressure drops, then additional DG driven and electric fire pumps would start. The indication of additional pumps starting would be the characteristics of a fire water actuation or leak inside containment. But since no indication of these pumps running is given, the fire protection water system is not the source of the leak inside containment.

With the data given, there is a leak from the CCW system. (Reference ANN 51D). CCW in not intact and is the source of the leak inside containment and should be isolated per FR-Z.2 step #1c.

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Correct – See above explanation

Technical Reference(s):

1. FR-Z.2, Response to Containment Flooding, Rev 7
2. OTA-RK-00020 Addendum 54A, Rev 2
3. OTN-KC-00001, Fire Protection System, Rev 22

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #30, FR-Z(s) – Objective N: OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of FR-Z.1, FR-Z.2, and FR-Z.3

Question Source: Bank # __X __L15787__
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2009_________

Question Cognitive Level:

Memory or Fundamental Knowledge ______
Comprehension or Analysis __X___

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:
k/a match as the question stem provides a set of plant parameters / characteristics during a LOCA and the candidate must determine what is cause of containment flooding based on the parameters given and take action (i.e. operate) to stabilize the plant.

Update stem to remove 2 annunciator titles as it cued the answer per the NRC comment – added information about CCW surge tank level.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
# __1.	**TRY To Identify Unexpected Source Of Water To Sump:**	

a. CHECK system annunciators:

- Essential Service Water
- Component Cooling Water
- Reactor Makeup Water
- Fire Protection Water

b. CHECK system parameters:

- Essential Service Water:
  1) ESW Pump Discharge Flow:
     - EF FI-53
     - EF FI-54
  2) ESW Pump Discharge Pressure:
     - EF PI-1
     - EF PI-2
  3) Containment Air Cooler Condensate Collection levels and valve cycle times
- Component Cooling Water:
  1) CCW surge tank levels:
     - EG LI-1
     - EG LI-2
- Reactor Makeup Water:
  1) Reactor makeup storage tank level:
     - BL LI-1
- Fire Protection Water:
  1) Fire Protection System Outer Containment Isolation Valve position:
     - KC HIS-253

c. WHEN Containment Flooding source is identified - CLOSE valve(s) and STOP pump(s) as necessary to limit flooding.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
LOCA Cooldown - Depress. / 4	Group #	2		
	K/A #	W/E03 EK3.2		
	Importance Rating	3.4		
Knowledge of the reasons for the following responses as they apply to the (LOCA Cooldown and Depressurization): Normal, abnormal and emergency operating procedures associated with (LOCA Cooldown and Depressurization).				

Question # 25

What is the PRIMARY reason the low steamline pressure SI is blocked during the performance of ES-1.2, Post LOCA Cooldown and Depressurization?

A. To allow manual operation of SI equipment.

B. To allow the use of condenser steam dumps.

C. To unblock the steam line isolation from high rate signal.

D. To establish normal charging flow and isolate the boron injection header.

Answer: B

Explanation:
Step #9 of ES-1.2 initiates RCS Cooldown to Cold Shutdown – and after RCS pressure is less than 1970 psig, Steam Line Pressure SI is blocked in part b. Dumping steam to the condenser occurs in step 9.e. and this is the PRIMARY reason why the low team line pressure is blocked – to be able to use the steam dumps and the condenser to maneuver/cooldown the plant instead of PZR PORVs or SG ASDs or other means.

A. Incorrect – Plausible as this is the reason for resetting SI in step 1. Furthermore, if the crew is performing ES-1.2, then a LOCA and a SI have occurred. The candidate may believe that the reason this SI signal is blocked is for the operator to have manual control of the SI equipment in order to conduct a control cooldown.
B. Correct – See above explanation
C. Incorrect – while this does occur when Steam Line Pressure SI is blocked, it is not the reason. It is plausible since it does occur and the candidate may believe that the reason to have this protection available is to prevent against excessive cooldown rates when performing the cooldown.
D. Incorrect – If a SI occurs (due to low steam line pressure) a swap over from the VCT to the RWST will occur along with the realignment to the Boron injection header from the normal charging flowpath. The candidate may falsely believe the SI is block such that this "swapover" does not occur or to allow a return to this normal flowpath. The fact that these steps would be performed if the SI wasn't blocked in step #9 makes this plausible but is not the reason why the low steam line pressure SI is blocked. Procedurally, if SI was in service at step #17 and #18 of

ES-1.2 the operator would swapping back to the normal charging lineup. The only way the operator would progress to these steps is if at step #11 SI was in service. Then the operator would perform the next several steps securing unneeded ECCS pumps and flowpaths and then return to normal charging flow. If SI is in not service, the operators at step #11 would perform the RNO and go to step #19 and not have to reestablish the normal charging flowpath.

Technical Reference(s):

1. BD-ES-1.2, Post LOCA Cooldown and Depressurization basis document, Rev 7
2. ES-1.2, Post LOCA Cooldown and Depressurization, Rev 14

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #10; Objective I & O:

I. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ES-1.2, Post LOCA Cooldown and Depressurization.

O. STATE when Low Steam Line Pressure SI should be blocked.

Question Source: Bank # _X__L16256___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2005________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised question per NRC comments on cueing in the stem and making the stem similar wording as other question "… What is the PRIMARY reason". Also added to the explanations for plausibility. Reordered choices based on length.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- Shutdown margin should be monitored during RCS cooldown.
- AFTer the low steamline pressure SI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	**INITIATE RCS Cooldown To Cold Shutdown:**	
	a. CHECK RCS pressure – LESS THAN 1970 PSIG • P–11 light – LIT	a. WHEN RCS pressure is less than 1970 PSIG, THEN PERFORM Step 9.b. CONTINUE with Step 9.c.
	b. BLOCK Steamline Pressure SI: • SB HS–9 • SB HS–10	
	c. MAINTAIN cooldown rate in RCS cold legs – LESS THAN 100°F/HR	
	d. USE RHR system if in service	
	e. DUMP steam to condenser from intact SG(s): 1) CHECK condenser – AVAILABLE • C–9 interlocks – LIT • MSIVs – ANY OPEN 2) PLACE Steam Header Pressure Controller in MANUAL and ZERO OUTPUT: • AB PK–507 3) PLACE Steam Dump Select switch in STM PRESS position: • AB US–500Z 4) ADJUST Steam Header Pressure Controller in STM PRESS mode to achieve desired cooldown rate: • AB PK–507	e. DUMP steam using intact SG ASD(s).
10.	**CHECK RCS Subcooling – GREATER THAN 30°F [50°F]**	Go To Step 23.

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	1		
W/E09&E10 Natural Circ. / 4	**Group #**	2		
	K/A #	W/E 9 G2.4.45		
	Importance Rating	4.1		
Ability to prioritize and interpret the significance of each annunciator or alarm.				

Question # 26

A Reactor trip occurred due to a loss of offsite power.

- The crew is performing ES-0.2, Natural Circulation Cooldown, Step #9 "Depressurize RCS to 1920 psig".
- The following annunciators are LIT:
  - 32A, Pressurizer Level High
  - 56A, RCS SATURATE

Per ES-0.2, Natural Circulation Cooldown, what is the NEXT action the crew should take?

A. Establish Auxiliary Spray.

B. Actuate SI and return to E-0, Step #1.

C. Block SI Actuation and maintain RCS temperature and pressure within cooldown limits.

D. Minimize charging and maximize letdown to establish pressurizer level at program level.

Answer: B

Explanation:

A. Incorrect –This is step #9.b and the fact that operators are told in the stem that they are performing step #9 makes this plausible. If the candidate does not properly determine that foldout page actions apply, the candidate would determine/ remember that this action would be next.
B. Correct - foldout page action of ES-0.2 is if RCS subcooling is less than 30F, Actuate SI and return to E-0. The RCS saturate annunciator provides indication that this plant conditions exists. This is the highest priority action for the crew to take based on the information given in the stem.
C. Incorrect – These actions are step #10 and #11 of ES-0.2 but are not the next actions the crew should take as foldout page actions are applicable and are the NEXT actions the crew should perform. (i.e the foldout page are of a higher priority)
D. Incorrect – This is an action / step from Annunciator 32A, PZR Level High and is plausible as

PZR level is high and at several points in ES-0.2 the operator is directed to maintain PZR level from 25% to 74%. The setpoint for the PZR Level High annunciator is 70%.

Technical Reference(s):
1. EOP Addendum 1, Natural Circulation Verification, Rev 2
2. ES-0.2, Natural Circulation Cooldown, Rev 11
3. The following list of Annunciator Response Procedures:
    a. OTA-RK-00018, Addendum 32A, PZR High Level
    b. OTA-RK-00020, Addendum 56A, RCS Saturate

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #7, ES-0.2, ES-0.3, ES-0.4 Natural Circulation Objective: E and H:

E.DESCRIBE the Criteria and Basis for information as stated on the Foldout Page of:

1. ES-0.2.

H.OUTLINE procedural flow path including major system and equipment operation in accomplishing the goal of the following procedures:

1. ES-0.2

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised one distractor of " Establish or verify letdown is in service" and reordered the distractors based on length per NRC comments.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	**DEPRESSURIZE RCS To 1920 PSIG:**	
	a. CHECK letdown — IN SERVICE	a. TRY to establish letdown: 1) ENSURE Letdown Containment System Isolation valves are OPEN: • BG HIS—8152 • BG HIS—8160 2) OPEN RCS Letdown To Regen HX valves: • BG HIS—459 • BG HIS—460 3) PLACE Letdown HX Outlet Pressure Controller in MANUAL at 75% or greater: • BG PK—131 4) OPEN Letdown Orifice Isolation valve(s) to establish desired letdown flow: • BG HIS—8149AA (45 gpm) • BG HIS—8149BA (75 gpm) • BG HIS—8149CA (75 gpm) 5) ADJUST demand on Letdown HX Outlet Pressure Control to establish desired pressure: • BG PK—131 6) PLACE BG PK—131 in AUTO.

(Step 9. continued on next page)

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F

     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 9. (continued from previous page)

IF letdown can NOT be established,
THEN PERFORM the following:

1) USE one PZR PORV.

2) Go To Step 10. OBSERVE CAUTION prior to Step 10.

b. USE auxiliary spray:

1) OPEN Regen HX To PZR Auxiliary Spray valve:

- BG HIS-8145

2) CLOSE Regen HX To Loop Cold Leg valves:

- BG HIS-8146
- BG HIS-8147

3) CONTROL depressurization using the following:

- NCP or CCP Discharge Flow Control valve:
  - BG FK-124

    OR

  - BG FK-121
- Charging Header Back Pressure Control valve:
  - BG HC-182

FOLDOUT PAGE FOR ES–0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E–0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling – LESS THAN 30°F

     OR
   - PZR level – CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

SI actuation circuits will automatically unblock if PZR pressure rises to greater than 1970 PSIG.

10. **BLOCK SI Actuation:**

- Steamline Pressure SI:
  - SB HS-9
  - SB HS-10
- PZR Pressure SI:
  - SB HS-7
  - SB HS-8

11. **MAINTAIN Following RCS Conditions:**

- RCS pressure - AT 1920 PSIG
- PZR level - BETWEEN 25% AND 74%
- Cooldown rate in RCS cold legs -
  - LESS THAN 50°F/HR IF ALL RCS LOOPS ACTIVE

    OR

  - LESS THAN MAXIMUM ALLOWABLE LIMITS OF FIGURE 1 IF AT LEAST ONE RCS LOOP INACTIVE
- RCS temperature and pressure - WITHIN COOLDOWN LIMITS
  - Refer To EOP Addendum 2, RCS Cooldown Limitations

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F

     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12.	**MONITOR RCS Cooldown:** • Core exit TCs – LOWERING • RCS hot leg temperatures – LOWERING • RCS subcooling – RISING	

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F
     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

If at any time it is determined that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel, one of the following procedures should be used:

ES—0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS) or
ES—0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)

13. INITIATE RCS Depressurization:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. CHECK THREE CRDM Fans — RUNNING • GN HIS—41 • GN HIS—42 • GN HIS—43 • GN HIS—44	a. ENSURE THREE CRDM Fans Running, IF Less Than THREE CRDM Fans RUNNING, PERFORM the following: 1) MAINTAIN RCS subcooling greater than 130°F. 2) Go To Step 13.c.
b. MAINTAIN RCS subcooling — GREATER THAN 80°F	
c. CHECK letdown — IN SERVICE	c. PERFORM the following: 1) USE one PZR PORV. 2) Go To Step 14.

(Step 13. continued on next page)

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F
     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 13. (continued from previous page)

d. USE auxiliary spray:

1) OPEN Regen HX To PZR Auxiliary Spray valve:

- BG HIS–8145

2) CLOSE Regen HX To Loop Cold Leg valves:

- BG HIS–8146
- BG HIS–8147

3) CONTROL depressurization using the following:

- NCP or CCP Discharge Flow Control valve:
  - BG FK–124

    OR

  - BG FK–121
- Charging Header Back Pressure Control valve:
  - BG HC–182

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F

     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

14. CONTINUE RCS Cooldown And Depressurization:

a. MAINTAIN cooldown rate in RCS cold legs —

- LESS THAN 50°F/HR IF ALL RCS LOOPS ACTIVE

  OR

- LESS THAN MAXIMUM ALLOWABLE LIMITS OF FIGURE 1 IF AT LEAST ONE RCS LOOP INACTIVE

b. MAINTAIN PZR level — BETWEEN 25% AND 74%

c. MAINTAIN required RCS subcooling from table:

RNO: c. STOP depressurization and REESTABLISH subcooling.

Number Of CRDM Fans Running	Minimum Required RCS Subcooling (°F)
3	80°F
Less Than 3	130°F

d. MAINTAIN RCS temperature and pressure — WITHIN COOLDOWN LIMITS

- Refer To EOP Addendum 2, RCS Cooldown Limitations

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F

     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15.	**CHECK That Steam Void In Reactor Vessel Does NOT Exist:** • PZR level – NO UNEXPECTED LARGE VARIATIONS • RVLIS Pumps Off indication – GREATER THAN 100%	PERFORM the following: a. RAISE RCS pressure within cooldown limits to collapse potential voids in RCS: • Refer To EOP Addendum 2, RCS Cooldown Limitations. b. CONTINUE RCS cooldown. c. IF RCS depressurization must continue, THEN Go To one of the following procedures: • ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS). <u>OR</u> • ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS).
16.	**MAINTAIN Letdown Flow:** a. OPEN Letdown Orifice Isolation valve(s) as necessary: • BG HIS-8149AA (45 gpm) • BG HIS-8149BA (75 gpm) • BG HIS-8149CA (75 gpm) b. ADJUST Letdown HX Outlet Pressure Control setpoint as necessary: • BG PK-131	

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F

     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

17. MAINTAIN Required RCP Seal Injection Flow:

a. MAINTAIN RCP seal injection flow between 8 GPM and 13 GPM per RCP using Charging Header Back Pressure Control valve:

- BG HC-182

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F
     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

18. CHECK If SI Accumulators Should Be Isolated:

a. RCS pressure — LESS THAN 1000 PSIG

RNO a. WHEN RCS pressure is less than 1000 PSIG, THEN PERFORM Steps 18.b through 18.d.

CONTINUE with Step 19.

b. Locally RESTORE power to SI accumulator isolation valves:

- NG01BGF3 (EP HV-8808A)
- NG01BGF2 (EP HV-8808C)
- NG02BGF3 (EP HV-8808B)
- NG02BHF2 (EP HV-8808D)

c. CLOSE all SI Accumulator Isolation valves:

- EP HIS-8808A
- EP HIS-8808C
- EP HIS-8808B
- EP HIS-8808D

RNO c. IF any accumulator(s) can NOT be isolated, THEN OPEN associated Accumulator Vent valve(s):

- For Accumulator A, OPEN:
  - EP HIS-8950A
- For Accumulator B, OPEN:
  - EP HIS-8950B or
  - EP HIS-8950C
- For Accumulator C, OPEN:
  - EP HIS-8950D or
  - EP HIS-8950E
- For Accumulator D, OPEN:
  - EP HIS-8950F

d. Locally OPEN breakers for SI accumulator isolation valves:

- NG01BGF3 (EP HV-8808A)
- NG01BGF2 (EP HV-8808C)
- NG02BGF3 (EP HV-8808B)
- NG02BHF2 (EP HV-8808D)

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F
     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	**CHECK If Cold Overpressure System Should Be Placed In Service:**	
	a. CHECK the following: • RCS cold leg temperatures — LESS THAN 350°F AND • RCS pressure — LESS THAN 650 PSIG	a. PERFORM the following: 1) MAINTAIN RCS cold leg temperatures greater than 275°F until Cold Overpressure Protection is in service. 2) WHEN RCS temperatures are less than 350°F AND RCS pressure is less than 650 PSIG, THEN PERFORM Step 19.b. 3) CONTINUE with Step 20.
	b. PLACE Cold Overpressure Block/Arm switches in ARM position: • BB HS—8000A • BB HS—8000B	

FOLDOUT PAGE FOR ES–0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E–0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling – LESS THAN 30°F
     OR
   - PZR level – CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

20. CHECK If ECCS Pumps Should Be Locked Out:

a. RCS temperatures − LESS THAN 350°F

Response Not Obtained:

a. PERFORM the following:

1) MAINTAIN RCS cold leg temperatures greater than 275°F until ECCS pumps are locked out.

2) WHEN RCS temperatures are less than 350°F, THEN PERFORM Step 20.b.

3) CONTINUE with Step 21.

b. LOCK OUT ECCS pumps:

1) PLACE both SI Pumps in PULL−TO−LOCK:

- EM HIS−4 (SI Pump A)
- EM HIS−5 (SI Pump B)

2) PLACE one NON−operating CCP in PULL−TO−LOCK:

- BG HIS−1A (CCP A)

  OR

- BG HIS−2A (CCP B)

3) RACK OUT both SI Pump breakers:

- NB0103 (SI Pump A)
- NB0202 (SI Pump B)

4) RACK OUT NON−operating CCP breaker:

- NB0104 (CCP A)

  OR

- NB0201 (CCP B)

FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F
     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21.	**CHECK If RHR System Can Be Placed In Service:**	
	a. CHECK the following: • RCS temperature - LESS THAN 350°F <u>AND</u> • RCS pressure - LESS THAN 360 PSIG	a. Return To Step 14.
	b. PLACE RHR System in service using OTN-EJ-00001, Residual Heat Removal System	
22.	**CONTINUE RCS Cooldown To Cold Shutdown**	

CAUTION

Depressurizing the RCS before the entire RCS is less than 200øF may result in void formation in the RCS.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23.	**CONTINUE Cooldown Of Inactive Portion Of RCS:** • COOL upper head region using CRDM fans • COOL SG U-tubes by dumping steam from all SGs	

FOLDOUT PAGE FOR ES–0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E–0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling – LESS THAN 30°F

     OR
   - PZR level – CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
24.	**DETERMINE If RCS Depressurization Is Permitted:**	
	a. Entire RCS – LESS THAN 200°F	a. Return To Step 22.
	b. ENSURE THREE CRDM Fans – RUNNING • GN HIS–41 • GN HIS–42 • GN HIS–43 • GN HIS–44	b. WHEN RHR cooling has been established for at least 88 Hours, THEN PERFORM Step 24.c. Return To Step 22.
	c. Go To OTG–ZZ–00006, Plant Cooldown From Hot Standby To Cold Shutdown	

-END-

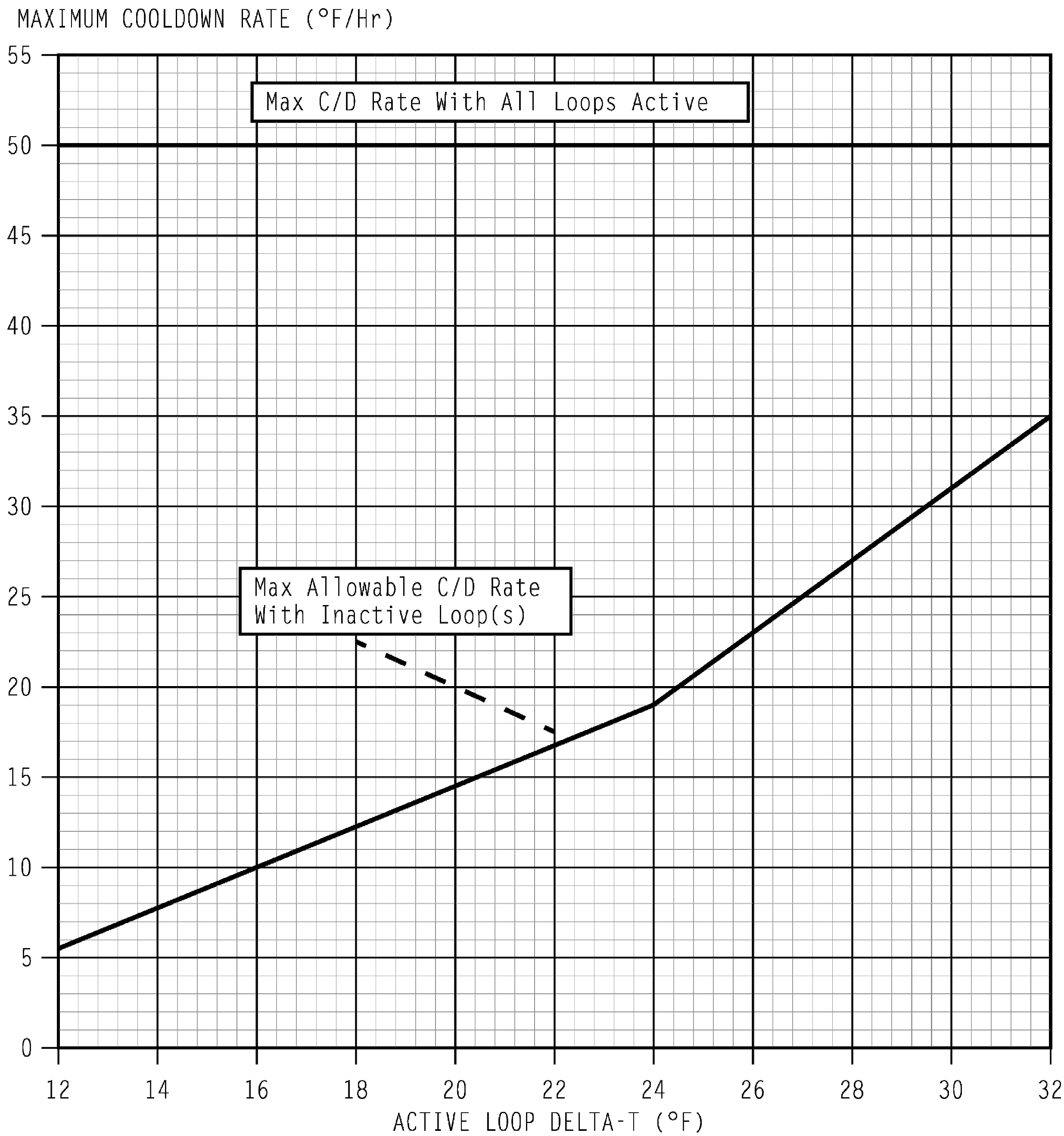
FOLDOUT PAGE FOR ES—0.2

1. **SI ACTUATION CRITERIA**
   IF either condition listed below occurs,
   THEN ACTUATE SI and Go To E—0, Reactor Trip Or Safety Injection, Step 1:
   - RCS subcooling — LESS THAN 30°F

     OR
   - PZR level — CANNOT BE MAINTAINED GREATER THAN 6%

2. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

FIGURE 1
MAXIMUM COOLDOWN RATE VERSUS ΔT



Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
RCS Overcooling - PTS / 4	Group #	2		
	K/A #	W/E08 EA2.1		
	Importance Rating	3.1		
Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.				

Question # 27

The Plant experienced a transient and reactor trip at 0730.

The crew has transitioned from the primary trip response procedure.

The following plant conditions exist at 0810:

- RCS Cold Leg Temps — A - 220°F B - 205°F C - 200°F D - 210°F
- RCS WR Pressure — 1500 psig
- Highest Core Exit TC — 800ºF
- RCS Subcooling — 20ºF
- Source Ranges — 200 cps – stable
- RCPs — OFF
- RVLIS Pumps OFF — 60%
- S/G Levels (NR) — 0%
- AFW Flow — 75,000 lbm/hr to each S/G

Which of the following procedures should be implemented FIRST?

A. FR-C.2, Response to Degraded Core Cooling

B. FR-C.1, Response to Inadequate Core Cooling

C. FR-H.1, Response to Loss of Secondary Heat Sink

D. FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

Answer: D

Explanation:

Based on the plant conditions and CSF-1, a red path on Integrity is present because there has *been a Temperature reduction of more than 100F in 1 hour and NOT all RCS cold leg values are to the right of the curve in figure 4a. This directs the operator to FR-P.1.*

Based on the Core Exit Thermal couples, RVLIS OFF, RCP status, a ORANGE path does exist on core cooling. (FR-C.2). To make all of the possible choices either RED or ORANGE path (aka RO knowledge per ES 401 page 21); FR-C.2 was chosen as the distractor as the determination between the 2 FR-C.2 or FR-C.1 as RVLIS Pumps OFF indication greater than 42%.

With AFW flow at 300,000 lbm/hr (75,000x4), FR-H.1 is not required to be implemented (AFW flow needs to be less than 285,000 lbm/hr with all SG levels less than 7% NR. FR-H.1 is plausible as since SG level is lower than 7% and AFW is close to the value at which a transition is required.

FR-C.1 is plausible as the candidate must compare the data given in the stem to the, thermocouple values and RVLIS pumps off value of 42% per CSF -1, Figure 2 to determine if a transition to FR-C.1 is required. It is wrong as RVLIS pumps off is 60% and no RED path on core cooling exists. As discussed above, an ORANGE path on core cooling does exist but since a RED path exists to FR-P.1, FR-P.1 is a higher priority and the reason why the word FIRST is in the stem.

A. Incorrect – see above explanation
B. Incorrect – see above explanation
C. Incorrect – see above explanation
D. Correct – see above explanation

Technical Reference(s):
1. CSF-1, Critical Safety Function Status Trees (CSFST), Rev 10
2. ODP-ZZ-00025, EOP/OTO User's Guide, Rev 27

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #28, FR-P.1 &2 Primary Integrity FRGs, Objective B: DESCRIBE the Symptoms and/or Entry Conditions for:
1. FR-P.1, Response To Imminent Pressurized Thermal Shock Condition.

Question Source: Bank # ___X__R13152______
Modified Bank # ____
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:
RO knowledge as it is asking for RED/Orange path information which is not SRO knowledge per ES401 Attachment 2 page 7.

Changed information in stem to support FR-c.1 as distractor also.

Figure 4
Integrity

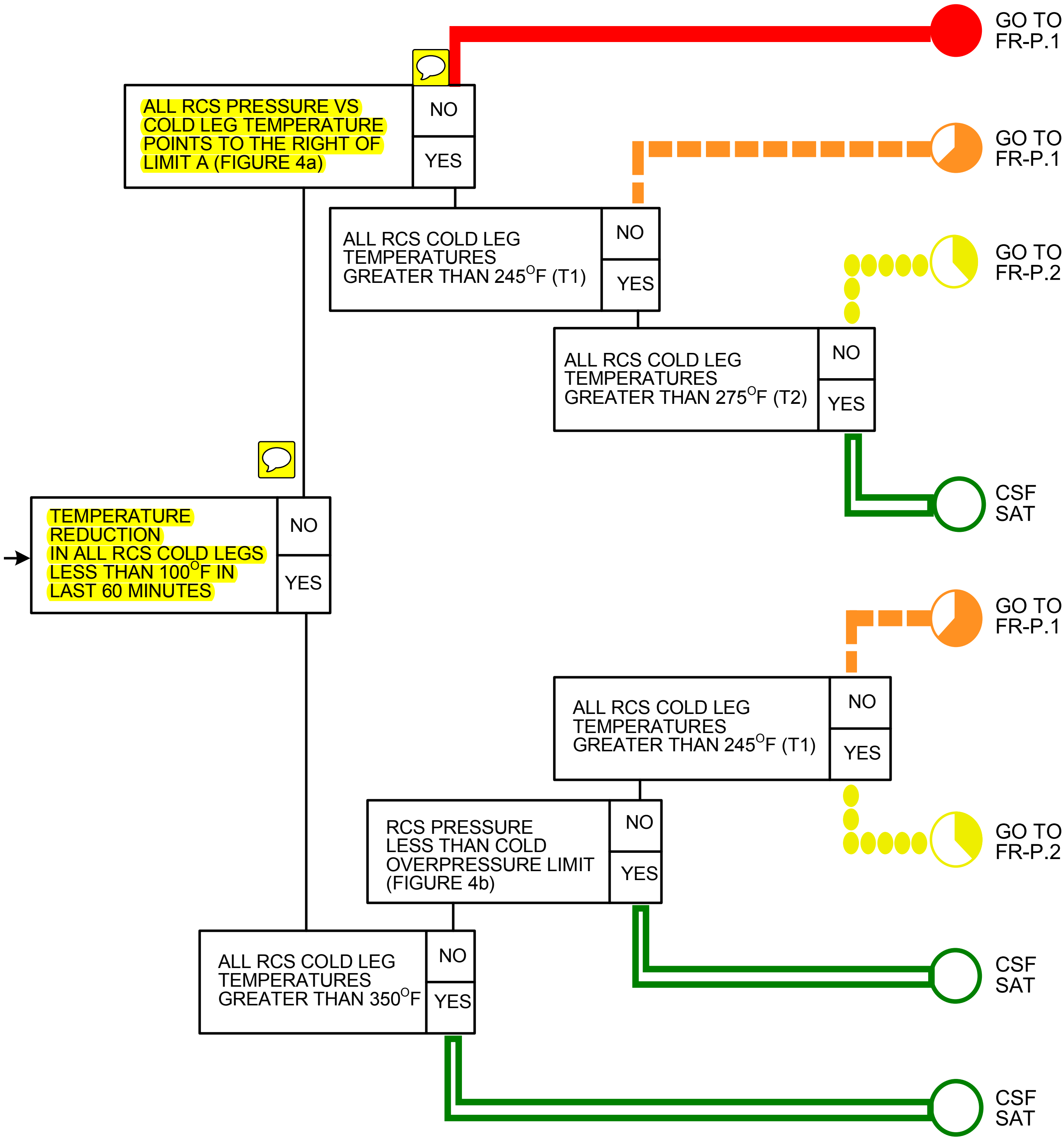
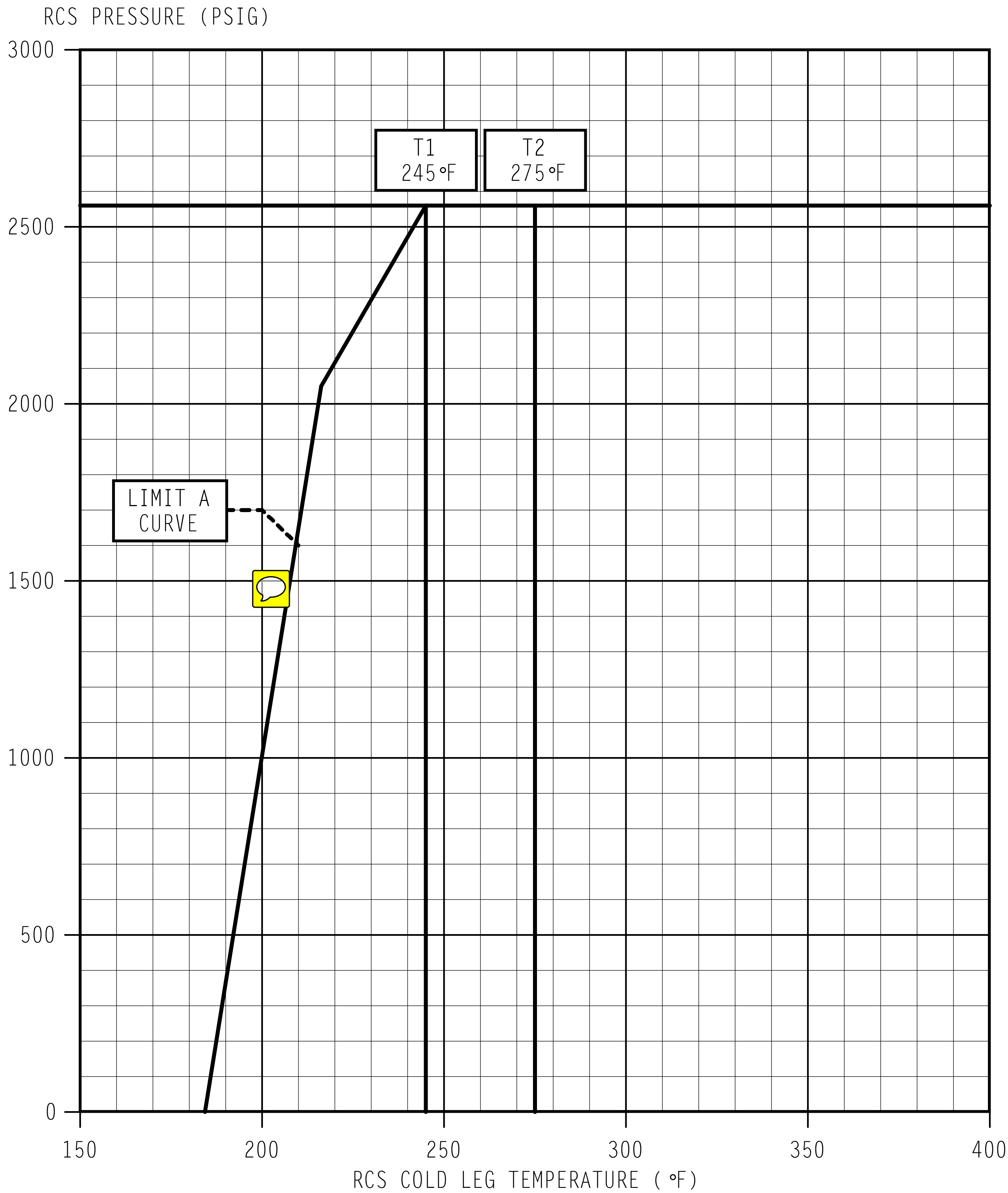


Figure 4a
Limit A Curve



Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Reactor Coolant Pump	Group #	1		
	K/A #	003 K6.04		
	Importance Rating	2.8		
Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation				

Question # 28

Reactor Power is 100%.

BG HV 8112, Seal Water Return Isolation Valve, has failed closed.

(1) RCP #1 Seal return flow will now go to the ….?

And

(2) Due to this new RCP Seal flowpath, what annunciator(s) will be LIT?

A. (1) Pressure Relief Tank
(2) 70-74F, RCP A-D Standpipe Level High

B. (1) Pressure Relief Tank
(2) 73A, RCP #2 Seal Flow High

C. (1) Reactor Coolant Drain Tank
(2) 70-74F, RCP A-D Standpipe Level High

D. (1) Reactor Coolant Drain Tank
(2) 73A, RCP #2 Seal Flow High

Answer: B

Explanation:
Per OTO-SA-00001, Attachment V, BG HV 8112 is a containment isolation valve that would close on a containment isolation signal, Phase A.

The normal seal return path through BG HV 8112 is to the seal return filters then HX and back to the VCT. If this valve closes with seal injection operating, BG 8121 will lift @150 psig and direct flow to the PRT.

The RCDT is plausible as it may be believed that since the #1 seal return flow path is isolated, there is no place for this flow to go (i.e. candidate does not understand or remember there is a relief valve in this flowpath) and all of the normally #1 seal return flow is forced up along the shaft to the #2 seal. The #2 seal leakoff is always directed to the RCDT making this a plausible choice.

Normal VCT pressure (and hence the normal seal return pressure) is @~20 psig but if this path is isolated and pressure rises to 150 psig (relief setpoint to the PRT) seal return flow (aka #1 seal leakoff flow) will go down (centrifugal pump laws). Furthermore, there will more flow directed past the #2 Seal than before the isolation valve failing closed. Per Annunciator 73A, "High flow at the #2 seal leakoff is most often accompanied by a drop in #1 seal leakoff flow (unless a #1 seal failure occurred)." Therefore Annunciator 73A will be LIT. This is supported by OTO-BB-00002 Section B which shows the symptoms for #2 Seal Leak off Flow High as RCDT level rising and Annunciator 73A or 72B in alarm

Annunciator 70 through 74F, RCP A-D Standpipe Level High, is plausible if the candidate believes that standpipe level will go up due to more flow going up along the shaft or does not understand the function of the #3 seal and standpipe.

A. Incorrect – see above explanation
B. Correct – see above explanation
C. Incorrect – see above explanation
D. Incorrect – see above explanation

Technical Reference(s):
1. OTO-SA-00001, EFSAS Verification and Restoration, Rev 39 Attachment V
2. OTO-BB-00002, RCP Off Normal, Rev 32
3. M-22BB03A, RCP Seals P&ID, Rev 9
4. M-22BG01, CVCS P&ID, Rev 32
5. OTA-RK-00022, Addendum 73A, RCP #2 Seal Flow High, Rev 0
6. OTA-RK-00022, Addendum 70-74F, RCP A-D Standpipe Level High, Rev 0

References to be provided to applicants during examination: None

Learning Objective:
T61.003B, Off Normal Operations, LP #9, OTO-BB-00002, Objective D: Given a set of plant conditions or parameters indicating a RCP Off-Normal condition, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

T61.0110, Systems, LP #9. RCS, Objective C: DRAW, LABEL and EXPLAIN a one line diagram of the RCS to include the following components: 3. Reactor Coolant Pumps (RCPs)

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:
Revised explanation per NRC comment

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Chemical and Volume Control	Group #	1		
	K/A #	004 K3.04		
	Importance Rating	3.7		
Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: RCPS				

Question # 29

Reactor Power is 100%.

A malfunction occurs causing VCT pressure to rise to 50 psig.

(1) RCP #1 Seal Leakoff flow will __(1)__

And

(2) What is the acceptable range of #1 Seal Leakoff flows?

A. (1) lower
   (2) 0.8 to 6 gpm

B. (1) lower
   (2) 8 to 13 gpm

C. (1) rise
   (2) 0.8 to 6 gpm

D. (1) rise
   (2) 8 to 13 gpm

Answer: A

Explanation: *RCP Seal arrangement and leakoff paths are as follows: #1 seal leakoff goes to the VCT through the Seal Water Heat Exchanger, and the #2 Seal leakoff goes to the RCDT. If VCT Pressure rises, there is more backpressure on the #1 Seal leakoff path and therefore #1 leakoff flow will go down and more flow will go through the #2 seal and Seal #2 leakoff.*

The distractors are plausible as the candidate may confuse the #1 and #2 leakoff paths plus understand the effect of backpressure vice a direct pressure control effect.

Per OTN-BB-00003, step 5.1.3 the normal range of ***seal injection flows*** *are 8 to 13 gpm and per the OTO-BB-00002 the normal range of #1 seal leakoff flows are 0.8 gpm to 6 gpm. 8 to 13 gpm is plausible if the candidate confuses seal injection parameters with the normal leakoff parameters.*

Note Seal leakoff values can go to 0.2 gpm during shutdown but must be checked against #1 seal dp using figure 1. 0.2 gpm was not given to prevent providing a reference that may assist with another question later in the exam.

A. Correct – see above explanation
B. Incorrect – wrong range
C. Incorrect – wrong direction
D. Incorrect – both are wrong

Technical Reference(s):
1. OTN-BG-00004, VCT Atmosphere Control, Rev 12
2. M-22BB03D, PI&D for RCS, Rev 8
3. OTO-BB-00002, RCPs Off-Normal, Rev32,

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #11 CVCS, Objective AA:
AA. OTN-BG-00004, Volume Control Tank Atmosphere Control

1. EXPLAIN the precautions and limitations and bases pertaining to each of the following:
   a. Minimum VCT pressure
   b. Maximum VCT Pressure

Question Source: Bank # __ __
Modified Bank # ___X__L13055______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 2 of 7)
RCP Seal Parameters Abnormal

NOTE

The RCP should be tripped within 5 minutes if seal leakoff flow is 6 gpm or greater OR 0.8 gpm or less with rising pump bearing OR seal injection temperatures.

# **B2**.	**CHECK No. 1 Seal Leakoff flow On All RCPs - LESS THAN 6 GPM** • BG FR-157 • BG FR-156 • BG FR-155 • BG FR-154	PERFORM ONE of the following: • IF Reactor power is greater than or equal to 48% (P-8 lit), THEN Go To Attachment D, RCP AND Reactor Trip. OR • IF Reactor power is less than 48% (P-8 extinguished), THEN Go To Attachment E, RCP Trip.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 3 of 7)
RCP Seal Parameters Abnormal

NOTE

During Low pressure operation seal leakoff flow can be 0.8 gpm or less, but should be maintained within the limits of Figure 1, No 1 Seal Normal Operating Range.

B3. **CHECK No. 1 Seal Leakoff flow On All RCPs - GREATER THAN 0.8 GPM**

- BG FR-157
- BG FR-156
- BG FR-155
- BG FR-154

RESPONSE NOT OBTAINED:

IF the affected RCP pump bearing temperatures OR seal injection temperatures are rising,
THEN Perform ONE of the following:

- IF Reactor power is greater than or equal to 48% (P-8 lit),
THEN Go To Attachment D, RCP AND Reactor Trip

OR

- IF Reactor power is less than 48% (P-8 extinguished),
THEN Go To Attachment E, RCP Trip

IF the affected RCP pump bearing temperatures AND seal injection temperatures are stable,
THEN PERFORM the following:

a. TRANSITION to Mode 3 within 6 hours using any of the following:
   - OTO-MA-00008, Rapid Load Reduction
   - OTG-ZZ-00004, Power Operation
   - OTG-ZZ-00005, Plant Shutdown 20% Power to Hot Standby

b. TRIP the affected RCP per Attachment E, RCP Trip

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C
(Page 4 of 4)
CCW To RCP Abnormal

C8.	**CHECK Both Of The Following:** • RCP Vibration - NORMAL • RCP Seal Parameters - NORMAL	PERFORM the following: a. IF RCP vibration is NOT normal, THEN Go To Attachment A, RCP High Vibration. b. IF RCP seal parameters are NOT normal, THEN Go To Attachment B, RCP Seal Parameters Abnormal.

NOTE

In Modes 1 and 2 when a RCP is stopped, the idle loop RTD channel is inoperable.

C9.	**CHECK Reactor Power - GREATER THAN 48% (P-8 lit)**	IF the plant is in Mode 1 or 2 AND any RCP is secured, THEN PERFORM OTO-BB-00004, RTD Channel Failures.
C10.	**CHECK Any RCPs - RUNNING**	PERFORM OSP-BL-00001, Rx M/U Wtr Iso Vlvs W/O RCS Loops In Operation/Mode 6 Alignment.
C11.	**REVIEW Applicable Technical Specifications:** • Refer To Attachment G, Technical Specifications	
C12.	**PERFORM Notifications Per ODP-ZZ-00001 Addendum 13, Shift Manager Communications**	
C13.	**Go To Appropriate Plant Procedure As Directed By The Shift Manager/Control Room Supervisor**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D
(Page 1 of 2)
RCP AND Reactor Trip

D1. Manually TRIP The Reactor

D2. VERIFY The Reactor Has TRIPPED

D3. TRIP The Affected RCP(s)

D4. PERFORM E-0, Reactor Trip Or Safety Injection, While Continuing With This Procedure.

D5. Check RCP A - RUNNING

RESPONSE NOT OBTAINED: PLACE Pressurizer Spray Loop 1 Controller in Manual at zero output:

- BB PK-455B for A RCP

D6. Check RCP B - RUNNING

RESPONSE NOT OBTAINED: PLACE Pressurizer Spray Loop 2 Controller in Manual at zero output:

- BB PK-455C for B RCP

D7. DEFEAT Tavg And ΔT For Idle RCS Loop:

- BB TS-412T for Tavg
- BB TS-411F for ΔT

D8. CHECK No. 1 Seal Leakoff Flow Was LESS THAN 6 GPM Prior To Securing The RCP:

- BG FR-157
- BG FR-156
- BG FR-155
- BG FR-154

RESPONSE NOT OBTAINED: WHEN the affected RCP has come to a stop (approximately 4 minutes),
THEN CLOSE #1 Seal Leakoff valve for the affected RCP:

- BB HIS-8141A (RCP A)
- BB HIS-8141B (RCP B)
- BB HIS-8141C (RCP C)
- BB HIS-8141D (RCP D)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D
(Page 2 of 2)
RCP AND Reactor Trip

D9.	**CHECK No. 1 Seal Leakoff Flow Was GREATER THAN 0.8 GPM Prior To Securing The RCP:** • BG FR-157 • BG FR-156 • BG FR-155 • BG FR-154	WHEN the affected RCP has come to a stop (approximately 4 minutes), THEN CLOSE #1 Seal Leakoff valve for the affected RCP: • BB HIS-8141A (RCP A) • BB HIS-8141B (RCP B) • BB HIS-8141C (RCP C) • BB HIS-8141D (RCP D)
D10.	**Return To Step In Effect**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 1 of 3)
RCP Trip

NOTE

Tripping an RCP with Reactor Power less than 48%:

- Will result in problems controlling the SG level in the stagnant loop. Control of SG level in the stagnant loop may be accomplished with the MFRV Bypass valve.
- May result in problems controlling the Pressurizer pressure. Control of Pressurizer pressure may be accomplished by cycling of Pressurizer heaters.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
E1.	**TRIP The Affected RCP**	
E2.	**Check RCP A - RUNNING**	PLACE Pressurizer Spray Loop 1 Controller in Manual at zero output. • BB PK-455B for A RCP
E3.	**Check RCP B - RUNNING**	PLACE Pressurizer Spray Loop 2 Controller in Manual at zero output. • BB PK-455C for B RCP
E4.	**DEFEAT Tavg And ΔT For Idle RCS Loop:** • BB TS-412T for Tavg • BB TS-411F for ΔT	
E5.	**CHECK No. 1 Seal Leakoff Flow Was LESS THAN 6 GPM Prior To Securing The RCP:** • BG FR-157 • BG FR-156 • BG FR-155 • BG FR-154	WHEN the affected RCP has come to a stop (approximately 4 minutes), THEN CLOSE #1 Seal Leakoff valve for the affected RCP: • BB HIS-8141A (RCP A) • BB HIS-8141B (RCP B) • BB HIS-8141C (RCP C) • BB HIS-8141D (RCP D)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 2 of 3)
RCP Trip

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
E6.	**CHECK No. 1 Seal Leakoff Flow Was GREATER THAN 0.8 GPM Prior To Securing The RCP:** • BG FR-157 • BG FR-156 • BG FR-155 • BG FR-154	WHEN the affected RCP has come to a stop (approximately 4 minutes), THEN CLOSE #1 Seal Leakoff valve for the affected RCP: • BB HIS-8141A (RCP A) • BB HIS-8141B (RCP B) • BB HIS-8141C (RCP C) • BB HIS-8141D (RCP D)
E7.	**CHECK Steam Generator NR Levels Within One Of The Following:** • Trending to between 45% and 55% <u>OR</u> • Between 45% and 55%	RESTORE Steam Generator NR level between 45% and 55%.
E8.	**CHECK Pressurizer Pressure Within One Of The Following:** • Trending to between 2220 psig and 2250 psig <u>OR</u> • Between 2220 psig and 2250 psig	RESTORE Pressurizer pressure between 2220 psig and 2250 psig.
E9.	**Refer To Technical Specification 3.4.4**	
E10.	**TRANSITION To Mode 3 Within 6 hours Using Any Of The Following:** • OTO-MA-00008, Rapid Load Reduction • OTG-ZZ-00004, Power Operation • OTG-ZZ-00005, Plant Shutdown 20% Power to Hot Standby	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 3 of 3)
RCP Trip

E11. Return To Step In Effect

-END-

ATTACHMENT F
(Page 1 of 3)
CCW Containment Isolation Valves

F1. PLACE Administrative Controls for any OPEN Containment Isolation CCW Bypass Valve:

- Dedicated operators must be briefed and able to CLOSE the open Ctmt Iso CCW Bypass Valve upon receipt of a valid CIS 'B' signal.
- A dedicated Control Room operator able to CLOSE the open Ctmt Iso CCW Bypass Valve or notify the local dedicated operator.
- A local dedicated operator able to CLOSE the OPEN Ctmt Iso CCW Bypass Valve.
- The local dedicated operator is in communication with the Control Room.
- The local dedicated operator is stationed near (in a low dose area if possible), the OPEN Ctmt Iso CCW Bypass Valve.

ATTACHMENT F
(Page 2 of 3)
CCW Containment Isolation Valves

F2. Use The Tables Below For Additional Containment Isolation CCW Valves Information:

- EGHV0058 (EG HIS-58) [PEN 74] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0127	EG HIS-127	EG HIS-127A	MCB and Local at EGHV0127

- EGHV0059 (EG HIS-59) [PEN 75] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0131	EG HIS-131	EG HIS-131A	MCB and Local at EGHV0131

- EGHV0060 (EG HIS-60) [PEN 75] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0130	EG HIS-130	EG HIS-130A	MCB and Local at EGHV0059

If inner containment valve EGHV0060 has failed CLOSED, allowing the CCW flowpath through EGHV0130 and EGHV0059, station Operator near EGHV0059 to isolate the flowpath. Both valves are powered from Separation Group 1.

(Step F2. continued on next page)

ATTACHMENT F
(Page 3 of 3)
CCW Containment Isolation Valves

Step F2. (continued from previous page)

- EGHV0061 (EG HIS-61) [PEN 76]
{CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0133	EG HIS-133	EG HIS-133A	MCB and Local at EGHV0133

- EGHV0062 (EG HIS-62) [PEN 76]
{CISB and EGFSH0062}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0132	EG HIS-132	EG HIS-132A	MCB and Local at EGHV0061

If inner containment valve EGHV0062 has failed CLOSED, allowing the CCW flowpath through EGHV0132 and EGHV0061, station Local operator near EGHV0061 to isolate the flowpath. Both valves are powered from Separation Group 1.

With EGHV0132 OPEN, the dedicated Control Room Operator is required to CLOSE EGHV0132 and ENSURE RCP thermal barrier isolations BB HIS-13, BB HIS-14, BB HIS-15, and BB HIS-16 are CLOSED during a high flow condition, as indicated by MCB Annunciator 74C, RCP THERM BAR CCW FLOW.

- EGHV0071 (EG HIS-71) [PEN 74]
{CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0126	EG HIS-126	EG HIS-126A	MCB and Local at EGHV0126

-END-

ATTACHMENT G
(Page 1 of 1)
Technical Specifications

G1. Refer To The Following:

- Technical Specification 3.3.1, Table 3.3.1-1 Item 6
- Technical Specification 3.3.1, Table 3.3.1-1 Item 7
- Technical Specification 3.3.1, Table 3.3.1-1 Item 10
- Technical Specification 3.3.1, Table 3.3.1-1 Item 12
- Technical Specification 3.3.1, Table 3.3.1-1 Item 13
- Technical Specification 3.3.1, Table 3.3.1-1 Item 14.c
- Technical Specification 3.3.2, Table 3.3.2-1 Item 5.e
- Technical Specification 3.3.2, Table 3.3.2-1 Item 6.d
- Technical Specification 3.3.9
- Technical Specification 3.4.4
- Technical Specification 3.4.5
- Technical Specification 3.6.3

-END-

1 **ID: L13055** **Points: 1.00**

<QQ 13055(1410)><<Given the following conditions:

- RCS Temperature is 155°F
- RCS Pressure is 330 psig
- RCP #1 seal leakoff flow is 0.82 gpm

If VCT pressure rises to 60 psig due to a regulator failure, RCP #1 seal leakoff flow will ___________ and RCP #2 seal leakoff flow will ____________.>>

A. <QQ 13055(1480:0)><<rise ; rise>>

B. <QQ 13055(1480:1)><<rise ; lower>>

C. <QQ 13055(1482)><<lower ; rise>>

D. <QQ 13055(1480:2)><<lower ; lower>>

Answer: <QQ 13055 (1419) ><<C >>

Question 1 Info	
Question Type:	<QQ 13055(1401)><<Multiple Choice>>
Status:	<QQ 13055(1405)><<Active>>
Always select on test?	<QQ 13055(1406)><<No>>
Authorized for practice?	<QQ 13055(1429)><<No>>
Points:	<QQ 13055(1441)><<1.00>>
Time to Complete:	<QQ 13055(1408)><<3>>
Difficulty:	<QQ 13055(1407)><<3.00>>
System ID:	<QQ 13055(1445)><<13055>>
User-Defined ID:	<QQ 13055(1404)><<L13055>>
Cross Reference Number:	
Topic:	<QQ 13055(1400)><<L13055 BG VCT pressure rises to 60 psig due to a regulator failure>>
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	

Question 1 History	
Exam Appearances:	<QQ 13055(1449)><<2>>
Student Encounters:	<QQ 13055(1448)><<16>>
Answered Right:	<QQ 13055(1452)><<14>>
Answered Wrong:	<QQ 13055(1453)><<2>>
Partially Correct:	<QQ 13055(1459)><<0>>
Answer Invalid:	<QQ 13055(1455)><<0>>
Unanswered:	<QQ 13055(1454)><<0>>
Ignore Response:	<QQ 13055(1460)><<0>>
Avg Points Awarded:	<QQ 13055(1450)><<0.88>>
... As % of Point Value:	88
Standard Deviation:	<QQ 13055(1456)><<0.34>>

Question 1 Table-Item Links

<TB 5818(1301)><<OPS Systems>>

<TB 5835(1305)><<BG, Chemical & Volume Control>>

<TB 5972(1301)><<OPS Question Catagory>>

<TB 5974(1305)><<LO Initial, Closed Book>>

<TB 5975(1305)><<LO Requalification, Open Book>>

Associated objective(s):

<OB 16031(1101)><< B DESCRIBE the purpose, operation and interlocks for the following CVCS components:
1.Letdown Isolation Valves
2.Regenerative Heat Exchanger (Regen Hx)
3.Orifice Isolation Valves
4.Letdown Orifices
5.Letdown Containment Isolation Valves
6.Letdown Reheat Divert Valve (TCV381B)
7.Letdown Hx
8.Letdown Hx Outlet Pressure Control Valve (PCV)
9.CVCS Demineralizer Inlet Divert Valve
10.CVCS Demineralizers
11.Reactor Coolant Filter
12.Volume Control Tank (VCT) Inlet Divert Valve
13.VCT
14.VCT Outlet Valves
15.Charging Pump Suction Valves
16.Normal Charging Pump (NCP)
17.Centrifugal Charging Pumps (CCPs)
18.CCP Flow Control Valve (FCV)
19.Charging Header FCV
20.Charging Containment Isolation Valves
21.Charging Return Flow Paths
a.Normal Charging
b.Alternate Charging
c.Auxiliary Pressurizer Spray
22.Seal Injection Filters
23.Seal Injection Isolation Valves
24.Seal Water Return Header Containment Isolation Valves
25.Seal Water Heat Exchanger
26.Excess Letdown Isolation Valves
27.Excess LD Heat Exchanger
28.Excess LD Flow Control Valve
29.Excess LD Divert Valve
30.Boron Concentration Measurement System
31.CCP Miniflow Isolation Valves>>

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Residual Heat Removal	Group #	1		
	K/A #	00005 K3.01		
	Importance Rating	3.9		
Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: RCS				

Question # 30

(REFERENCE PROVIDED)

The Plant is preparing to start up from a refueling outage with the following conditions:

- Prior to the refueling outage, the Reactor was online for 300 days.
- The Reactor was shutdown 28 days ago.
- Core alterations have been completed.

Currently;

- The Plant is in Mode 5.
- The RCS is drained to 50".
- RCS Temperature is 140°F.

A Loss of All RHR occurs.

What is the estimated Time to Boil?

A. 23 minutes

B. 29 minutes

C. 40 minutes

D. 51 minutes

Answer: D

Explanation:
Per OTG-ZZ-00007 step 3.2.3, Reduced Inventory is defined as RCS Level lower than three feet below the Reactor Vessel Flange (less than 64.0 inches), with Fuel in the Reactor Vessel. Per OOA-BB-00003, MidLoop is defined as 14.5 inches.

OTO-EJ-00003, Step #4 – Determine time to boil based on Existing Conditions directs the candidate to use the appropriate figure based on the definition of cold core (fresh fuel) or hot core and RCS inventory – mid loop or reduced inventory.

Using Figure 4 of OTO-EJ-00003, the time to boil is 51 minutes since the RCS inventory is at reduced inventory not midloop conditions for a cold core. The distractor of 23 minutes is if the mid loop, Figure 1, is used. The distractor of 29 minutes is if Figure 2 is used. The distractor of 40 minutes is if Figure 3 is used.

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Correct - See above explanation

Technical Reference(s):
1. OTG-ZZ-00007, Refueling Preparation, Performance and Recovery, Rev 38
2. OOA-BB-00003, Refuel Level Indications, Rev 13
3. OTO-EJ-00003, Loss of RHR while at Reduced Inventory or mid loop conditions, Rev 9

References to be provided to applicants during examination:
1. OTO-EJ-00003, Loss of RHR while at Reduced Inventory or mid loop conditions, Rev 9 Figures 1 through 4

Learning Objective: T61.003B Off-normal Operations, LP #B-62 Objective H: STATE major action categories and symptoms/entry conditions for OTO-EJ-00003, Loss of RHR while operating at reduced inventory or mid loop.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised question per NRC comment making the use of cold core graphs correct.

Figure 1
Mid-Loop Time To Boil (Hot Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

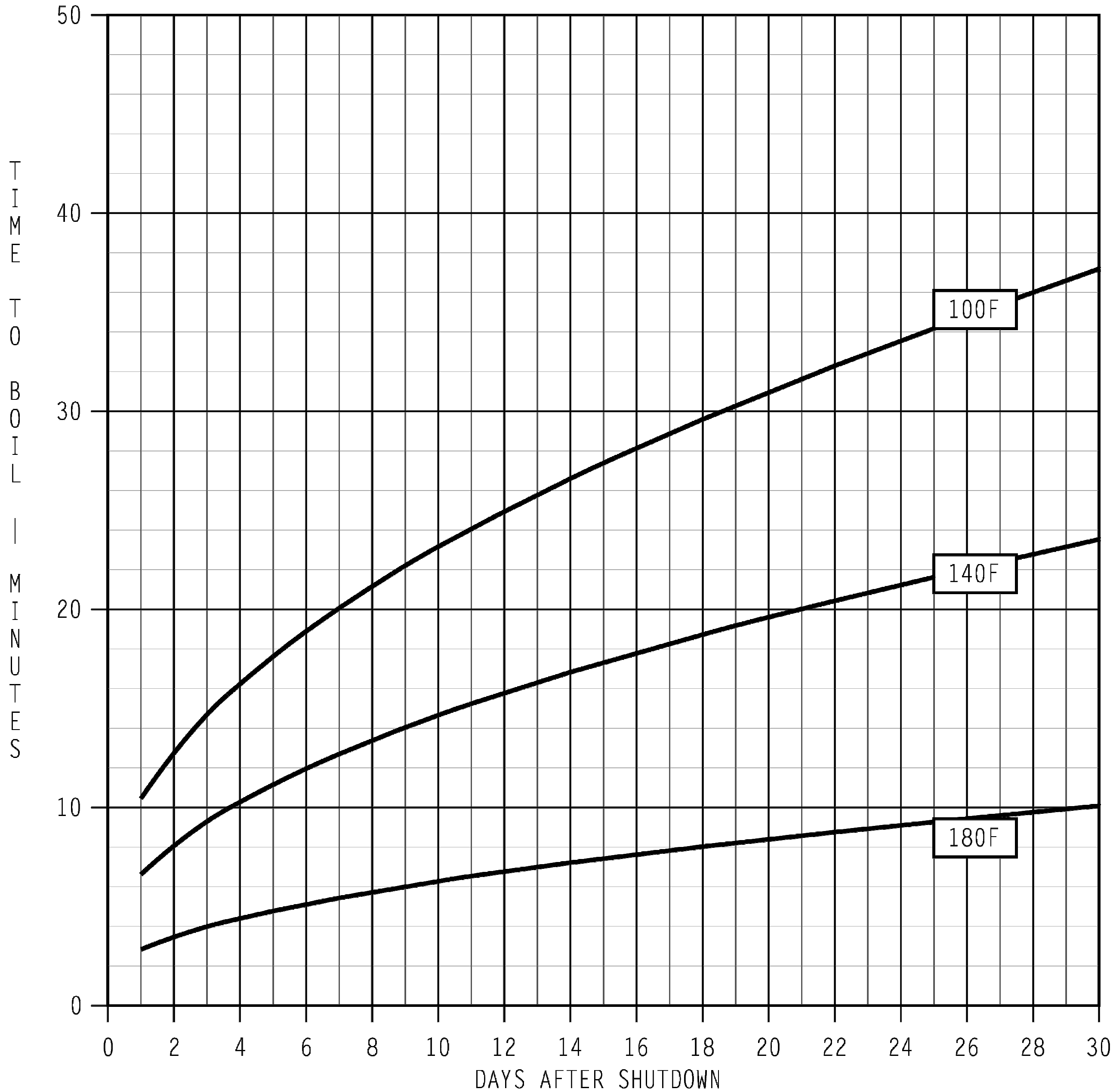


Figure 2
Reduced Inventory Time To Boil (Hot Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

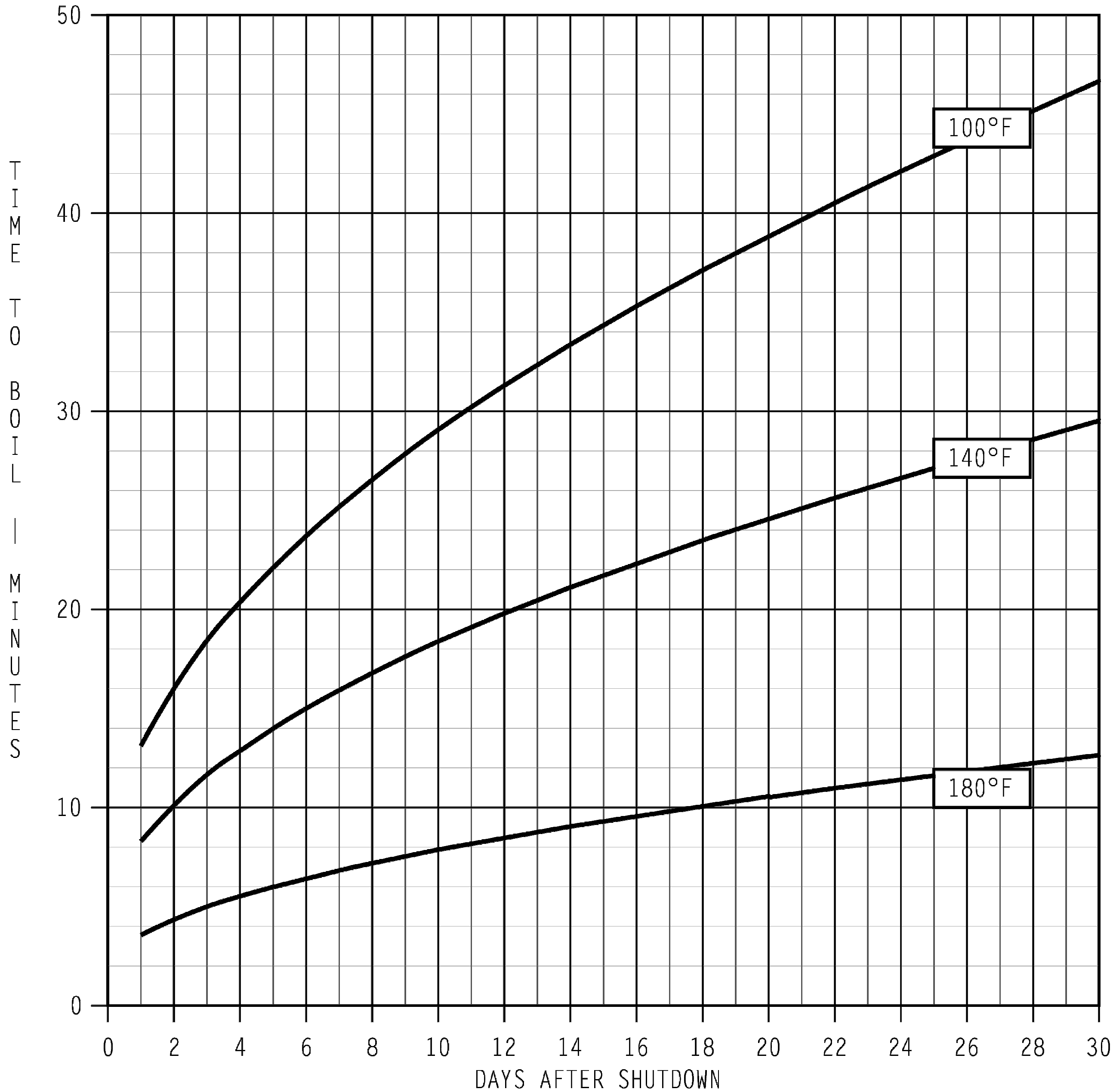


Figure 3
Mid-Loop Time To Boil (Cold Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

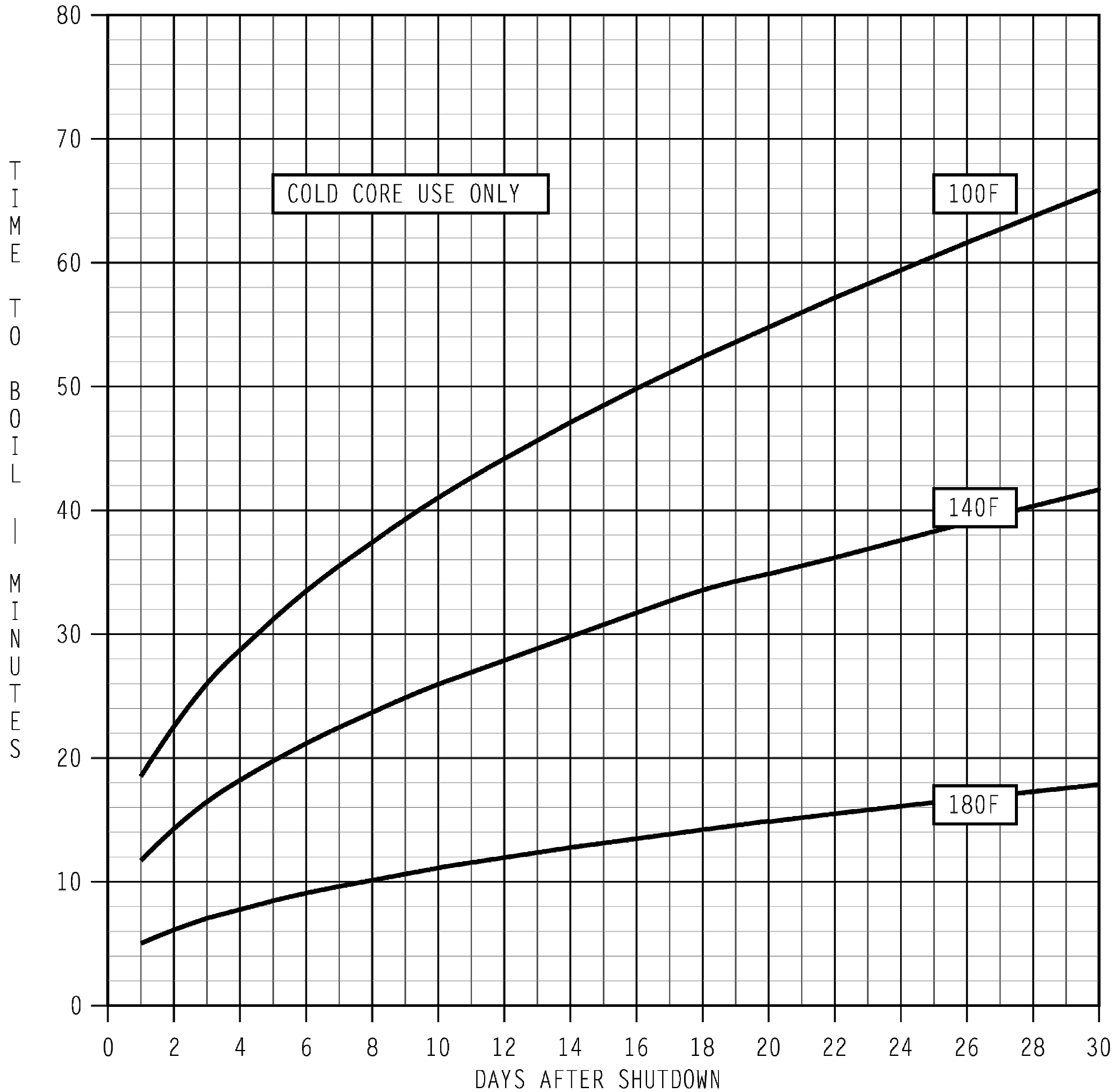


Figure 4
Reduced Inventory Time To Boil (Cold Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

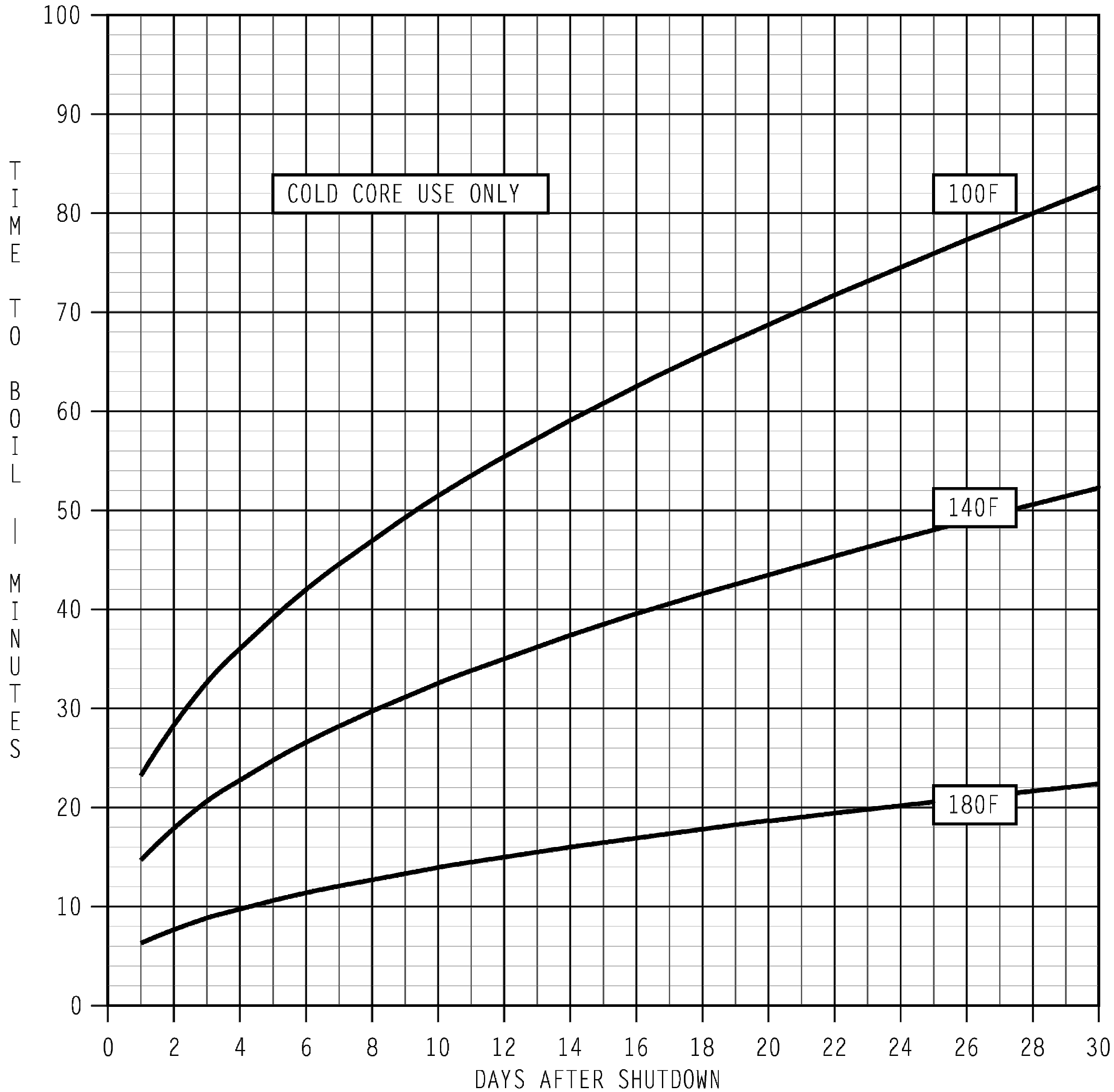


Figure 1
Mid-Loop Time To Boil (Hot Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

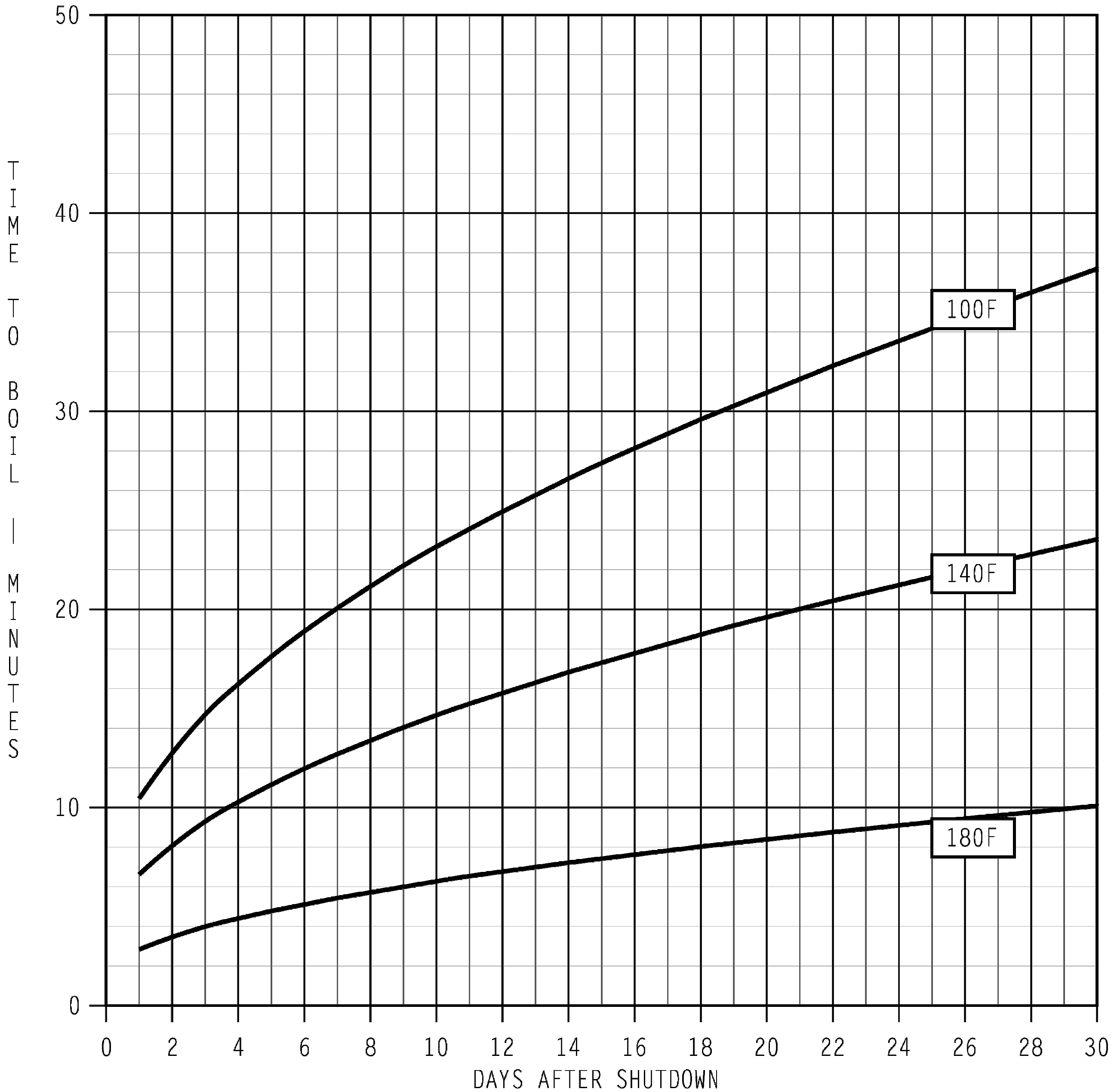


Figure 2
Reduced Inventory Time To Boil (Hot Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

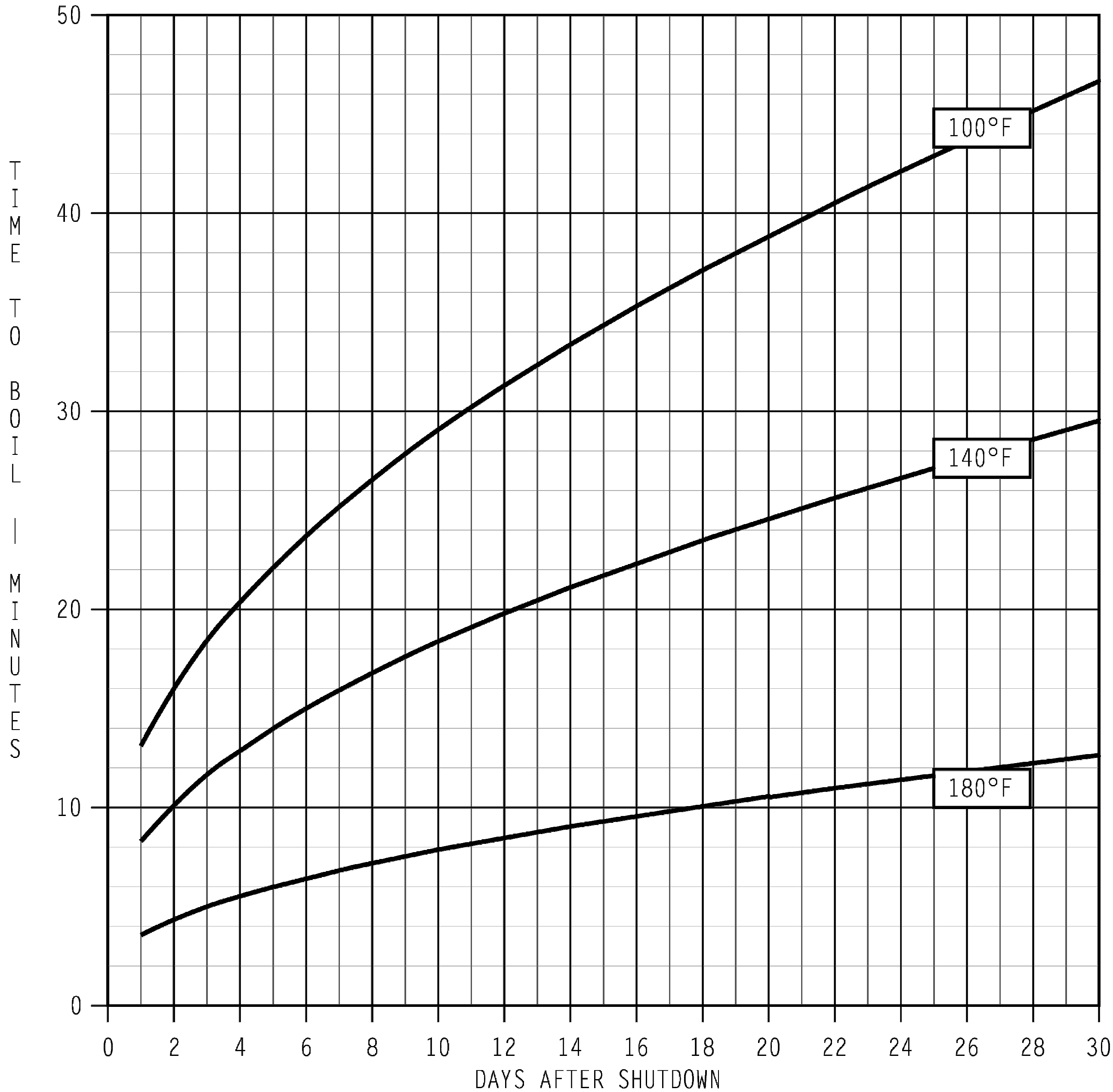


Figure 3
Mid-Loop Time To Boil (Cold Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

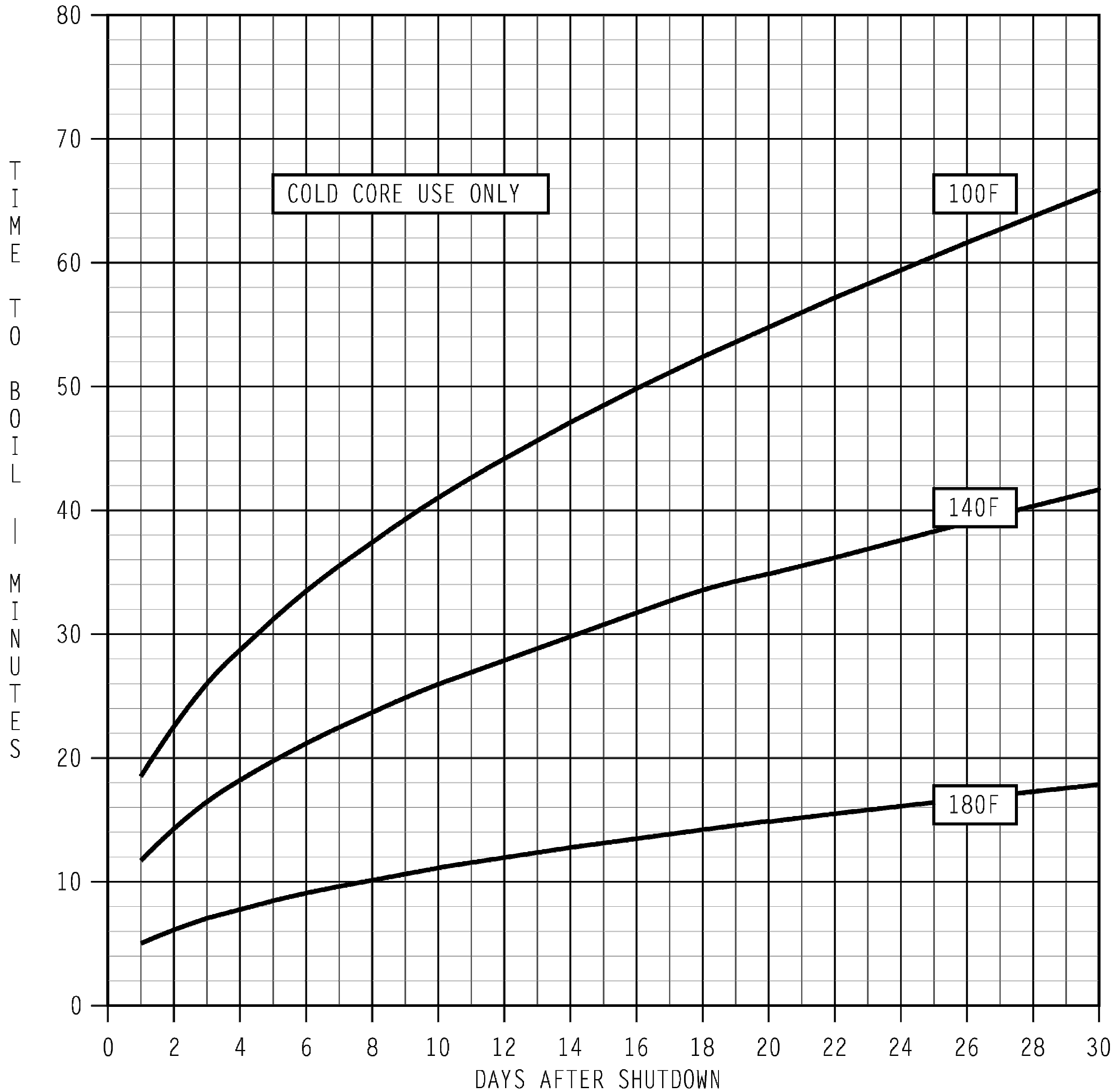
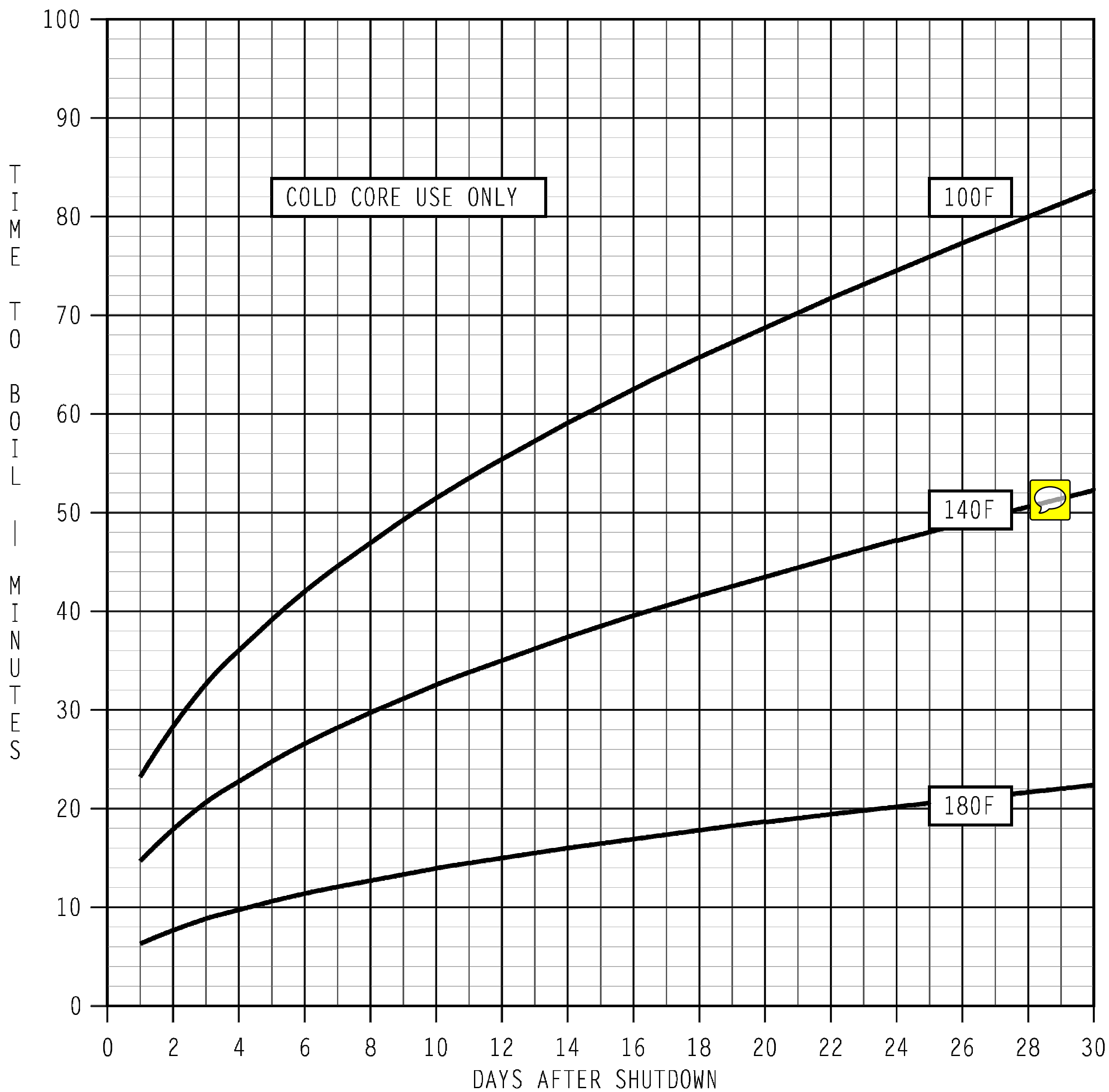


Figure 4
Reduced Inventory Time To Boil (Cold Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.



Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Residual Heat Removal	**Group #**	1		
	K/A #	005 A4.03		
	Importance Rating	2.8		
Ability to manually operate and/or monitor in the control room: RHR temperature, PZR heaters and flow, and nitrogen				

Question # 31

The Plant is in MODE 4.

- The crew has placed the “A” RHR Train in service for RCS Cooldown.
- The RCS Cooldown Rate is 98ºF/hr.

Per OTN-EJ-00001 Addendum 3, Placing A RHR Train in Service for RCS Cooldown, to lower the RCS cooldown rate, the reactor operator should …?

A. Throttle CLOSE EG HV0101, CCW to RHR HX A Valve.

B. Throttle OPEN the RHR Pump A Miniflow Valve, EJ FCV-610.

C. Throttle CLOSE EJ HCV-606, RHR Heat Exchanger Outlet Valve.

D. Throttle OPEN the RHR Heat Exchanger Bypass Valve, EJ FCV-618.

Answer: C

Explanation: *Per Step #5.3.5, to increase the Cooldown rate the reactor operator will throttle close EJ FK-618. Specifically “IF desired to increase cooldown rate:*
a. PLACE EJ FK-618, RHR HX A BYPASS CTRL, in MAN.
b. THROTTLE CLOSED EJ FK-618, RHR HX A BYPASS CTRL.”

Per step #5.3.11, MAINTAIN RCS temperature per the following:
a. ENSURE RCS temperature specified by SM/CRS is ATTAINED.
b. Using EJ HIC-606, RHR HX A FLOW CTRL, THROTTLE EJHCV0606 as necessary to maintain RCS temperature.

A. Incorrect – the procedure does not direct this action but plausible as less CCW cooling water flow would lower the heat transfer / cooling in the RHR HX.
B. Incorrect – this action is not directed in the procedure but is plausible if it is believed the min flow taps off prior to the HX and opening it would lower flow through the HX. Another valve going in the open direction was needed to maintain plausible distractors i.e. more than one “throttle open” in the possible answers.

C. Correct, step 5.3.11 provides direct to manipulate EJ HCV 606 as necessary for maintain an RCS or lower the RCS cooldown rate. Additionally, a Caution prior to step 5.3.4 states "The desired cooldown rate can be established and controlled more easily if EJ HIC-606 is slowly manipulated."
D. Incorrect – Manipulation of EJ-FCV-618 is listed step 5.3.5b but only if it is desired to increase the plant cooldown rate which is opposite than the stem asks. This action is not directed in the procedure to lower cooldown rate put is plausible as RHR flow would be bypassing the RHR HX.

Technical Reference(s):

1. OTN-EJ-00001, Addendum 3, Placing A RHR Train in Service for RCS Cooldown, Rev 22

References to be provided to applicants during examination: None

Learning Objective: T61.0110, systems, LP #&, RHR System, Objective J: IDENTIFY the RHR System Main Control Board (MCB) controls, alarms, and indications and DESCRIBE how each is used to predict, monitor, or control changes in the RHR System.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(14)

Comments:

5.3. Initiate RCS Cooldown with A RHR Train

5.3.1. ENSURE Section 5.2 is complete.

5.3.2. IF NG01ACF6, FDR BKR TO EJFCV0610 A RHR PMP MINI FLOW RECIRC FCV, was NOT opened in Step 5.1.17.a, Go To Step 5.3.11.

NOTE

The next step restores power to EJFCV0610.

5.3.3. AFTER A RHR pump has been running for approximately one hour, PERFORM the following: [Ref: 6.2.12]

a. CLOSE NG01ACF6, FDR BKR TO EJFCV0610 A RHR PMP MINI FLOW RECIRC FCV

b. ENSURE EJFCV0610, A RHR PMP MINI FLOW RECIRC FLOW CTRL VLV, valve position is appropriate for pump flow rate per RHR System Data provided in OTN-EJ-00001, Residual Heat Removal System.

CAUTION

Excessive cooldown rate may cause the Technical Specification limits to be exceeded.

The desired cooldown rate can be established and controlled more easily if EJ HIC-606 is slowly manipulated.

5.3.4. Using EJ HIC-606, RHR HX A FLOW CTRL, THROTTLE OPEN EJHCV0606 and MAINTAIN RCS Cooldown up to 100° per hour.

NOTE

With two RCPs in operation it may be necessary to fully close EJ FK-618, RHR HX A BYPASS CTRL, in order to maintain maximum cooldown rate.

5.3.5. IF desired to increase cooldown rate:

a. PLACE EJ FK-618, RHR HX A BYPASS CTRL, in MAN.

b. THROTTLE CLOSED EJ FK-618, RHR HX A BYPASS CTRL.

5.3.6. IF required, SECURE Fuel Pool Cooling by performing the following:

a. Using EC HIS-11, SFP HX A CCW OUTLET VLV, ENSURE ECHV0011 is CLOSED.

b. Using EC HIS-27, SFP COOL PUMP A, STOP PEC01A.

c. IF Spent Fuel Pool temperature approaches value listed in Curve Book, Table 8-8b, RESTORE Fuel Pool Cooling to service per OTN-EC-00001, Fuel Pool Cooling and Cleanup System.

5.3.7. RESUME logging requirements of OSP-BB-00007, RCS Heatup and Cooldown Limitations.

5.3.8. IF BOTH of the following conditions exist:

- performing a forced circulation cooldown

  - AND -

- a higher cooldown rate is desired,

SECURE unneeded RCPs per OTG-ZZ-00006, Plant Heatup Cold Shutdown To Hot Standby.

5.3.9. As directed by SM/CRS, SECURE ONE of the following:

- atmospheric dumps
- condenser dumps

5.3.10. IF EJ FK-618, RHR HX A BYPASS CTRL, is NOT in AUTO, PERFORM the following:

a. Manually SET EJ FK-618, RHR HX A BYPASS CTRL, to approximately 1.7.

b. PLACE EJ FK-618, RHR HX A BYPASS CTRL, in AUTO.

c. ENSURE EJ FK-618, RHR HX A BYPASS CTRL, is controlling flow at 2000 gpm to 2500 gpm as indicated on EJ FI-618, RHR TO ACC INJ LOOPS 1 & 2 FLOW. [Ref: 6.2.11]

5.3.11. MAINTAIN RCS temperature per the following:

a. ENSURE RCS temperature specified by SM/CRS is ATTAINED.

b. Using EJ HIC-606, RHR HX A FLOW CTRL, THROTTLE EJHCV0606 as necessary to maintain RCS temperature.

-END OF SECTION-

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Emergency Core Cooling	Group #	1		
	K/A #	006 K5.08		
	Importance Rating	2.9		
Knowledge of the operational implications of the following concepts as they apply to ECCS: Operation of pumps in parallel				

Question # 32

During a safety injection, the SI pumps, PEM01A&B, operate in _____(1)_____ with the CCP Pumps, PBG05A&B, to provide flow to the RCS.

Per Technical Specifications, the RWST is required to have a MINIMUM of ____(2)_____ gallons of borated water to be operable.

A. (1) series
   (2) 281,000

B. (1) series
   (2) 394,000

C. (1) parallel
   (2) 281,000

D. (1) parallel
   (2) 394,000

Answer: D

Explanation:
During a safety injection, the SI and CCP pumps will inject in parallel flow path to the RCS cold legs. This can be seen on M-22BB01 as the CCP pumps inject through the SIS Boron Injection header (from print M-22EM02). The SI Pumps inject using the SI Accumulator cold leg injection taps as seen from print M-22EP01. Therefore these ECCS pumps inject in parallel flow paths into the RCS cold legs. Series is plausible as the RHR pumps and SI pumps (and CCPs) are in series when placed in Cold leg recir and Hot leg recirc alignments per ES-1.3 and ES-1.4. Additionally, Series is also plausible if the candidate falsely believes that due to the different shutoff heads of the pumps (SI ~1700 and CCPs ~2400 psig) they are designed to run in a series alignment with SI pumps acting as a booster pump to the CCPs.

Per Technical Specification 3.5.4 the RWST shall be operable in Modes 1-4 (above the line and RO knowledge). The SRs for this Technical Specifications contain the requirements for RWST operability. Per SR 3.5.4.2, the RWST borated water volume is greater than or equal to 394,000 gallons. The distractor 281,000 gallons is the volume required for the CST per T.S. 3.7.6 and SR 3.7.6.1.

A. Incorrect – both are wrong
B. Incorrect – wrong system configuration
C. Incorrect – wrong volume
D. Correct

Technical Reference(s):

1. M-22EM02 P&ID, High Pressure Coolant Injection System, Rev 23
2. M-22BB01, P&ID, RCS, Rev 31
3. M-22EP01, P&ID, Accumulator Safety Injection, Rev 18
4. FSAR Section 6.3, Emergency Core Cooling System
5. Technical Specification 3.5.4, RWST

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #56, ECCS, Objective G: DRAW and/or LABEL the normal ECCS lineup and flowpath for Cold Leg Injection.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam __N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(8)

Comments:

Revised part 2 of the question and the explanation for part 1 per NRC comments

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1	Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1	Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	-------------------------------- NOTE --------------------------- Only required to be performed when ambient air temperature is < 37°F or > 100°F. -- Verify RWST borated water temperature is ≥ 37°F and ≤ 100°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is ≥ 394,000 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2350 ppm and ≤ 2500 ppm.	In accordance with the Surveillance Frequency Control Program

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Pressurizer Relief/Quench Tank	Group #	1		
	K/A #	007 A1.03		
	Importance Rating	2.6		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Monitoring quench tank temperature				

Question # 33

Annunciator 34D, Pressurizer Relief Tank (PRT) Temperature High, has just alarmed.

PRT Temperature is rising at 5°F/hr.

(1) How long before the PRT reaches its design temperature setpoint of 200°F?

And

(2) What method of PRT Cooling is more effective?

A. (1) 17 hours
   (2) Spray Cooling

B. (1) 17 hours
   (2) RCDT Heat Exchanger

C. (1) 22 hours
   (2) Spray Cooling

D. (1) 22 hours
   (2) RCDT Heat Exchanger

Answer: A

Explanation:

Annunciator 34D alarms when the PRT temperature is 115F. Per the FSAR section 5.4.11, the design temperature of the PRT is 200F. At a rate of 5F/hr it would take 17 hours to go from 115F to 200F. The distractor of 22 hours is using a the ultimate heat sink temperature limit of 90F per Tech Spec SR 3.7.9.2. 200F-90F = 110F / 5F/hr = 22 hours

OTN-BB-0004, Section 5.8 – Note prior to step #1 Cooling time required following a design maximum discharge is approximately one (1) hour by spraying or eight (8) hours by RCDT heat *exchanger. Therefore, Spray Cooling is more effective and RCDT HX is plausible as it is a method in the procedure to cool the PRT.*

A. Correct
B. Incorrect – wrong method
C. Incorrect – wrong time
D. Incorrect – wrong method and wrong time

Technical Reference(s):

1. OTN-BB-00004, Pressurizer Relief Tank, Rev 37
2. OTA-RK-00018, Addendum 34D, PRT Temp High, Rev 0
3. FSAR 5.4.11 Pressurizer Relief Discharge System, page 5.4-42

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #9, RCS, Objective B: DESCRIBE the purpose and operation of the following RCS components to include interlocks, controller operations and power supply: 9. Pressurizer Relief Tank (PRT)

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

5.8. PRT Cooling by Spraying

NOTE

Cooling time required following a design maximum discharge is approximately one (1) hour by spraying or eight (8) hours by RCDT heat exchanger.

5.8.1. IF PRT is going to be cooled by recirculation through RCDT heat exchanger use Section 5.9.

5.8.2. *Radwaste* - ENSURE RCDT Pump switches are in PULL TO LOCK: (HB115)

- HBHS/1003A, RCDT PMP A HAND SW
- HBHS/1003B, RCDT PMP B HAND SW

5.8.3. Using BB HIS-8045, REACTOR M/U WTR TO PRT, OPEN BBHV8045. (RL021)

5.8.4. WHEN PRT Level increases to 81%, CLOSE BBHV8045.

5.8.5. *Radwaste* - CLOSE HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.8.6. *Radwaste* - CLOSE HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.8.7. Using BB HIS-8031, PRT TO RCDT, OPEN BBHV8031. (RL021)

5.8.8. Using HB HIS-7176, RCDT HX OUTLET INNER CTMT ISO VLV, ENSURE OPEN HBHV7176. (RL021)

5.8.9. Using HB HIS-7136, RCDT HX OUTLET OUTER CTMT ISO VLV, ENSURE OPEN HBHV7136. (RL021)

5.8.10. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.8.11. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.8.12. ENSURE BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, is set to control between 3 and 4 psig. (1322)

5.8.13. ESTABLISH communications between Radwaste Control Room, and Main Control Room.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Component Cooling Water	**Group #**	1		
	K/A #	00008 A3.04		
	Importance Rating	2.9		
Ability to monitor automatic operation of the CCWS, including: Requirements on and for the CCWS for different conditions of the power plant				

Question # 34

'D' CCW pump is supplying 'B' safety loop and all non-safety CCW loads.

A Safety Injection occurs.

What is the automatic response of the 'A' CCW Train?

A. BOTH 'A' and ‘C’ CCW pumps START at the 0 second step of the LOCA sequencer.

B. 'A' CCW pump STARTS at the 5 second step of the LOCA sequencer.

C. 'C' CCW pump STARTS at the 5 second step of the LOCA sequencer.

D. BOTH 'A' and ‘C’ CCW pumps START at the 10 second step of the LOCA sequencer.

Answer: B

Explanation:
On a LOCA Sequencer, the A CCW pump will start @5 seconds and if this pump fails to start the C CCW pump will start at the 10 second interval. The combination of the 'A' CCW pump and the 'C' CCW pump makeup the A CCW Train.

A. Incorrect – when referring to the LOCA sequencer, the first step (at time = 0 seconds) of the sequencer is to start that trains CCP and block CCW pumps unsequenced auto-start signal (Per E-22NF01). *This is a plausible distractor as the LOCA sequencer does send a signal to the pumps but it is not a start signal, it is a blocking signal such that the pumps do not start at times in which the EDG cannot assume the load. Instead of using instantaneously or immediately in this distractor, it was decided to use the t=0 step terminology as it is applicable to the technical topic, how operators speak when referring to the LOCA sequencer, and similar to the other choices such that it could not be easily eliminated because of its appearance.*
B. Correct
C. Incorrect –Both the 'A' and 'C' CCW pumps are apart of the A CCW Train. This distractor has the wrong “a” train pump starting at 5 seconds. See explanation above.

D. Incorrect – Incorrect – the train specific CCW pumps have a staggered starting time to prevent overloading the EDG.

Technical Reference(s):
1. E-22NF01, Load Shedding and Emergency Load Sequencing Logic, Rev 8

References to be provided to applicants during examination: None

Learning Objective: T61.0110, systems, LP #10, CCW, Objective C:
DESCRIBE the purpose and operation of the following CCW System components:
1. CCW Pumps
2. CCW Heat Exchangers (H/Xs)
3. CCW Surge Tanks
4. CCW Chemical Addition Tank
5. CCW Radiation Monitors
6. Surge Tank Vent Valves
7. Essential Service Water (ESW) to CCW Valves
8. CCW HX Temperature Control Valves
9. Containment Isolation and Bypass Valves
10. Radwaste Isolation Valves
11. RCP Coolers and Thermal Barriers
12. Thermal Barrier Isolation valves and Flow Elements

Question Source: Bank # _X_L16179____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2005________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

k/a/ match as a Safety Injection is a different condition of the plant which generates an automatic operation of the CCW system.

Revised question per NRC comment on the plausibility of the A distractor. Reworded stem by removing "if any".

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
063 DC Electrical Distribution	Group #	1		
	K/A #	00063 A4.03		
	Importance Rating	3.0		
Ability to manually operate and / or monitor in the control room: Battery Discharge Rate				

Question # 35

A Loss of All AC Power has occurred. NK12 battery discharge amps are 200 amps.

Which of the following is the MAXIMUM time that NK12 could supply NK02 bus? (Assume NK12 was fully charged at the time of the loss of all AC power.)

A. 4.5 hours

B. 6 hours

C. 8.25 hours

D. 12 hours

Answer: A

Explanation:

Per OTO-NB-00002 step #25, the capacity of the NK and PK batteries are as follows:
NK12 = 900 amp hours
NK14 = 1650 amp hours
PJ battery = 1200 amp hours
PK12 = 2400 amp hours

Note: The PJ rating is on reference is E-21PJ01.

The distractors are plausible as the candidate must correctly remember the NK12 amp hour rating value and if they falsely remember any other battery rating (safety or non safety related), they will choose an incorrect answer

A. Correct – 900/200 = 4.5 hours
B. Incorrect – 1200/200 = 6 hours
C. Incorrect –1650/200 = 8.25 hours
D. Incorrect – 2400/200 = 12 hours

Technical Reference(s):

1. OTO-NB-00002, Loss of Power to NB02, Rev 29
2. E-21PJ01, 250 VDC meter and relay Diagram non-class 1E Power System, Rev 4

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #9 Safeguards Power, Objective M:
EXPLAIN the precautions, limitations and bases for the following components/conditions associated with OTN-NK-00001, “Class 1E 125 VDC Electrical System”:

1. Battery capacity

2. Maximum NK Battery Charge amperage output

Question Source: Bank # __X_L4521_
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2013_question #49_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(8)

Comments:

Question was replaced per NRC Comments. Revised the stem (removed earthquake wording) and one distractor from bank question

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
25.	**CHECK The Following Buses - SUPPLIED FROM ASSOCIATED BATTERY** • NK04, (1650 amp hours) • NK02, (900 amp hours) • PK02, (2400 amp hours local) • PK04, (2400 amp hours local)	TRANSFER deenergized bus to associated battery using the following as appropriate: • OTN-NK-00001, Class 1E 125 VDC Electrical System • OTN-PK-00001, Non-Class 1E 125 VDC Electrical System
26.	**PERFORM The Following To Maintain Battery Life:** a. CONTACT System Engineering To Determine Expected Battery Life b. MONITOR battery load c. CROSS-TIE Buses PK02 And PK04 To Functional Battery Chargers per OTN-PK-00001, Non-Class 1E 125 VDC Electrical System	
27.	**CHECK Swing Charger NK26 - AVAILABLE**	Go To Step 30.
28.	**TRANSFER Swing Charger NK26 To Alternate Power Supply From PG Bus Per OTN-NK-00001, Class 1E 125 VDC Electrical System**	
29.	**CHARGE Batteries NK12 Or NK14 Per OTN-NK-00001, Class 1E 125 VDC Electrical System**	
30.	**REVIEW Applicable Technical Specifications:** • Refer To Attachment G, Technical Specifications	
31.	**PERFORM Notifications Per ODP-ZZ-00001 Addendum 13, Shift Manager Communications**	

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Pressurizer Pressure Control	Group #	1		
	K/A #	010 A2.02		
	Importance Rating	3.9		
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: Spray valve failures				

Question # 36

Reactor Power is 40%.

- Pressurizer Spray Valve, BB PCV0455B, FAILS OPEN and can NOT be manually closed.
- Pressurizer Pressure begins to slowly lower.

(1) What is the LOWEST pressure that the Pressurizer Backup Heaters should automatically energize?

And

(2) If Pressurizer Pressure continues to lower after the Pressurizer Backup Heaters are energized, the crew should stop which RCPs per OTO-BB-00006, Pressurizer Pressure Control Malfunction?

A. (1) 2210 psig
(2) A and D

B. (1) 2210 psig
(2) B and D

C. (1) 2220 psig
(2) A and D

D. (1) 2220 psig
(2) B and D

Answer: A

Explanation:

OTO-BB-00006 will be performed and at step #20, when both PZR Sprays valves are checked *closed, the RNO action will be performed. In this RNO as pressure continues to lower after the backup heaters are on, a reactor and turbine trip are required along with securing the appropriate RCPs: For BB PCV455B it is the A and D RCP. The distractors of the B and D RCP are for BB PCV455C.*

The variable heater would be fully energized at 2220 psig and the backup heaters would be on @2210 psig

A. Correct – See above explanation
B. Incorrect – wrong RCPs
C. Incorrect – Wrong pressure. This is when variable heaters are fully energized
D. Incorrect – both are wrong

Technical Reference(s):

1. OTO-BB-00006, Pressurizer Pressure Control Malfunction, Rev 20
2. OTN-BB-00005, Attachment 1, Master Pressure Controller, Rev 14

References to be provided to applicants during examination: None

Learning Objective:
T61.003B, Off Normal Operations, LP #41, OTO-BB-00006, Objective C: Given a set of plant conditions or parameters indicating a Pressurizer Pressure Control Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

T61.0110, Systems, LP #9, RCS, Objective B: DESCRIBE the purpose and operation of the following RCS components to include interlocks, controller operations and power supply:

5. Power Operated Relief Valves (PORVs)

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

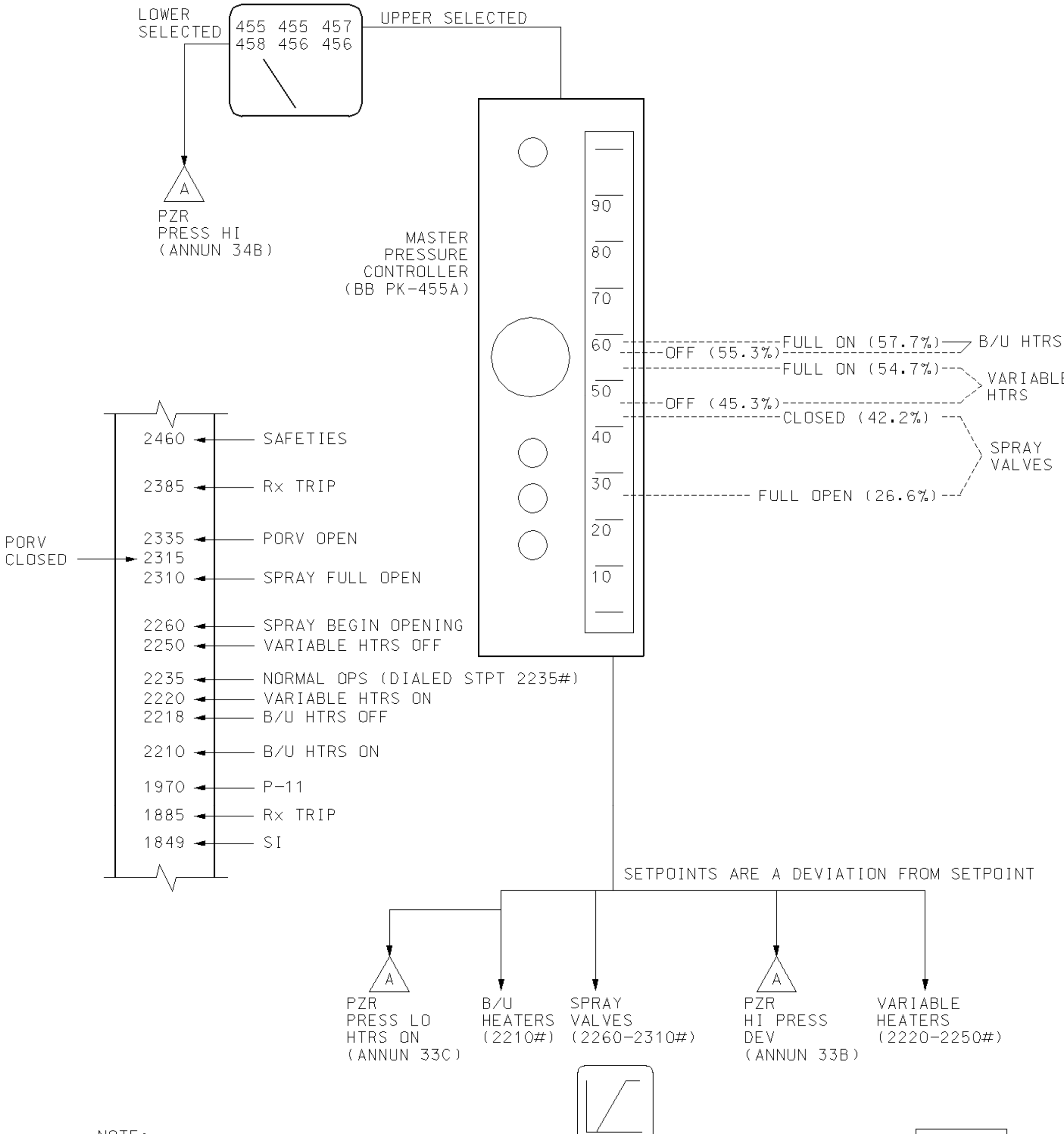
k/a match as the impact of a spray valve failures in PZR PCS will be that PZR Pressure will lower. There are no other plausible PZR Pressure responses. The impacts of this failure are the automatic system response (backup heater automatically turning on) and if this operation is not successful in stopping PZR Pressure from lowering, stopping the required RCPs is directed by the off normal procedure to mitigate the spray valve failure. The impacts and interrelationships of PZR PCS and RPS are tested in the previous question so to prevent overlap, RPS impacts are N/A for this question.

Revised wording in stem for part 2 per NRC comments.

Attachment 1

Master Pressure Controller

Sheet 1 of 1



NOTE:

PORV'S ARE OPERATED BY 2 OF 4 PRESSURIZER PRESSURE NARROW RANGE INSTRUMENTS OR COMS SETPOINT IF ARMED.

PROC-0239
07/26/05

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20.	**CHECK Both Pressurizer Spray Valves - CLOSED** • BB ZL-455B • BB ZL-455C	PERFORM the following: a. PLACE the affected Pressurizer Spray Loop Controller in MANUAL and CLOSE the valve: • BB PK-455B • BB PK-455C b. ENERGIZE Pressurizer Backup Heaters as necessary to stabilize Pressurizer pressure: • BB HIS-51A • BB HIS-52A c. IF Pressurizer pressure continues to lower in an uncontrolled manner, THEN PERFORM the following: 1) Manually TRIP the Reactor. 2) ENSURE Main Turbine is tripped. 3) IF BB PCV-455B can NOT be closed, THEN STOP RCP A and RCP D. 4) IF BB PCV-455C can NOT be closed, THEN STOP RCP B and RCP D. 5) PERFORM E-0, Reactor Trip Or Safety Injection. 6) IF PZR pressure continues to lower, THEN STOP all but one RCP.
21.	**CHECK Pressurizer Pressure - GREATER THAN 2250 PSIG**	Go To Step 23.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22.	**CHECK Pressurizer Heaters - DEENERGIZED** • BB HIS-50 • BB HIS-51A • BB HIS-52A	PERFORM the following: a. DEENERGIZE Pressurizer Heaters: • BB HIS-50 • BB HIS-51A • BB HIS-52A b. IF the Pressurizer Pressure Master Controller is the source of the problem, THEN PLACE the controller in MANUAL: • BB PK-455A c. IF Pressurizer pressure is greater than 2335 PSIG, THEN ENSURE at least one Pressurizer PORV is open: • BB HIS-455A • BB HIS-456A d. RESTORE Pressurizer pressure between 2220 psig and 2250 psig. e. IF Pressurizer pressure lowers to less than 2315 PSIG, THEN ENSURE the Pressurizer PORV close: • BB HIS-455A • BB HIS-456A f. IF Pressurizer pressure lowers to less than 2315 psig AND the PORV does not close, THEN CLOSE the associated Pressurizer PORV Block Valve: • BB HIS-8000A • BB HIS-8000B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23.	**CHECK Pressurizer Pressure - BETWEEN 2220 PSIG AND 2250 PSIG**	WHEN Pressurizer Pressure is between 2220 psig and 2250 psig, THEN CONTINUE with the next step.
24.	**CHECK Pressurizer Pressure Master Controller - CONTROLLING IN AUTO** • BB PK-455A	IF the Pressurizer Pressure Master Controller is NOT the source of the problem, THEN PERFORM the following: a. PLACE the controller in AUTO. b. ENSURE Pressurizer pressure is being controlled between 2220 psig and 2250 psig. IF the Pressurizer Pressure Master Controller is the source of the problem, THEN manually CONTROL Pressurizer pressure between 2220 psig and 2250 psig.
25.	**CHECK Pressurizer Heaters - ALIGNED FOR AUTOMATIC CONTROL** • BB HIS-50 • BB HIS-51A • BB HIS-52A	IF automatic Pressurizer heater control is NOT the source of the problem, THEN ALIGN Pressurizer heaters for automatic operation: • BB HIS-50 • BB HIS-51A • BB HIS-52A IF automatic Pressurizer heaters control is the source of the problem, THEN manually OPERATE Pressurizer heaters.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Reactor Protection	Group #	1		
	K/A #	012 G2.2.22		
	Importance Rating	4.0		
Knowledge of limiting conditions for operations and safety limits				

Question # 37

Reactor Power is 100%.

- AE LT-551, 'A' S/G NR Level Channel, fails to 0%.

(1) Per Technical Specifications, the S/G Water Level LO LO Reactor Trip Function is REQUIRED in what MODE(s)?

And

(2) What is the remaining logic for generating a S/G Water Level LO LO Reactor Trip?

A. (1) MODE 1 ONLY
(2) 1 / 2

B. (1) MODE 1 ONLY
(2) 1 / 3

C. (1) MODEs 1 and 2
(2) 1 / 2

D. (1) MODEs 1 and 2
(2) 1 / 3

Answer: D

Explanation:

There are 4 S/G level detectors that are input into the logic. The logic for a SG LO LO RX Trip is 2/4. With one of these failing low, the result will be three remaining channels with a low signal in any 1 of these 3 generates a RPS protective action. The distractor of 2 remaining is from the PZR high level trip logic in which there are 3 total level channels and with one removed due to the failure one out of the remain 2 would generate a protective action.

Per Technical Specification Tables 3.3.1-1, the S/G Water level LO LO Function #14 is required in MODEs 1 and 2. Mode 1 ONLY is plausible as several RPS Function are ONLY required in MODE 1 such as PZR Water Level, RCP UV and low flows, and PZR Pressure LOW.

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Correct – See above explanation

Technical Reference(s):

1. Technical Specifications 3.3.1, RTS Instrumentation
2. E-0, Reactor Trip or Safety Injection, Rev 16

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #27, Reactor Protection, Objective C and J

Objective C: LIST all the Reactor Trip Signals supplied to RPS, including setpoint, coincidence, interlocks and protection afforded.

Objective J: STATE the Limiting Conditions for Operation (LCO) and Bases associated with the RPS related Technical Specifications.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Replaced k/a and revised question per NRC Comment

TABLE 3.3.1-1 (PAGE 1 OF 8)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
1.	Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
		3(b), 4(b), 5(b)	2	C	SR 3.3.1.14	NA
2.	Power Range Neutron Flux					
	a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 112.3% RTP
	b. Low	1(c),2(f)	4	V	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
		2(h),3(i)	4	Y, Z	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
3.	Power Range Neutron Flux Rate - High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 6.3 % RTP with time constant ≥ 2 sec
4.	Intermediate Range Neutron Flux	1(c), 2(d)	2	F, G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 35.3% RTP

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(f) With $k_{eff} \geq 1.0$.
(h) With $k_{eff} < 1.0$, and all RCS cold leg temperatures ≥ 500°F, and RCS boron concentration ≤ the ARO critical boron concentration, and Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(i) With all RCS cold leg temperatures ≥ 500°F, and RCS boron concentration ≤ the ARO critical boron concentration, and Rod Control System capable of rod withdrawal or one or more rods not fully inserted

TABLE 3.3.1-1 (PAGE 2 OF 8)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
5.	Source Range Neutron Flux	2[e]	2	I, J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.6 E5 cps
		3[b], 4[b], 5[b]	2	J, K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.6 E5 cps
6.	Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (at the end of this Table)
7.	Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (at the end of this Table)
8.	Pressurizer Pressure					
	a. Low	1[g]	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1874 psig
	b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2393 psig

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

TABLE 3.3.1-1 (PAGE 3 OF 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.8% of instrument span
10. Reactor Coolant Flow - Low	1(g)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 88.8% of indicated loop flow
11. Not Used					
12. Undervoltage RCPs	1(g)	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 10105 Vac
13. Underfrequency RCPs	1(g)	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 57.1 Hz
14. Steam Generator (SG) Water Level Low-Low(l)					
a. Steam Generator Water Level Low-Low (Adverse Containment Environment)	1, 2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 20.6% (q) of Narrow Range Instrument Span

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

(l) The applicable MODES for these channels in Table 3.3.2-1 are more restrictive.

(m) Not used.

(q)
1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases

TABLE 3.3.1-1 (PAGE 4 OF 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
14. Steam Generator (SG) Water Level Low-Low[l]					
b. Steam Generator Water Level Low-Low (Normal Containment Environment)	1[p],2[p]	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 16.6% [q] of Narrow Range Instrument Span
c. Not used.					
d. Containment Pressure - Environmental Allowance Modifier	1,2	4	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2.0 psig
15. Not Used					

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

(l) The applicable MODES for these channels in Table 3.3.2-1 are more restrictive.

(n) Not used.

(o) Not used.

(p) Except when the Containment Pressure - Environmental Allowance Modifier channels in the same protection sets are tripped.

(q) 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases

TABLE 3.3.1-1 (PAGE 5 OF 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[(a)]
16. Turbine Trip					
a. Low Fluid Oil Pressure	1[(j)]	3	O	SR 3.3.1.10 SR 3.3.1.15	≥ 539.42 psig
b. Turbine Stop Valve Closure	1[(j)]	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2[(e)]	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 51.3% RTP
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 53.3% RTP

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(j) Above the P-9 (Power Range Neutron Flux) interlock.

TABLE 3.3.1-1 (PAGE 6 OF 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[(a)]
18. Reactor Trip System Interlocks					
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6.7% RTP and ≤ 12.4% RTP
f. Turbine Impulse Pressure, P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	≤ 12.4% turbine power
19. Reactor Trip Breakers (RTBs)[(k)]	1,2	2 trains	R	SR 3.3.1.4	NA
	3[(b)], 4[(b)], 5[(b)]	2 trains	C	SR 3.3.1.4	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms[(k)]	1,2	1 each per RTB	U	SR 3.3.1.4	NA
	3[(b)], 4[(b)], 5[(b)]	1 each per RTB	C	SR 3.3.1.4	NA
21. Automatic Trip Logic	1,2	2 trains	Q	SR 3.3.1.5	NA
	3[(b)], 4[(b)], 5[(b)]	2 trains	C	SR 3.3.1.5	NA

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

TABLE 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.23% of ΔT span (1.85% RTP).

$$\Delta T \frac{(1 + \tau_1 s)}{(1 + \tau_2 s)}\left(\frac{1}{1 + \tau_3 s}\right) \leq \Delta T_o \left\{K_1 - K_2\left[\frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} T \frac{1}{1 + \tau_6 s} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_o is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ *°F.

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure = * psig.

K_1 = *	K_2 = */°F	K_3 = */psig
$\tau_1 \geq$ * sec	$\tau_2 \leq$ * sec	τ_3 = * sec
$\tau_4 \geq$ * sec	$\tau_5 \leq$ * sec	τ_6 = * sec

$f_1(\Delta I)$ =	* {*% + (q_t - q_b)}	when q_t - q_b < * %RTP
	0% of RTP	when * %RTP ≤ q_t - q_b ≤ * %RTP
	* {(q_t - q_b) - *}	when q_t - q_b > * %RTP

where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

The values denoted with * are specified in the COLR.

TABLE 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.21% of ΔT span (1.82% RTP).

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \leq \Delta T_o \left\{ K_4 - K_5 \frac{(\tau_7 s)}{(1+\tau_7 s)}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{(1+\tau_6 s)} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_o is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature,°F.
T″ is the nominal T_{avg} at RTP, ≤ *°F.

K_4 = *	K_5 = */°F for increasing T_{avg} */°F for decreasing T_{avg}	K_6 = */°F when T > T″ */°F when T ≤ T″
τ_1 ≥ * sec	τ_2 ≤ * sec	τ_3 = * sec
τ_6 = * sec	τ_7 ≥ * sec	

$f_2(\Delta I)$ = *

The values denoted with * are specified in the COLR.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Engineered Safety Features Actuation	Group #	1		
	K/A #	013 K2.01		
	Importance Rating	3.6		
Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control				

Question # 38

What is the 120 VAC power supply to SA066B, ESFAS Logic Cabinet?

A. NN01

B. NN02

C. NN03

D. NN04

Answer: D

Explanation:
Per OOA-SA-C066X step 4.1 and E23-SA22, the power supply to SA066B is NN0421. All other 120VAC NN buses are plausible as the all safety related NN buses.

A. Incorrect
B. Incorrect
C. Incorrect
D. Correct

Technical Reference(s):
1. OTS-SA-00001, Operations of EFSAS, Rev 19
2. OTA-RK-00018, Addendum 47F, EFSAS Not Normal, Rev 1
3. E-23SA22, Schematic Diagram, ESFAS Cabinets, Rev 1
4. OOA-SA-C066X, EFSAE Panel SA066X Alarm Information, Rev 15

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #52, EFSAS, Objective E & C & F

F: DISCUSS the purpose and scope of the following: OTS-SA-00001, "De-energizing and Energizing Engineered Safety Feature Actuation System”.

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E: IDENTIFY the ESFAS status panel controls, alarms and indications and DESCRIBE how each is used to predict, monitor, test or control the ESFAS.

C; DISCUSS the following concerning the ESFAS power up/down sequence:

1. The purpose of blocking crosstrips from the de-energized channel prior to de-energization.
2. De-energizing the 48 VDC output relay power before de-energizing the dual voltage electronics power supply on a down power.
3. Ensuring that all actuations are reset prior to energizing the 48 VDC output relay power on an up power.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

4.0 SA066X GENERAL INFORMATION

4.1. The power supplies to the ESF logic/termination racks are:

- SA066A - NN0104
- SA066B - NN0421
- SA066C - NN0219

Each rack has a power supply which feeds the Visual Display Panel lamps.

4.2. This OOA is providing detailed information for alarm conditions (RED with actuation, or RED without actuation); however, the three modes of operation (DARK, WHITE, RED) are briefly described:

DARK (unlit)

- When a device is NOT in its safeguards condition AND an actuation signal is present, the device window and the system level window are DARK.
- When no actuation signal is present and the panel has been reset (using SA HS-23 on RL018), the window is DARK
- All windows normally are DARK

WHITE

- A window turns WHITE if an actuation signal is present AND the device is in it's correct safeguards position. For the system level windows, ALL devices have to be correctly positioned in order for the white light to be lit.

NOTE

The system light does NOT illuminate red when power is removed from devices if the device is in the safeguards position when the power is removed. The system light will only illuminate red if the device is unable to go to its safeguards position or, in most cases, if the component level windows are illuminated red.

RED

- A window turns RED (alarm condition) and a HORN sounds when a device is potentially prevented from performing its safeguards function. This condition inputs to the system level window and cause that window to also alarm.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Engineered Safety Features Actuation	Group #	1		
	K/A #	013 K6.01		
	Importance Rating	2.7		
Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors				

Question # 39

Reactor Power is 100%.

Pressurizer Pressure Protection Channel PT-455 fails and is properly removed from service per the off normal procedure.

What is the ESF actuation logic required, from the remaining in-service channels, to initiate a safety injection on low pressurizer pressure?

A. 1 / 2

B. 1 / 3

C. 2 / 2

D. 2 / 3

Answer: B

Explanation:

Per E-0, Section B the normal logic to actuate a safety injection due to a low PZR Pressure is 2/4 above P-11 @1849 psig. All other SI signals (i.e Steam Line Pressure and Containment Pressure) are normally 2/3 logic.

The failed PZR pressure channel will be removed per OTO-BB-00006 Attachment B. It will be tripped and not bypassed and therefore one of the required logic channels for an SI will be present. It would only take 1 of the 3 remaining in service channels to generate an SI.

The distinction between tripping and bypassing is important for the distractors. Furthermore, the required logic with nothing out of service (i.e. 2 out of 4 or 2 out of 3) will been tested by the distractors.

A. Incorrect – Plausible if the candidate assumes the initial logic is 2/3 not the correct 2/4 and applies the fact that one channel is tripped. I.e. 2 / 3 goes to 1 / 2 for this plant condition
B. Correct. 2/4 normal logic with one tripped results in a 1 / 3 logic for this plant condition

C. Incorrect – Plausible if the candidate assumes the initial logic is 2/3 not the correct 2/4 and applies bypassing the channel not that one channel is tripped. i.e the 2 / 3 logic would be 2 / 2.
D. Incorrect - Plausible if the candidate knows initial logic is 2/4 but incorrectly applies bypassing the channel not that one channel is tripped. i.e 2 / 4 logic would now be 2 / 3.

Technical Reference(s):
1. E-0, Reactor Trip or Safety Injection, Rev 16
2. OTO-BB-00006, Attachment B, Rev 20

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #17, Safety Injection, Objective D and F:
D. STATE the conditions that will initiate a Safety Injection Signal and DESCRIBE the conditions necessary to reset the signal.

F. LIST the systems that interface with the SI System and EXPLAIN how a loss of the interfacing system or a loss of the SI System affects the other.

Question Source: Bank # ______ X L16229____
Modified Bank # __________
New _______

Question History: Last NRC Exam ____2007________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:
k/a tie as it presents a loss of a required EFSAS sensor or detector (i.e. pressurize pressure channel) and the effect that it has on the EFSAS logic required to generate a Safety Injection signal (an EFSAS signal) when it is correctly removed from service. As there are only four signals that generate this EFSAS signal (e.g Safety Injection), and the logic is function specific (i.e. how many PZR Pressure low channels generate a SI), PZR Pressure channel failure had to be specified.

B. SYMPTOMS OR ENTRY CONDITIONS

1) The following are symptoms that require a reactor trip, if one has not occurred:

Reactor Trip	Logic, Interlock	Setpoint
SR high flux	(1/2, P-10 and P-6)	10^5 CPS
IR high flux	(1/2, P-10)	25% of RTP
PR high flux low level	(2/4, P-10)	25% of RTP
PR high flux high level	(2/4)	109%
PR positive rate trip	(2/4) (two seconds)	+4.25%
Overtemperature ΔT	(2/4)	122.6% ±
Overpower ΔT	(2/4)	110.73% -
PZR pressure - Low	(2/4, P-7)	1885 PSIG
PZR pressure - High	(2/4)	2385 PSIG
PZR water level - High	(2/3, P-7)	92%
Rx coolant flow - Low	(2/3, 2/4 P-7, 1/4 P-8)	90% Design
RCP bus undervoltage	(1/2, 2/2 P-7)	10584 VAC
RCP bus underfrequency	(1/2, 2/2 P-7)	57.2 Hz
SG NR level - Low-Low	(2/4, 1/4) Normal {EAM}	17% {21%}
Turb trip - Low oil press	(2/3, P-9)	598.94 PSIG
Turb trip - Stop valves	(4/4, P-9)	1% OPEN
Safety Injection ESFAS	(1/4 signals)	SI
SSPS General warning	(2/2)	N/A

2) The following are symptoms of a reactor trip:

- Any reactor trip annunciator lit.
- Rapid lowering of neutron flux on nuclear instrumentation.
- All shutdown and control rods are fully inserted.
- Rod bottom lights are lit.

3) The following are symptoms that require a reactor trip and safety injection, if one has not occurred:

Reactor Trip & Safety Injection	Logic, Interlock	Setpoint
PZR pressure - Low	(2/4, P-11)	1849 PSIG
Steamline pressure - Low	(2/3 on 1/4, P-11)	615 PSIG
Containment pressure - High-1	(2/3)	3.5 PSIG

4) The following are symptoms of a reactor trip and SI:

- Any SI annunciator lit.
- ECCS pumps running.

5) This procedure should also be entered any time a manual reactor trip or safety injection is actuated.

C. CONDITIONS FOR [ADVERSE CONTAINMENT]

- Containment Radiation — HAS BEEN GREATER THAN 10^5 R/HR

  OR
- Containment Pressure — GREATER THAN 3.5 PSIG

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	2		
Containment Cooling	**Group #**	1		
	K/A #	00022 K4.04		
	Importance Rating	2.8		
Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Cooling of control rod drive motors				

Question # 40

(1) CRDM Cooling Fan shroud temperature is maintained less than a MAXIMUM of ___(1)_____ to prevent exceeding the CRDM coil design temperature.

And

(2) If a Safety Injection were to occur, which CRDMs should be load shed? (Assume there is NO loss of offsite power.)

A. (1) 120°F
(2) A & C

B. (1) 120°F
(2) B & D

C. (1) 165°F
(2) A & C

D. (1) 165°F
(2) B & D

Answer: D

Explanation:
OTN-GN-00001, step 3.2 states that "Whenever reactor coolant temperature exceeds 200°F or whenever the CRDMs are energized, sufficient CRDM Cooling Fans shall be in service to maintain less than 165°F on computer points GNT0045 and GNT0046."

The distractor of 120F is the limit for the PZR skirt in step 3.4 of OTN-GN-00001. This is also the normal temperature containment air limit.

The power supplies for the CRDM fans are as follows (see E-23 drawing for supplies):
CGN01A – PG20G
CGN01B – NG02B
CGN01C – PG19G
CGN01D – NG01B

On a SI signal, a non safety load shed occurs and NG01B and NG02B are shed, therefore B&D would be shed. A &C would still be powered from PG19 and 20 since these are still powered (no loss of offsite power).

From the UFSAR section 9.4.6.1.2 The CRDM cooling system is designed to maintain all CRDM coils below the design temperature limit of 392°F and is in operation any time the reactor coolant temperature is greater than 200°F. The cooling of the air is provided by the containment cooler. During two- and three-fan operation, this is achieved with a containment air temperature of 120°F and a CRDM cooling fan inlet temperature not greater than 165°F.

A. Incorrect – both are wrong
B. Incorrect – wrong temperature
C. Incorrect – wrong CRDM fans shed
D. Correct

Technical Reference(s):
1. OTN-GN-00001, Containment Cooling and CRDM Cooling, Rev 28
2. FSAR, Section 9.4.6.1.2, Power Generation Design Bases, page 9.4-55
3. E-23GN03A, CRDM Cooling Fans and Discharge Dampers FED from 1E Bus, Rev 2
4. E-23GN03, CRDM Cooling Fans and Discharge Dampers, Rev 11

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #40, Containment Ventilation, Objective A &B:

A:STATE the function and EXPLAIN the design criteria of the containment cooling system.

B: DESCRIBE the purpose and operation of the following containment cooling system components.

4. CRDM Cooling Fans

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Revised both stems per NRC comment.

CONTAINMENT COOLING AND CRDM COOLING

1.0 PURPOSE

1.1. This procedure provides instructions for alignment and operation of the Containment Cooling System.

1.2. This procedure provides instructions for operation of the Containment Cooling System in the event of a system malfunction.

2.0 SCOPE

This procedure is applicable to the operation of the Containment Coolers, Hydrogen Mixing Fans, Control Rod Drive Mechanism Cooling Fans, Cavity Cooling Fans, Pressurizer Cooling Fan, and Elevator Machine Room Exhaust Fan.

3.0 PRECAUTIONS AND LIMITATIONS

3.1. At a minimum, either Containment Cooler Fan A or C AND one Containment Cavity Cooling Fan should be in operation whenever reactor coolant temperature exceeds 175°F. Action to reduce cavity temperatures should be taken if cavity concrete temperatures exceed 140°F.

3.2. Whenever reactor coolant temperature exceeds 200°F or whenever the CRDMs are energized, sufficient CRDM Cooling Fans shall be in service to maintain less than 165°F on computer points GNT0045 and GNT0046.

3.3. No more than THREE (3) CRDM Cooling Fans can be run at the any given time.

3.4. Either Containment Cooler D or the Pressurizer Cooling Fan must be in service to limit the area below the pressurizer skirt to 120°F.

3.5. The Containment Coolers and Hydrogen Mixing Fans will shift to slow speed on a safety injection signal.

3.6. Starts or attempted starts should be a minimum of 15 minutes apart to allow sufficient time for the thermal overloads to cool.

3.7. High pressure (greater than 25 in WG) and cool temperatures (outlet temp less than 65°F) across the containment coolers will cause the coolers to operate close to the setpoint of the thermal overloads. Therefore, the containment coolers should be operated in slow speed during periods of cool service water temperatures (SW/ESW water less than 60°F).

3.8. If SW/ESW flow is stopped, the affected Containment Cooler(s) shall be isolated by closing both the inlet and outlet isolation valves prior to restoring flow through the ESW system. This is necessary to prevent a "water hammer" event due to partial draining of the ESW supply/return lines.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Containment Spray	**Group #**	1		
	K/A #	026 K1.01		
	Importance Rating	4.2		
Knowledge of the physical connections and/or cause/effect relationships between the CSS and the following systems: ECCS				

Question # 41

What pumps can be lined up to take a DIRECT suction from the containment recirculation sump?

A. RHR and SI

B. RHR and Containment Spray

C. CCPs and Containment Spray

D. CCPs and SI

Answer: B

Explanation:

Both the RHR and Containment Spray pumps can be lined up to take a suction directly from the containment recirculation sump. See M22 references. Other ECCS pumps can be lined up to take an INDIRECT suction from the containment sumps during the performance of several Emergency procedures. See explanation below.

A. Incorrect - The RHR pumps can be lined up to directly take a suction from the containment recirculation sump. The SI pumps cannot be directly lined up to take a suction from the containment recirculation sump. Plausible because when the RWST is low and the SI pumps are used the SI pumps take a suction on the RHR header that is lined up to the containment recirculation sump.
B. Correct
C. Incorrect - The CCPs cannot be directly lined up to take a suction from the containment recirculation sump. Containment Spray pumps can be lined up to take a suction directly from the containment recirculation sump. Plausible because when the RWST is low the CCPs can be lined up to take a suction on the RHR header that is lined up to the containment recirculation sump. Additionally, the CCP suctions are realigned during the performance of emergency procedures and the candidate may falsely believe / remember that during the performance of these the CCP is aligned to the containment sump. Specifically, in ECA 1.1 Sump Blockage Mitigation, there are several actions and steps to verify CCPs show no signs cavitation and if there is cavitation the operator is directed to secure the pump. The operator could falsely believe that the CCPs can be directly aligned to the sump by the actions provided in this procedure. (i.e. the pump is cavitation

when performing actions to mitigate sump blockage, I must secure the pump as its suction source is blocked).
D. Incorrect – see above explanations for CCP and SI pumps.

Technical Reference(s):
1. M-22EJ01, P&ID RHR System, Rev 62
2. M-22EN01, P&ID Containment Spray System, Rev 16

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #18, Containment Spray, Objective B: DESCRIBE the purpose, operation and location of the following Containment Spray System components:
1. Containment Spray Pumps
2. Containment Recirculation Sump

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

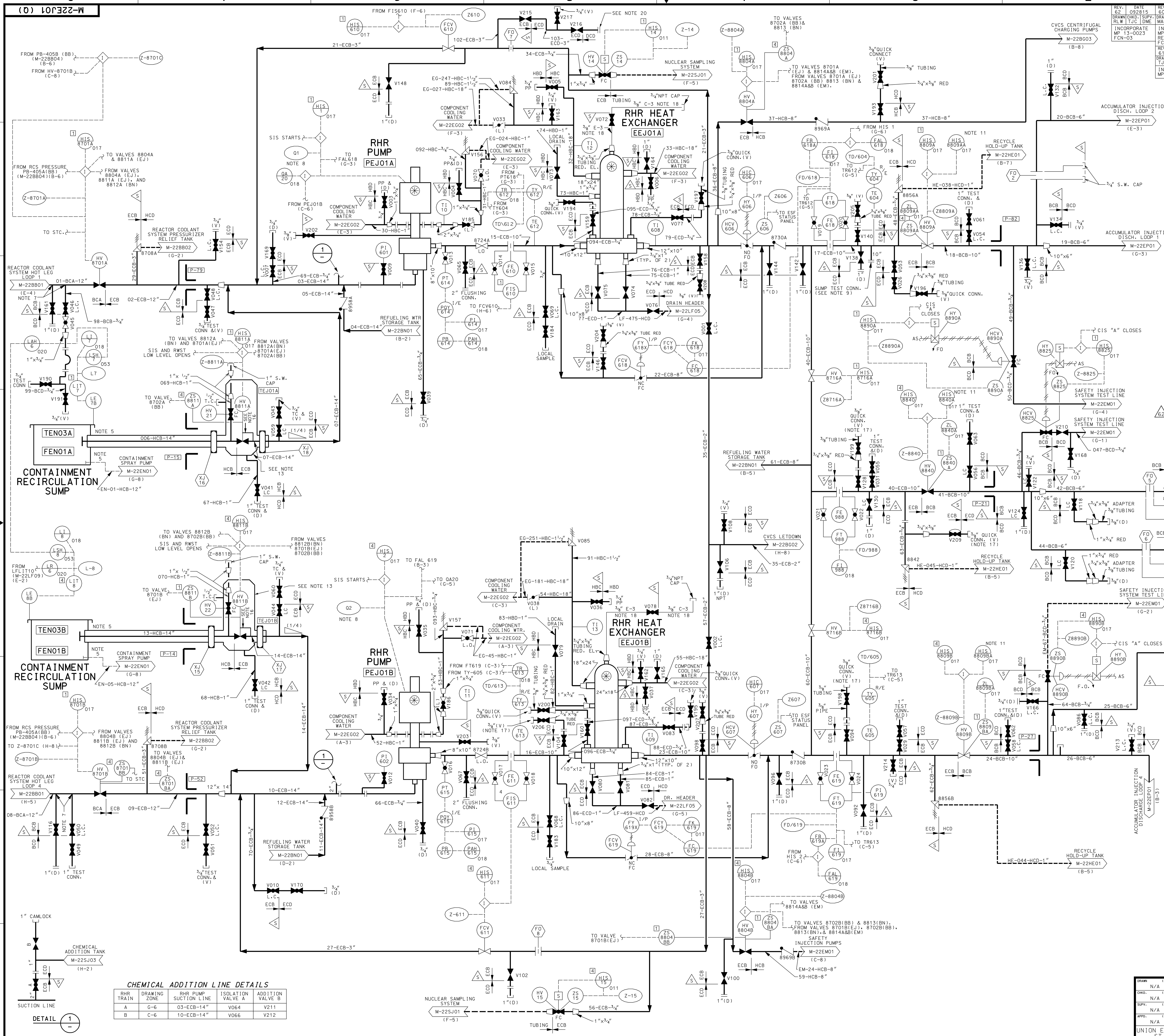
Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Revised explanation for the CCP plausibility per NRC comment.



REV.	DATE	DRAWN	CHKD.	SUPV.	DESCRIPTION
62	092815	RLW	TJC	DME	INCORPORATE MP 13-0023 FCN-03
61	110414	TJC	MAL	LMP	INCORP. MP 14-0024.
60	102914	MAL	RLW	LMP	INCORP. MP 13-0023 REV. 0 & FCN-01.
59	051010	RLW	EWM	LMP	INCORPORATE MP 08-0016 FCN-04
58	042910	RLW	EWM	LMP	INCORPORATE MP 08-0016 FCN-03
57	011708	RLW	EWM	TWS	INCORPORATE CR CAR 07-000257; CAR 200711457
56	041807	EWM	RLW	TWS	INCORPORATE MP 06-0029
55	103105	RLW	EWM	MWH	INCORPORATE MP 03-1014A
54	051005	RLW	MAL	TWS	INCORPORATE MP 01-1018 FCN-02
53	050605	RLW	MAL	SKC	INCORP. MP 01-1018 FCN-02 (PARTIAL PER REDLINE).
52	110904	MAL	RLW	TWS	INCORP. RFR-22941A

NOTES:

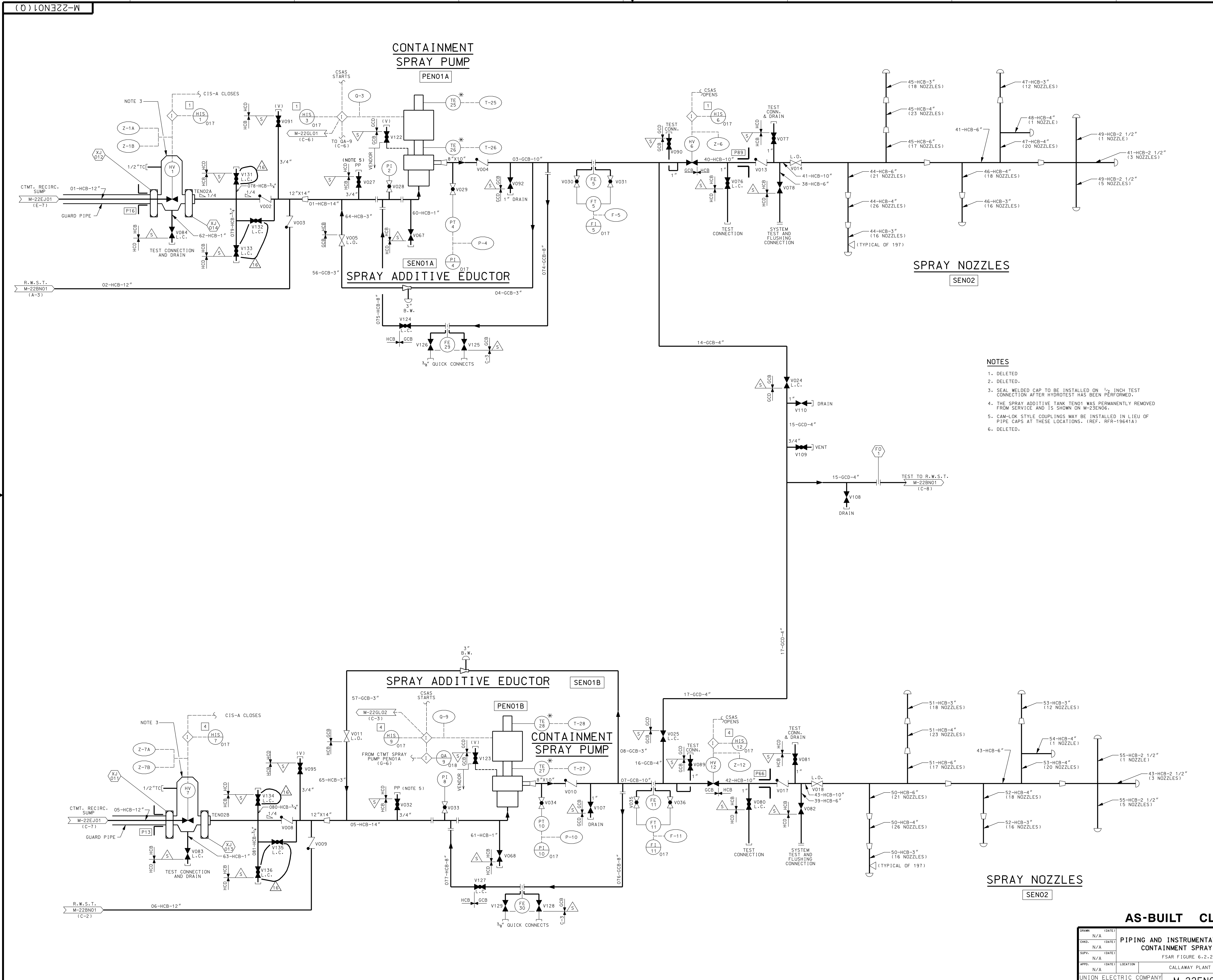
1. DELETED
2. DELETED
3. THIS DRAWING IS BASED ON WESTINGHOUSE DRAWING 1145E03
4. DELETED
5. VORTEX BREAKER LOCATED IN PIPE INLET.
6. DELETED.
7. PROVIDED 3/8"ID FLOW RESTRICTOR FOR INTERFACE BOUNDARY BETWEEN QUALITY GROUP CLASSIFICATIONS "A" AND "B".
8. WESTINGHOUSE NSSS COMPUTER INPUT.
9. TEE-TYPE 10" FITTING FOR CTMT. RECIRC. SUMP TEST. TO BE CLOSED OFF AFTER SUMP TEST. FITTING TO BE LOCATED IN PIPING PENETRATION ROOM.
10. COMPLETE AUTOMATIC SWITCHOVER TO BE DETERMINED LATER FOLLOWING RECEIPT OF WESTINGHOUSE INFORMATION.
11. ISOLATION SWITCH PROVIDED IN THE CONTROL ROOM FOR POWER LOCKOUT OF VALVES HV-8840 AND HV-8809A AND B.
12. DELETED.
13. WELD CAP TO BE INSTALLED ON 1/2" TEST CONNECTION AFTER HYDROTEST HAS BEEN PERFORMED.
14. DELETED
15. DELETED.
16. SEE DRAWING 135128 FOR ANTI-PRESSURE LOCKING BYPASS.
17. TYGON HOSES ARE INSTALLED AT THE NOTED LOCATIONS.
18. FOR INSTRUMENT PIPING CLASS SPECIFICATION FOR SS TUBING SEE DWG. J-27G10.
19. POWER HAS BEEN REMOVED FROM EJHV0021 & EJHV0022 BY MP 01-1002.
20. 62 THIS CONNECTION IS FOR BEYOND-DESIGN-BASIS (FLEX) REACTOR COOLANT SYSTEM MAKE-UP.

CHEMICAL ADDITION LINE DETAILS

RHR TRAIN	DRAWING ZONE	RHR PUMP SUCTION LINE	ISOLATION VALVE A	ADDITION VALVE B
A	G-6	03-ECB-14"	V064	V211
B	C-6	10-ECB-14"	V066	V212

AS-BUILT CLASS 1

DRAWN N/A	PIPING AND INSTRUMENTATION DIAGRAM RESIDUAL HEAT REMOVAL SYSTEM
CHKD. N/A	FSAR FIGURE 5.4-7
SUPV. N/A	
APPD. N/A	LOCATION: CALLAWAY ENERGY CENTER
UNION ELECTRIC COMPANY ST. LOUIS, MO	M-22EJ01(Q) REV. 62



NOTES

1. DELETED.
2. DELETED.
3. SEAL WELDED CAP TO BE INSTALLED ON 1/2 INCH TEST CONNECTION AFTER HYDROTEST HAS BEEN PERFORMED.
4. THE SPRAY ADDITIVE TANK TEN01 WAS PERMANENTLY REMOVED FROM SERVICE AND IS SHOWN ON M-23EN06.
5. CAM-LOK STYLE COUPLINGS MAY BE INSTALLED IN LIEU OF PIPE CAPS AT THESE LOCATIONS. (REF. RFR-19641A)
6. DELETED.

REV.	DATE	DRAWN	CHKD.	SUPV.	APPD.	DESCRIPTION
2	9/11/91	RLW	DLBr	TJM	N/A	REDRAWN FOR CLARITY
3	040495	HLP	SKC	AMR	N/A	INCORP. CMP 92-1053
4	101195	JHK	EWM	AMR	N/A	INCORP. CMP 92-1053 FCN-05.
5	052098	RLW	DJB	JAM	N/A	INCORP. MP89-1035.
6	042699	DLBr	SKC	JHC	N/A	INCORP. RFR-19641A
7	010604	MAL	EWM	TWS		INCORP. CARS 200307568 & RFR-22936A (PARTIAL).
8	012704	JHK	RLW	TWS		INCORP. RFR-22936A.
9	110904	MAL	RLW	TWS		INCORP. RFR-22941A
10	022008	MAL	TJC	LMR		INCORP. CHANGE REQUEST MP 05-1001 P01.
11	022708	MAL	TJC	TWS		INCORP. CHANGE REQUEST MP 05-1001 P02.
12	031808	RLW	TJC	SKC		INCORPORATE MP 05-1001 P03
13	032508	EWM	RLW	LMR		INCORPORATE MP 05-1001 FCN01 FINAL
14	102508	MAL	TJC	LMR		INCORP. CHANGE REQUEST MP 08-0016 P01
15	102808	EWM	RLW	LMR		INCORPORATE MP 08-0016 FCN01 FINAL
16	010611	RLW	EWM	TWS		INCORPORATE CR CAR 10-000309; CAR 201009292.

AS-BUILT CLASS 1

DRAWN N/A
CHKD. N/A
SUPV. N/A
APPD. N/A

PIPING AND INSTRUMENTATION DIAGRAM
CONTAINMENT SPRAY SYSTEM
FSAR FIGURE 6.2.2-1

LOCATION: CALLAWAY PLANT

UNION ELECTRIC COMPANY
ST. LOUIS, MO

M-22EN01(Q) REV. 16

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Containment Spray	**Group #**	1		
	K/A #	026 K2.01		
	Importance Rating	3.4		
Knowledge of the bus power supplies to the following: Containment spray pumps				

Question # 42

The "B" Containment Spray Pump, PEN01B, is directly powered from ….?

A. NB01

B. NB02

C. NG03

D. NG04

Answer: B

Explanation:

Per E-23nb04, the power supply to the B Containment Spray Pump is NB0203. NG04 is plausible as it is a "B" train MCC but it is 480VAC not 4160 VAC. NB01 and NG03 are "A" Train power 4160 and 480 VAC supplies respectively. NB01 powers the A Train containment spray pump and NG03 is an "A" Train MCC.

A. Incorrect – See above explanation
B. Correct – See above explanation
C. Incorrect – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. E-23NB04, Lower Medium Voltage System Class 1E 4.16 kv, Rev 6

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #18, Containment Spray, Objective C: EXPLAIN the interlocks, controls and power supplies to:
1. Containment Spray Pumps

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Revised question away from a 2 part question to simply a power supply based on NRC comments.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Main and Reheat Steam	**Group #**	1		
	K/A #	039 K5.08		
	Importance Rating	3.6		
Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity				

Question # 43

The plant was shutdown 10 days ago after operating for 9 months.

- A reactor startup is in progress in accordance with OTG-ZZ-00002, Reactor Startup – IPTE.
- The reactor is at normal operating temperature and pressure.
- The reactor is near criticality.
- Reactor startup rate (SUR) is stable at zero.

A Condenser Steam Dump valve fails open and remains stuck open.

The operator immediately ensures **NO** control rod motion is occurring and takes **NO** further action.

As a result of the valve failure, SUR will initially become ____________; and reactor power will stabilize __________ the point of adding heat.

A. negative, above

B. negative, at

C. positive, above

D. positive, at

Answer: C

Explanation: *The steam dump failure places a continuous heat load on the Reactor Coolant System, lowering its temperature. Positive reactivity is inserted due to the moderator's negative temperature coefficient at this time in core life. No xenon is present due to the time after shutdown. This causes a POSITVE SUR. Power increases to above the point of adding heat until an energy rate balance is achieved between primary and secondary systems. Reactor power will trend toward leveling off at this balanced condition. A static reactivity balance between the fuel reactivity effect (negative) and the coolant reactivity effect (positive) will occur, rendering the reactor critical*

A. Incorrect, see above. Plausible if the candidate assumes a positive moderator coefficient at this time in core life. Callaway can have a positive moderator coefficient at the beginning of core life.
B. Incorrect, see above. Plausible if the candidate assumes a positive moderator coefficient at this time in core life. Callaway can have a positive moderator coefficient at the beginning of core life.
C. Correct see above
D. Incorrect, see above. Plausible if the candidate misunderstands the feedback of temperature and its effects on reactor power near the POAH.

Technical Reference(s):
1. OTG-ZZ-00002, Reactor Startup - IPTE., Rev 56

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, A5, Reactor Startup, Obj D, APPLY the requirements of the Precautions and Limitations of OTG-ZZ-00002, to include: 2. Positive Reactivity additions

Question Source: Bank # ____X_P3567____
Modified Bank # _____
New _______

This question is modified to be plant specific from a question in the NRC Generic Fundamentals Examination Question Bank—PWR. These candidates took the September 2014 GF Exam and this question was not on that exam.

Question History: Last NRC Exam ___NA_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(1)

Comments:

This question covers the concepts that licensed operator failed to understand during the recriticality event that occurred at Oyster Creek on July 8, 2014

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Main Feedwater	Group #	1		
	K/A #	059 A1.07		
	Importance Rating	2.5		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Feed Pump speed, including normal control speed for ICS				

Question # 44

Reactor Power is 28%.

- A shutdown is in progress.
- ‘A’ MFP is in service.
- The transfer from MFRVs to MFRV Bypass Valves is being performed using the Automatic Method.

What is the predicted response of the ‘A’ MFP?

A. Speed will decrease to maintain the current psid across the MFRV Bypass Valve.

B. Speed will decrease to maintain 45 psid across the MFRV Bypass Valve.

C. Speed will increase to maintain 149 psid across the MFRV Bypass Valve.

D. Speed will increase to maintain 215 psid across the MFRV Bypass Valve.

Answer: D

Explanation: *Per OTN-AE-00001 step 5.8.20 – 22 when the automatic method of transfer from the MFRV the MFRV Bypass valves is used the Feed Pump Master Control DELTA P setpoint automatically raises to 215 psid.*

A. Incorrect – Plausible if the student believes that the system maintains the current psid value when the transfer occurs. Physically possible because with the transfer to the MFRV bypass valves occurs, flow will be through a smaller valve creating more head loss and to maintain the same differential pressure; centrifugal pumps speed will lower thereby lowering flow and raising discharge head. This would maintain a constant psid from pump to S/G. Plausible if the candidate believes that the same psid is maintained when using the MFRVs.
B. Incorrect - This is the minimum value for the Feed Pump Master Control DELTA P setpoint. This is a plausible distractor if the student incorrectly assumes the system goes to minimum value when transfer to the MFRV bypass occurs
C. Incorrect - This is the maximum value for the Feed Pump Master Control DELTA P setpoint when the MFRV is in use. This is a plausible distractor if the student incorrectly assumes the

system goes to maximum value for the MFRV when transfer to the MFRV bypass occurs
D. Correct, see above

Technical Reference(s):

1. OTN-AE-00001, Feedwater System, Rev 54

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #23, MAIN FEEDWATER SYSTEM – AE, Obj E, DESCRIBE the operation, including signal inputs, of the MFW pump speed control system and EXPLAIN the control response to input failures.

Question Source: Bank # ______
Modified Bank # _X___L17566____
New _______

Question History: Last NRC Exam ____Modified from 2014 ILT exam________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(4)

Comments:

Revised question per NRC comment.

NOTE

It may be necessary to RAISE the Low Power Automatic Delta P setpoint once the LOW POWER control transfer is complete based on Power Level and MFRV Bypass Valves positions.

On the DFWCS when FC PS 509, MFP ΔP CTRL SEL, is in LO PWR and FC SK 509A, MFP TURBS MASTER SPEED CTRL STA, is in AUTO, Delta P SET PT on FC SK 509A is controlled by the raise/lower arrows.

5.8.21. ACTUATE "ALL Steam Generators" XFR button on AE SS-500A, SG Level Control Valve Selection.

5.8.22. VERIFY the following sequence occurs:

a. FC PS-509 MFP DELTA P Control Selector, HIGH Power and LOW Power both FLASHING.

b. FC SK-509A, Feed Pump Master Control DELTA P Setpoint RAISES to 215 psid.

c. MFP in OPERATION Speed Increasing.

d. FC SK-509A, Feed Pump Master Control DELTA P RISING.

e. FC PS-509 MFP DELTA P Control Selector LOW POWER light ON, and HIGH POWER light OFF, When DELTA P exceeds approximately 210 psid.

f. AE SS-500A, SG Level Control Valve Selection, each SG XFR button GREYS out.

g. ALL SG MFRV and MFRV Bypass Valves controllers AUTO/MAN indicators begin FLASHING.

h. ALL MFRVs begin CLOSING.

i. ALL MFRV Bypass Valves begin OPENING.

j. WHEN MFRVs CLOSE, MFRV controllers transfer to MANUAL.

k. WHEN MFRVs CLOSED, MFRV Bypass Valve controllers transfer to AUTOMATIC.

5.8.23. ENSURE SG levels are stable.

5.8.24. Go to Step 5.8.27.

1 **ID: L17566** **Points: 1.00**

<QQ 17566(1410)><<Given the following plant conditions:

- A plant shutdown is in progress
- Reactor power is currently 25%
- Feed Pump Master Control DELTA P setpoint is currently 72 psid
- The transfer from MFRVs to MFRV Bypass Valves is being performed using the Automatic Method

What is the epxected response of the Feed Pump Master Control DELTA P setpoint after feed flow is shifted to the MFRV Bypass Valves?>>

A. <QQ 17566(1482)><<Raises to 215 psid>>

B. <QQ 17566(1480:0)><<Lowers to 45 psid>>

C. <QQ 17566(1480:1)><<Stays at 71 psid>>

D. <QQ 17566(1480:2)><<Raises to 149 psid>>

Answer: <QQ 17566 (1419) ><<A >>

Answer Explanation:

<QQ 17566(1412)><<***Explanation: Per OTN-AE-00001 when the automatic method of transfer from the MFRV the MFRV Bypass valves is used the Feed Pump Master Control DELTA P setpoint automatically raises to 215 psid.***
A. Incorrect – This is the minimum value for the Feed Pump Master Control DELTA P setpoint.
This is a plausible distractor if the student incorrectly assumes the system goes to minimum value when transfer to the MFRV bypass occurs
B. Incorrect – This is the current program value. This is a plausible distractor if the student incorrectly assumes the system maintains the current value when transfer to the MFRV bypass occurs
C. Incorrect – This is the maximum value for the Feed Pump Master Control DELTA P setpoint when the MFRV is in use. This is a plausible distractor if the student incorrectly assumes the system goes to maximum value for the MFRV when transfer to the MFRV bypass occurs
D. Correct – see explanation above>>

Question 1 Info	
Question Type:	<QQ 17566(1401)><<Multiple Choice>>
Status:	<QQ 17566(1405)><<Active>>
Always select on test?	<QQ 17566(1406)><<No>>
Authorized for practice?	<QQ 17566(1429)><<No>>
Points:	<QQ 17566(1441)><<1.00>>
Time to Complete:	<QQ 17566(1408)><<0>>
Difficulty:	<QQ 17566(1407)><<0.00>>
System ID:	<QQ 17566(1445)><<17566>>
User-Defined ID:	<QQ 17566(1404)><<L17566>>
Cross Reference Number:	
Topic:	<QQ 17566(1400)><<L17566 Response of MFP Master Control DP after shifting to Bypass Valves>>
Num Field 1:	<QQ 17566(1414)><<2.5>>
Num Field 2:	
Text Field:	<QQ 17566(1413)><<059 A3.04>>
Comments:	<QQ 17566(1411)><<2014 ILT NRC>>

Question 1 History	
Exam Appearances:	<QQ 17566(1449)><<0>>
Student Encounters:	<QQ 17566(1448)><<0>>
Answered Right:	<QQ 17566(1452)><<0>>
Answered Wrong:	<QQ 17566(1453)><<0>>
Partially Correct:	<QQ 17566(1459)><<0>>
Answer Invalid:	<QQ 17566(1455)><<0>>
Unanswered:	<QQ 17566(1454)><<0>>
Ignore Response:	<QQ 17566(1460)><<0>>
Avg Points Awarded:	<QQ 17566(1450)><<0.00>>
... As % of Point Value:	0
Standard Deviation:	<QQ 17566(1456)><<0.00>>

Question 1 Table-Item Links

<TB 5114(1301)><<OPS Procedures>>

<TB 8069(1305)><<OTN-AE-00001, FEEDWATER SYSTEM>>

<TB 5818(1301)><<OPS Systems>>

<TB 5823(1305)><<AE, Feedwater>>

<TB 5972(1301)><<OPS Question Catagory>>

<TB 5974(1305)><<LO Initial, Closed Book>>

<TB 5976(1305)><<NRC Quality>>

Associated objective(s):

<OB 16446(1101)><< E DESCRIBE the operation, including signal inputs, of the MFW pump speed control system and EXPLAIN the control response to input failures.>>

<OB 30131(1101)><<2014 ILT NRC Exam>>

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Main Feedwater	Group #	1		
	K/A #	059 A3.06		
	Importance Rating	3.2		
Ability to monitor automatic operation of the MFW, including: Feedwater isolation				

Question # 45

Reactor Power is 100%.

- A malfunction causes all S/G levels to lower to 5% NR level.
- The operating crew inserted a manual reactor trip during the transient.
- Auxiliary Feedwater has restored all S/G levels to 51% NR level.

(1) At SB069, the Lo Lo S/G Level Bistables are currently ….?

And

(2) What is the status of the S/G FRV Bypass Valves, AEFCV550/560/570/580?

A. (1) LIT
(2) The S/G FRV Bypass Valves will indicate RED

B. (1) LIT
(2) The S/G FRV Bypass Valves will indicate GREEN

C. (1) OFF
(2) The S/G FRV Bypass Valves will indicate RED

D. (1) OFF
(2) The S/G FRV Bypass Valves will indicate GREEN

Answer: D

Explanation:

Per OTO-SA-00001, Attachment K, step K.1 and the note prior to this step, the Bistables should be extinguished. This is because " The S/G Lo Lo Level FWIS Does not have a reset. When the S/G Lo Lo Level bistables on - SB069 are clear, the S/G Lo Lo Level FWIS will clear automatically." These Bistables and the associated FWIS reset automatically when SG NR level clears the Lo Lo setpoint of 21%. LIT is plausible if the candidate believes that manual action is required to reset the Lo Lo S/G FWIS as all other FWIS require some type of manual operator action to reset the signal.

Per OTO-SA-00001, Attachment AL, the S/G FRV Bypass Valves are closed (indicate GREEN) during a FWIS. RED (remaining open) is plausible if the candidate confuses these valves with valves in the SB blowdown system (which close on SGBIS) or confuses the indication they would observe on a control room panel.

A. Incorrect – both are wrong – see above explanation
B. Incorrect – the bistables would be off as SG NR level has returned to program level
C. Incorrect – The S/G FRV Bypass valves would be Green as they are closed
D. Correct – see above explanation

Technical Reference(s):

1. OTA-RK-00026, Addendum 126B, Rev B
2. OTO-SA-00001, ESFAS Verification and Restoration, Rev 39 Attachment AL, AQ,and K
3. 7250D64 S013, Functional Diagram Feedwater Control and Isolation, Rev 15
4. E-0, Reactor Trip or Safety Injection, Rev 16 Step #7

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #23, Main Feedwater, Objective C:
STATE the conditions, including the setpoints and coincidences, that will cause a FWIS and EXPLAIN the system response to the signal.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT K
(Page 1 of 2)
FWIS Actuation Recovery

NOTES

- LO TAVG & P-4 FWIS - Can be reset, with conditions present by using the FW ISO RESET pushbutton. Once FW ISO RESET has been pushed and either condition clears, the signal is ready to process another LO TAVG & P-4 FWIS. This is the ONLY FWIS reset by the FW ISO RESET Pushbuttons on RL018. If Trip Breakers are opened with a LO TAVG, a LO TAVG & P-4 FWIS (MCB ann. 126B) will be generated. S701 switches in SB029D and SB032D may be placed in BYPASS to prevent a LO TAVG & P-4 FWIS when opening Reactor Trip Breakers during shutdown plant conditions.

- SI and P-14 (Hi S/G Level) FWIS - This FWIS is sealed in by P-4. The Reactor Trip Breakers are reclosed to clear the seal in if a P-4 occurred or was present.

- The S/G Lo Lo Level FWIS - Does not have a reset. When the S/G Lo Lo Level bistables on SB069 are clear, the S/G Lo Lo Level FWIS will clear automatically.

K1. CHECK The Following Conditions Are Met On SB069 To Reset FWIS:

- Lo Lo S/G Level Bistables for all S/G's- EXTINGUISHED

- Hi Hi S/G Level Bistables (P-14) for all S/G's - EXTINGUISHED

- NO Active SI Signals present

ATTACHMENT AL
(Page 1 of 2)
FWIS Verification

NOTES

- (1) Train A only (SA066X).
- (2) Train B only (SA066Y).
- (3) Does not have ESFAS status panel indication.

AL1. FWIS Verification:

- 13—Q AEFV43, 'A' S/G CHEM INJ
  - AE HIS—43 — CLOSED See Note (1) [Normal position: Closed]
- 14—Q AEFV45, 'C' S/G CHEM INJ
  - AE HIS—45 — CLOSED See Note (1) [Normal position: Closed]
- 13—Q AEFV44, 'B' S/G CHEM INJ
  - AE HIS—44 — CLOSED See Note (2) [Normal position: Closed]
- 14—Q AEFV46, 'D' S/G CHEM INJ
  - AE HIS—46 — CLOSED See Note (2) [Normal position: Closed]
- 13—L AEFV39, 'A' S/G FWIV
  - AE HIS—39 — CLOSED [Normal position: Open]
- 13—M AEFV40, 'B' S/G FWIV
  - AE HIS—40 — CLOSED [Normal position: Open]
- 13—N AEFV41, 'C' S/G FWIV
  - AE HIS—41 — CLOSED [Normal position: Open]
- 14—N AEFV42, 'D' S/G FWIV
  - AE HIS—42 — CLOSED [Normal position: Open]
- AEFCV510, 'A' S/G MFRV
  - AE FK—510 — CLOSED See Note (3) [Normal position: Open]

(Step AL1. continued on next page)

ATTACHMENT AL
(Page 2 of 2)
FWIS Verification

Step AL1. (continued from previous page)

- AEFCV520, 'B' S/G MFRV
  - AE FK-520 - CLOSED See Note (3) [Normal position: Open]
- AEFCV530, 'C' S/G MFRV
  - AE FK-530 - CLOSED See Note (3) [Normal position: Open]
- AEFCV540, 'D' S/G MFRV
  - AE FK-540 - CLOSED See Note (3) [Normal position: Open]
- AEFCV550, 'A' S/G FRV BYPASS
  - AE LK-550 - CLOSED See Note (3) [Normal position: Closed]
- AEFCV560, 'B' S/G FRV BYPASS
  - AE LK-560 - CLOSED See Note (3) [Normal position: Closed]
- AEFCV570, 'C' S/G FRV BYPASS
  - AE LK-570 - CLOSED See Note (3) [Normal position: Closed]
- AEFCV580, 'D' S/G FRV BYPASS
  - AE LK-580 - CLOSED See Note (3) [Normal position: Closed]

-END-

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Auxiliary/Emergency Feedwater	Group #	1		
	K/A #	061 K5.02		
	Importance Rating	3.2		
Knowledge of the operational implications of the following concept as they apply to the AFW: Decay heat sources and magnitude				

Question # 46

Per the FSAR, the Auxiliary Feed System is designed so that a MINIMUM of ____(1)____ MDAFW pump(s) can sufficiently remove decay heat and cooldown the RCS at ____(2)_____ °F/hr within 1 hour of a Reactor trip from 100% power.

A. (1) 1
   (2) 50

B. (1) 2
   (2) 50

C. (1) 1
   (2) 100

D. (1) 2
   (2) 100

Answer: A

Explanation:

UFSAR Section 10.4.9.2.1, Each motor-driven auxiliary feedwater pump will supply 100 percent of the feedwater flow required for removal of decay heat from the reactor. The turbine-driven pump is sized to supply up to twice the capacity of a motor-driven pump. This capacity is sufficient to remove decay heat and to provide adequate feedwater for cooldown of the reactor coolant system at 50°F/hr within 1 hour of a reactor trip from full power. 100F.hr is plausible as that is the technical specification cooldown limit.

A. Correct – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. FSAR Section 10.4.9.2.1

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #25, Aux Feedwater, Objective A
A: STATE the function of the Auxiliary Feedwater (AFW) System.

Question Source: Bank # __X L16606____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2009________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Replaced question per NRC comment

assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

SAFETY DESIGN BASIS SIX - The AFS, in conjunction with the condensate storage tank (nonsafety-related and not credited for accident mitigation) or essential service water system, which is credited for accident mitigation, provides feedwater to maintain sufficient steam generator level to ensure heat removal from the reactor coolant system in order to achieve a safe shutdown following a main feedwater line break, a main steamline break, or an abnormal plant situation requiring shutdown. The auxiliary feedwater system is capable of delivering full flow when required, after detection of any accident requiring auxiliary feedwater (refer to Chapter 15.0).

SAFETY DESIGN BASIS SEVEN - The capability to isolate components or piping is provided, if required, so that the AFS safety function will not be compromised. This includes isolation of components to deal with leakage or malfunctions and to isolate portions of the system that may be directing flow to a broken secondary side loop.

SAFETY DESIGN BASIS EIGHT - The AFS has the capacity to be operated locally as an alternate, redundant means of feedwater control, in the unlikely event that the control room must be evacuated.

10.4.9.1.2 Power Generation Design Bases

The condensate and feedwater system is designed to provide a continuous feedwater supply to the steam generators during startup normal plant operation, and shutdown. If the normal motor-driven startup feedwater pump is not available the AFS may be operated with the auxiliary feedwater pump discharge valves throttled when the reactor is below 10% power to maintain steam generator water levels during plant heatups or cooldowns. Refer to Section 10.4.7.

10.4.9.2 <u>System Description</u>

10.4.9.2.1 General Description

The system consists of two motor-driven pumps, one steam turbine-driven pump, and associate piping, valves, instruments, and controls, as shown on Figure 10.4-9 and described in Table 10.4-12. Figure 10.4-10 shows the piping and instrumentation for the steam turbine.

Each motor-driven auxiliary feedwater pump will supply 100 percent of the feedwater flow required for removal of decay heat from the reactor. The turbine-driven pump is sized to supply up to twice the capacity of a motor-driven pump. This capacity is sufficient to remove decay heat and to provide adequate feedwater for cooldown of the reactor coolant system at 50°F/hr within 1 hour of a reactor trip from full power.

The nonsafety-related condensate storage tank (CST) provides a source of water to the auxiliary feedwater pumps. However, since this tank is not seismic Category I and not credited for accident mitigation, two redundant safety-related back-up sources of water from the essential service water system (ESWS) are provided for the pumps. For a more detailed description of the automatic sequence of events, refer to Section 10.4.9.2.3.

The condensate storage tank capacity allows the plant to remove decay heat from the primary system during a 4 hour Station Blackout event, as discussed in Table 8.3A-1, item III.A. Refer to Section 9.2.6 for a description of the condensate storage system.

The non-safety auxiliary feedwater pump (NSAFP) can be manually aligned to provide an alternate source of cooling water to the steam generators through the Auxiliary Feedwater System as shown in Figure 10.4-9. The NSAFP will be aligned upon the following events occurring simultaneously: loss of offsite power, loss of onsite power, and failure of the turbine-driven auxiliary feedwater pump.

In order to remove decay heat by the steam generators, auxiliary feedwater must be supplied to the steam generators in the event that the normal source of feedwater is lost. The minimum auxiliary feedwater flow rate required to fulfill the acceptance criteria for the heatup events can be found in Section 15.2.

Provisions are incorporated in the AFS design to allow for periodic operation to demonstrate performance and structural and leaktight integrity. Leak detection is provided by visual examination and in the floor drain system described in Section 9.3.3.

10.4.9.2.2 Component Description

Codes and standards applicable to the AFS are listed in Tables 3.2-1 and 10.4-12. The AFS is designed and constructed in accordance with quality groups B and C and seismic Category I requirements.

MOTOR-DRIVEN PUMPS - Two auxiliary feedwater pumps are driven by ac-powered electric motors supplied with power from independent Class 1E switchgear busses. Each horizontal centrifugal pump takes suction from the nonsafety-related condensate storage tank, or alternatively, from the ESWS. Pump design capacity includes minimum flow recirculation, which is controlled by automatic recirculation control check valves.

TURBINE-DRIVEN PUMP - A turbine-driven pump provides system redundancy of auxiliary feedwater supply and diversity of motive pumping power. The pump is a horizontal centrifugal unit. Pump bearings are cooled by the pumped fluid. Pump design capacity includes continuous minimum flow recirculation. AC powered valves required for operability of the turbine driven pump are aligned in accordance with Technical Specifications such that their positions are not required to change upon a loss of all ac power. Air operated valves, controls and instrumentation required for operation of the turbine driven pump are powered by the Class 1E dc system or dc backed vital ac system. Swapover to ESW supply is not postulated during a loss of all ac power as

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Auxiliary/Emergency Feedwater	Group #	1		
	K/A #	061 A1.01		
	Importance Rating	3.9		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: S/G level				

Question # 47

Reactor Power is 100%.

- The TDAFP is out of service for repair.
- A transient occurs and the reactor and main turbine trip.
- A lockout of NB01 occurs and power cannot be restored.

The Reactor Operator will be able to control Auxiliary Feedwater flow to which Steam Generators (SGs)?

A. A and B

B. A and D

C. B and C

D. B and D

Answer: B

Explanation:

The 'A' MDAFP can provide flow to the B and C SGs and is powered from NB01. The 'B' MDAFP can provide flow to the A and D SGs and is powered from NB02. The TDAFP can supply water to all 4 SGs. The steam supply to the TDAFP is from the B and C SGs.

With the above combinations of plant events in the stem, only the 'B' MDAFP is available to provide AFW flow; therefore only SGs A and D will receive AFW flow.

The distractors are either steam supplies for the TDAFP or the SGs that the 'A' MDAFP can supply or some combination of these 2 items.

A. Incorrect – see above explanation. While A S/G does receive AFW flow from the 'B' MDAFP, the B SG does not making this an incorrect choice.
B. Correct – 'B' MDAFP will only supply the A and D SGs
C. Incorrect – these are the SGs that supply steam to the TDAFP

D. Incorrect – these are the SGs that the 'A' MDAFP pump supplies AFW to.

Technical Reference(s):
1. M22AL01, P&ID, Auxiliary Feedwater System, Rev 44

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #25, Aux Feedwater, Objective D:
Explain the operation of the auxiliary Feedwater system under the following conditions:
- Normal operation
- Low-Low steam generator level
- AFAS
- SIS
- Loss of off-site power
- Low CST level (Low Suction Pressure to Pumps)

Question Source: Bank # _X L13502_____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2005_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(8)

Comments:

Revised question per NRC comment on explanations. Also removed SG is each choice asit was redundant to the stem.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
AC Electrical Distribution	Group #	1		
	K/A #	062 K4.10		
	Importance Rating	3.1		
Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Uninterruptable ac power sources				

Question # 48

Reactor power is 100% when a transient results in the loss of 125 VDC Bus NK04.

Power will be restored to 120 VAC Bus NN04 after the Static Transfer switch is _____(1)_____ transferred to _____(2)_____.

A. (1) manually
(2) the bypass transformer

B. (1) manually
(2) transformer XNN06

C. (1) automatically
(2) the bypass transformer

D. (1) automatically
(2) transformer XNN06

Answer: C

Explanation:

Per OTN-NN-00004, Attachment 1 (diagram of Train B NN Inverter UPS Schematic), the static transfer switch operates to either have the inverter supplying NN04 or the constant voltage transformer (aka the bypass source or bypass transformer) in service supplying NN04. XNN06 transformer is an alternate source but must be placed in service with the sliding link interlock not the static transfer switch. Step #3.10 of OTN-NN-00004, lists in order the preferred NN Bus power supplies for reference.

Therefore, the bypass transformer will be the power alternate AC source for the UPS. The XNN06 transformer is plausible as it is an alternate source but will not automatically come online, manual action using the sliding link is required.

Step #3.11 of OTN-NN-00004, shows the conditions that will cause the static transfer switch to automatically transfer. With a loss of the inverter feed NK04, an inverter undervoltage will occur *causing an automatic transfer of the static transfer switch. Manual transfer is plausible as there are several manual switch actions that can occur on the UPS such as S1, manual bypass switch, sliding link and its associate kirk key interlock*

A. Incorrect – the static switch will automatically transfer
B. Incorrect – both are wrong
C. Correct
D. Incorrect – wrong alternate power source

Technical Reference(s):

1. OTO-NK-00002, Loss of Vital 125 DC Bus, Rev 14
2. OTN-NN-00004, 120 Vital AC Instrument Power Class 1E (Channel 4), Rev6

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #6 Safeguards Power, Objective B: DESCRIBE the purpose and operation of the following Safeguards Power System components and subsystems:

5. 125 VDC System (NK)
6. 120 VAC System (NN)

Question Source: Bank # __X L16236____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Revised A and C choices by moving "the" from the stem to A &C per NRC Comment

3.8. If the IN SYNC amber light (P11) is off:

3.8.1. The following pushbuttons should NOT be operated:

- S201, INVERTER TO LOAD
- S202, BYPASS SOURCE SUPPLYING LOAD

3.8.2. Maintenance should be contacted.

3.9. S1, MANUAL BYPASS SWITCH, is a make before break switch. To prevent personal injury:

- It should NOT be transferred to NORMAL SOURCE unless the IN SYNC amber light (P11) is on.
- It should NOT be transferred between NORMAL SOURCE and BYPASS SOURCE without the Inverter Static Transfer Switch in BYPASS SOURCE SUPPLYING LOAD - red light (P202) is on.

3.10. The following are the preferred methods for supplying the associated NN Bus with Alternate, Manual Bypass or Backup power sources.

- First Choice - Alternate
  Using the Static Transfer Switch to supply the NN Bus via the alternate power source NG Bus.
- Second Choice – Manual Bypass
  Using S1, MANUAL BYPASS SWITCH, to bypass the inverter and the Static Transfer Switch to supply the NN Bus from the bypass power source NG Bus.
- Third Choice – Backup (Maintenance Bypass)
  Using the sliding link breakers to supply the NN Bus from the backup power source XNN06 Transformer (per OTS-NN-00014, NN14 Inverter Outage).

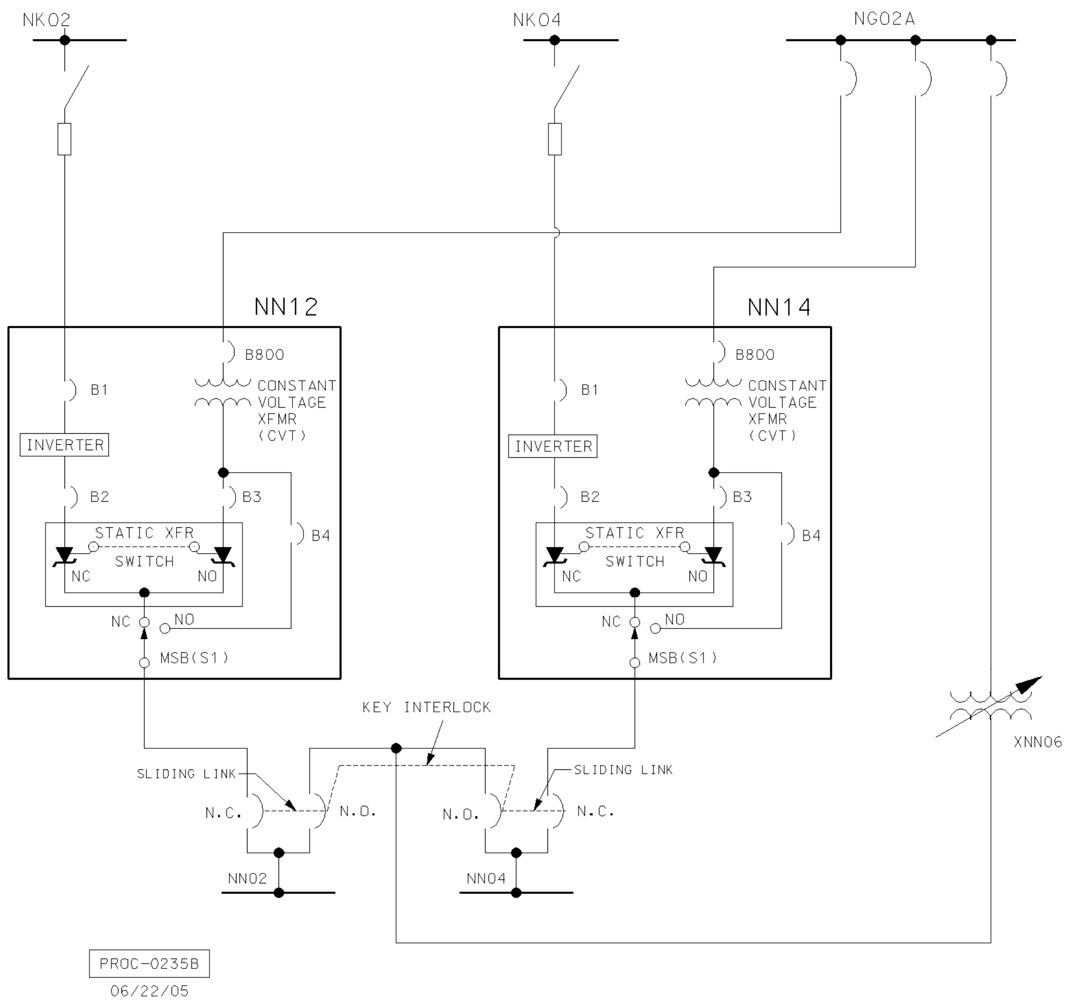
3.11. The Static Transfer Switch:

- Automatically transfers to BYPASS SOURCE SUPPLYING LOAD - red light (P202) is on for any of the following conditions:
  - Inverter output overcurrent
  - Inverter undervoltage on bridge output
  - Inverter undervoltage on inverter output
- Does NOT automatically transfer back to INVERTER SUPPLYING LOAD - amber light (P201) on.

Attachment 1

Train B NN Inverter UPS Schematic

Sheet 1 of 1



CONTINUOUS USE

Attachment 2

Inverter Alignment Conditions

Sheet 1 of 1

NOTE

During Plant Operations, the associated 120VAC NN buses are in service and are energized from one of the following power SOURCES:

Condition One

NORMAL power source from NK Buses via an NN Inverter with the STATIC TRANSFER SWITCH in INVERTER SUPPLYING LOAD (amber light P201 is on). S1, MANUAL BYPASS SWITCH, is in NORMAL SOURCE. INVERTER IN SERVICE.

Inverter	Source	Bus
NN14	NK0411, FDR BKR TO 7.5 KVA INVERTER NN14	NN04

Condition Two

ALTERNATE power source from NG Bus bypassing the inverter via the Static Transfer Switch in BYPASS SOURCE SUPPLYING LOAD (P202 red light on). S1, MANUAL BYPASS SWITCH, is in NORMAL SOURCE. INVERTER NOT IN SERVICE.

Inverter	Source	Bus
NN14	NG02AGF3, 480 V SUPPLY TO NN14 BYPASS REGULATING TRANSFORMER	NN04

Condition Three

BYPASS power source from NG Bus bypassing the Static Transfer Switch(1) via S1, MANUAL BYPASS SWITCH, in BYPASS SOURCE. INVERTER AND STATIC TRANSFER SWITCH NOT IN SERVICE.

Inverter	Source	Bus
NN14	NG02AGF3, 480 V SUPPLY TO NN14 BYPASS REGULATING TRANSFORMER	NN04

(1) The Static Transfer Switch should be in BYPASS SOURCE SUPPLYING LOAD (P202 light on).

Condition Four

BACKUP power source from Bus NG02A via transformer XNN06(3) with the sliding interlock link in the "backup source breaker" position. INVERTER AND STATIC TRANSFER SWITCH NOT IN SERVICE.

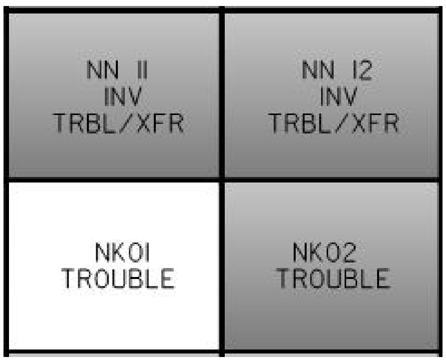
Inverter	Source	Bus
NN14	NG02AFF3, FDR BKR TO XNN06 ALT FEED TO NN02 AND NN04	NN04

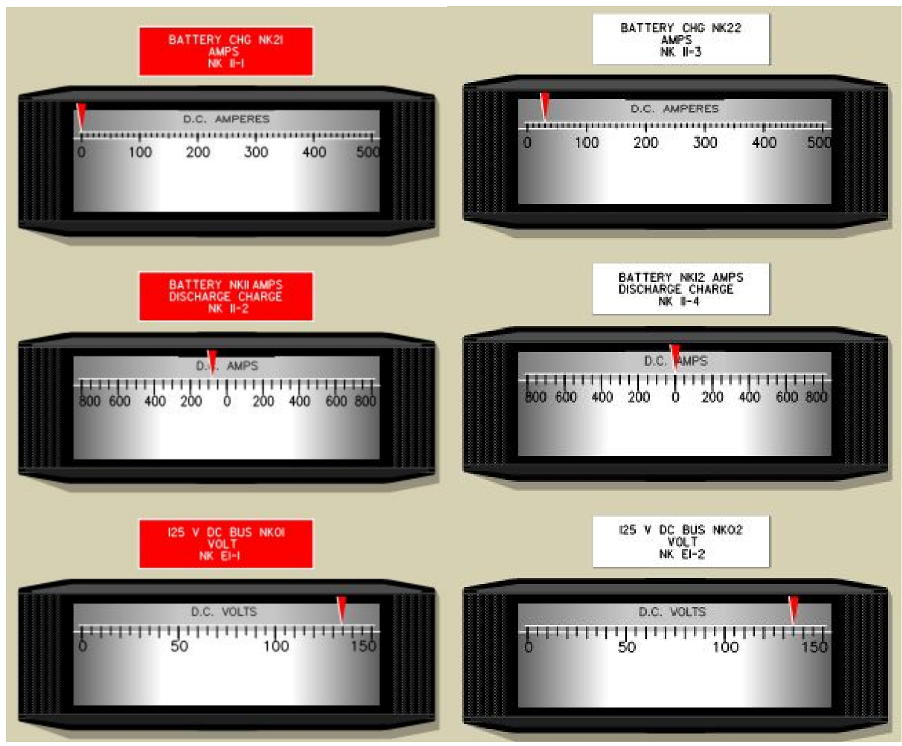
(3) The backup source can only supply one NN Bus at a time.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
063 DC Electrical Distribution	Group #	1		
	K/A #	063 A3.01		
	Importance Rating	2.7		
Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights				

Question # 49

The plant is operating at 100% power when the following indications are observed by the crew:





Based on these conditions, 120V AC Bus NN01 is being supplied power from ____(1)____, and the crew will implement ______(2)_____ ?

A. (1) NK01
(2) OTO-NK-00001, Failure of NK Battery Charger

B. (1) NK01
(2) OTO-NK-00002, Loss of Vital 125VDC Bus

C. (1) NG01A

(2) OTO-NK-00001, Failure of NK Battery Charger

D. (1) NG01A
(2) OTO-NK-00002, Loss of Vital 125VDC Bus

Answer: A

Explanation:

A. Correct. Based on the indications of the annunciators NN11 inverter trouble/transfer and NK01 Trouble, it can be determined that a malfunction on the DC electrical system has occurred. With the indications of NK11 showing a discharge and bus voltage, with no current flow from the charger NK21, it is indicative of the battery supplying power to NK01 which in turn is supplying power to NN01 due the normal lineup. The indications of NK12 and NK02 are shown for comparison to normal values. Entry conditions for OTO- NK-00001, Failure of NK Battery Charger are met and entry to this procedure is required.
B. Incorrect. Indications show that NK01 still has bus voltage and therefore has not lost power, so entry into OTO-NK-00002, Loss of Vital 125VDC Bus is an incorrect action.
C. Incorrect. Plausible because NG01A is the alternate power supply through both the static transfer switch and the SOLA Transformer. In the event NK01 is lost, then NG01A will be supplying NN01 via the static transfer switch. If the static transfer switch is lost then the SOLA transformer can be placed in service to directly supply NN01 via a manual breaker transfer on NN01. See explanation A &B.
D. Incorrect. Plausible because NG01A is the alternate power supply through both the static transfer switch and the SOLA Transformer. In the event NK01 is lost, then NG01A will be supplying NN01 via the static transfer switch. If the static transfer switch is lost then the SOLA transformer can be placed in service to directly supply NN01 via a manual breaker transfer on NN01. See explanation A &B.

Technical Reference(s):
1. OTO-NK-00001, Failure of NK Battery Charger Rev 13

References to be provided to applicants during examination: None

Learning Objective: T61.003B LP-B-26, Obj. B. Describe symptoms or entry conditions for OTO-NK-00001, Failure of NK Battery Charger.

Question Source: Bank # __X_L17571___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2014________

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

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:
Question #6 on the RO 2014 IL exam

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Emergency Diesel Generator	Group #	1		
	K/A #	064 K6.08		
	Importance Rating	3.2		
Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks				

Question # 50

Reactor Power is 100%.

- A leak exists in the 'A' EDG Fuel Oil Storage Tank.
- Annunciator 89B, DG Fuel Tank A Level Low, is LIT.
- Currently, 'A' EDG Fuel Oil fuel tank volume is 82,000 gallons and is lowering at 100 gallons per hour.
- The 'B' EDG, NE02, is Operable.

(1) What is the MAXIMUM length of time until the Technical Specification Condition for the 'A' EDG Fuel Oil Storage Tank is NOT met?

And

(2) With the 'B' EDG remaining Operable, in what MODES is the 'A' EDG Fuel Oil Storage Tank level REQUIRED to be within its limits?

A. (1) 5 hours
(2) MODES 1,2,3 ONLY

B. (1) 5 hours
(2) MODES 1 through 4

C. (1) 11 hours
(2) MODES 1,2,3 ONLY

D. (1) 11 hours
(2) MODES 1 through 4

Answer: D

Explanation: *Per the OTA (Annunciator 89B) note prior to step 3.1, T.S. 3.8.3 requires 80,900 gallons of fuel oil. This is the same volume required per T.S. 3.8.3. Condition A. This means* *there is 1100 gallons of margin and at 100 gallons per hour; there is 11 hours before the LCO is not met.*

The distractor of 5 hours was chosen as it is approximately one half of the correct answer. There are no other volume set points before the 80,900 that are of T.S. concern. The candidate must know the T.S required volume to correctly answer the question, 5 hours corresponds to a volume of 81,500 gallons which is plausible number.

Per T.S. 3.8.3, the stored diesel fuel oil is required to be within limits for each required EDG "When the associated EDG is required to be OPERABLE". T.S. 3.8.1, AC Sources Operating, 2 EDG are required to be Operable in Modes 1 through 4. Only one EDG is required in MODEs 5 & 6 per T.S. 3.8.2. and the reason why the B EDG was listed as Operable in the stem.

MODEs 1,2,3 ONLY are plausible as both trains of ECCS equipment are REQUIRED to be operable in these modes (per T.S. 3.5.2; ECCS – Operating) and the EDGs provide emergency power for this equipment.

A. Incorrect – both are wrong
B. Incorrect – wrong time
C. Incorrect – wrong modes of applicability
D. Correct – see above explanation

Technical Reference(s):
1. OTA-RK-00024, Addendum 89B, DG Fuel TK A Level Lo, Rev 4
2. OTN-JE-00001, Emergency Fuel Oil Storage and Transfer System, Rev 11
3. Technical Specifications 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air
4. Technical Specifications 3.8.1, AC Sources - Operating

References to be provided to applicants during examination: None

Learning Objective: T61.0110.6, Systems, LP #3, Standby Generation, Objective O: STATE the applicable Technical Specifications for the Standby Diesel Generators.

Question Source: Bank # _ _____
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Revised part 2 of the question and updated the stem of part 1 per NRC comments

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

-- NOTE --
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with fuel level < 80,900 gal and > 69,800 gal in storage tank.	A.1 Restore fuel oil level to within limits.	48 hours
B. One or more DGs with lube oil inventory < 750 gal and > 686 gal.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One or more DGs with stored fuel oil total particulates not within limit.	C.1 Restore fuel oil total particulates within limit.	7 days
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more DGs with two starting air receivers in service with pressure < 435 psig and ≥ 250 psig. <u>OR</u> One or more DGs with only one starting air receiver in service with pressure < 610 psig and ≥ 300 psig.	E.1	Restore two starting air receivers with pressure ≥ 435 psig.	48 hours
	<u>OR</u>		
	E.2	Restore one starting air receiver with pressure ≥ 610 psig.	48 hours
F. Required Action and associated Completion Time not met. <u>OR</u> One or more DGs diesel fuel oil, lube oil, or starting air subsystems not within limits for reasons other than Condition A, B, C, D, or E.	F.1	Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 80,900 gal of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	Verify lubricating oil inventory is ≥ 750 gal.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify pressure in two starting air receivers is ≥ 435 psig or pressure in one starting air receiver is ≥ 610 psig, for each DG starting air subsystem.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.6	Not used.	

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Process Radiation Monitoring	Group #	1		
	K/A #	073 G2.4.31		
	Importance Rating	4.2		
Knowledge of annunciator alarms, indications, or response procedures.				

Question # 51

Reactor Power is 100%.

RM-11, Radiation Monitor Control Panel, has been declared inoperable.

(1) For process radiation monitors required by technical specifications, Operators will be required to establish and perform OSP-ZZ-00001 Attachment 8, Rad Monitor Communication Failure Log, a MINIMUM of every ____(1)_____ or declare the monitor inoperable and comply with technical specifications?

And

(2) With the RM-11 inoperable, what is the FIRST control room indication of a fuel handling incident in the fuel building?

A. (1) 30 minutes
(2) elevated RM-23 readings

B. (1) 30 minutes
(2) Annunciator 61B, Process Rad HI, in alarm

C. (1) 4 hours
(2) elevated RM-23 readings

D. (1) 4 hours
(2) Annunciator 61B, Process Rad HI, in alarm

Answer: A

Explanation:
Note: All of the Annunciators, 61A- 61C, direct the operator to OTA-SP-RM011

Section 3.4 of OTA-SP-RM011, section 3.4.1 step a directs the establishment of the log every 30 minutes. The distractor of 4 hours is from OSP-ZZ-00001, Attachment 6 for an inoperable rod position deviation monitor. The "or declare the monitor inoperable and comply with technical *specifications" is to account for the 4 PRMs that don't have RM-23s that are discussed on section 3.4.2.*

Per the note in OTA-SP-RM011 section 3.4, "With the RM-11 out of service, main control board annunciators 61A, PROCESS RAD HIHI; 61B, PROCESS RAD HI; and 61C, PROCESS RAD MON FAIL, will be inoperable." and therefore will not respond to the transient. The first indication will be the RM-23 units for GG-RE-27 and 28, Fuel Building Ventilation PRMs. The annunciator 61B is plausible as it would be the first indication if the RM-11 was operable. For part b, Fuel Building radiation monitors were selected as they have RM-23 units.

A. Correct
B. Incorrect- Per the note, annunciator 61B would not work if RM-11 is inoperable
C. Incorrect – wrong time
D. Incorrect – both are wrong

Technical Reference(s):
1. OTA-SP-RM011, Radiation Monitor Control Panel RM-11
2. OTA-RK-00020, Addendum 61A, PR Hi Hi, Rev 0
3. OTA-RK-00020, Addendum 61B, PR Hi, Rev 0
4. OTA-RK-00020, Addendum 61B, PR Hi, Rev 0
5. OSP-ZZ-00001, CR Shift and Daily Log Reading and Channel Checks, Rev 87

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #36, Process and Area Radiation Monitors, Objective C: IDENTIFY the Process and Area Radiation Monitoring Control Room controls, alarms, and indications and DESCRIBE how each is used to predict, monitor and control the Process and Area Radiation Monitoring System.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(11)

Comments:

3.4. RM-11 Inoperability Compensatory Actions

NOTE

With the RM-11 out of service, main control board annunciators 61A, PROCESS RAD HIHI; 61B, PROCESS RAD HI; and 61C, PROCESS RAD MON FAIL, will be inoperable.

ESFAS actuations and control features are NOT affected by RM-11 operability because the RM-80 units send these signals directly to ESFAS or the components.

3.4.1. Technical Specification / FSAR 16.0 CHAPTER Monitors with RM-23 units

Monitor	Process
GG-RE-27	Fuel Building Ventilation
GG-RE-28	Fuel Building Ventilation
GH-RE-10B	Radwaste Vent
GK-RE-04	Control Room Ventilation
GK-RE-05	Control Room Ventilation
GT-RE-31	Containment Atmosphere
GT-RE-32	Containment Atmosphere
GT-RE-22	Containment Purge
GT-RE-33	Containment Purge
GT-RE-59	CHARMS
GT-RE-60	CHARMS
GT-RE-21B	Unit Vent

NOTE

The monitors are operable as long as their respective RM-23 units are operable.

OSP-ZZ-00001 Attachment 8 is SAT if the green operate light is lit.

If performing Shift and Daily log readings, channel check surveillances are required to be performed using RM-23s.

a. ESTABLISH a 30 minute surveillance for the above monitors per OSP-ZZ-00001, Control Room Shift And Daily Log Readings And Channel Checks Attachment 8, Rad Monitor Communication Failure Log.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Service Water	Group #	1		
	K/A #	076 K1.21		
	Importance Rating	2.7		
Knowledge of the physical connections and/or cause- effect relationships between the SWS and the following systems: Auxiliary backup SWS				

Question # 52

Reactor Power is 100%.

- 2 Service Water Pumps are running.
- An NB02 undervoltage condition occurs.

What is the status of and actions required for the Service Water system?

A. Service Water will be supplying Turbine building loads ONLY. Secure one service water pump within 5 minutes.

B. Service Water will be supplying Turbine building loads and "A" Train ESW loads. Secure one service water pump within 5 minutes.

C. Service Water will be supplying Turbine building loads ONLY. Open the Cooling Tower Bypass Valves to lower service water pressure.

D. Service Water will be supplying Turbine building loads and "A" Train ESW loads. Open the Cooling Tower Bypass Valves to lower service water pressure.

Answer: A

Explanation:

As a result of the undervoltage, the B EDG will start and close in on the NB02 which will start the shutdown sequencer which will start the B ESW pump. This will cause ESW to isolate from service water as both the supply and return cross connects will close. Note: Each train has 2 cross connects on the supply and the return side, one of the supply and one return are opposite train powered and logic controlled. This in turn would start the opposite train ESW pump due to undervoltage on the opposite train bus (NB02) with low flow through the A trains containment cooler. So the final configuration would be both ESW pumps running each supplying their own trains loads.

While it is listed in both OTN-EA-00001 and OTN-EF-00001, a caution in OTN-EF-00001 explains it the best " If this will be the second ESW pump started and two SW pumps will be supplying only *the Turbine Building loads, a SW pump will have to be secured within five minutes of ESW Pump start."*

The distractor of opening the cooling tower bypass valve is plausible as the Service water and Circ Water lines come together and return to the cooling tower and if the candidate believes there is procedural direction to open this valve to divert service water to the cooling tower and hence lower service water pressure, it is plausible. The bypass valve is only done for cold weather operation to raise cooling tower basin temperature per OTN-DA-00001 ADD 4.

The distractor of SW still supplying A Train ESW load is plausible this is the normal flowpath and the stem does not provide any A Train cues; i.e. the candidate does not understand or remember the cross train powered valves for the supply or return cross connects nor the ESW pumps auto start on opposite train UV and low flow (reference OTA-RK-00020, ADD 54A)

A. Correct – See above explanation
B. Incorrect – this is the wrong flowpath
C. Incorrect – wrong action as securing one of the SW pumps is required
D. Incorrect – both are wrong

Technical Reference(s):
1. OTN-EA-00001, Service Water System, Rev 37
2. OTN-EF-00001, Essential Service Water, Rev 70
3. M-22EF01, P&ID, ESW, Rev 79
4. M-22EF02, P&ID, ESW, Rev 75
5. OTA-RK-00020, Addendum 54A, ESW A Pressure Low / Flow Low, Rev 2
6. OTN-DA-00001, Addendum 4, Cooling Tower Operation, Rev 11

References to be provided to applicants during examination: None

Learning Objective: T61.0110, System, LP #4 Circ and Service Water, Objective F & G:

F: LIST the systems that interface with the Circulating and Service Water Systems and EXPLAIN how a loss of the interfacing system or a loss of the Circulating or Service Water Systems affects the other.

G.DESCRIBE the response of the Service Water System to a Safety Injection Signal, Loss of Offsite Power, Aux Feed Low Suction Pressure.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam __N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:
In terms of this K/A, the Service Water (SW) systems is the SWS and the ESW system is the auxiliary backup SWS.

The undervoltage condition in the question stem is merely for the auto start of the ESW pump in the same train (B Train) and provides half of the A Trains ESW start signal (low flow still needed but will occur when ESW trains disconnect from SW). The UV sets up the cause and effect situation between SW and ESW i.e. what is the status of the SW system when the auxiliary SW train(s) receive an autostart signal.

5.0 PROCEDURE INSTRUCTIONS

5.1. ESW Train A - Manual System Operation

CAUTION

Shifting ESW supply to and from SW on the train of CCW supplying the service loop could impact letdown temperature and affect core reactivity. The letdown demineralizers release boron on rising temperature and absorb boron on lowering temperature. Therefore, operation of BG TK-130, LTDN HX OUTLET TEMP CTRL, should be closely monitored to ensure letdown temperature remains stable as read on BG TI-130, LTDN HX OUTLET TEMP.

If this will be the second ESW pump started and two SW pumps will be supplying only the Turbine Building loads, a SW pump will have to be secured within five minutes of ESW Pump start.

5.1.1. ENSURE EF HIS-51, ESW TRN A TO CCW HX A, is OPEN.

5.1.2. ENSURE EF HIS-59, ESW TRN A FROM CCW HX A, is CLOSED.

NOTE

When SW is supplying ESW, operating experience indicates that starting an ESW Pump could actuate MCB Annunciator 12D, SERV WTR PMP TROUBLE, or 13D, CIRC WTR PMP TROUBLE.

5.1.3. Using EF HIS-55A, ESW PUMP A, START PEF01A and CHECK pump is running as indicated by the following:

- EF PI-1, ESW PUMP A DISCH PRESS
- EF FI-53, ESW PUMP A DISCH FLOW

5.1.4. Using the following, CLOSE EFHV0023 and EFHV0025:

- EF HIS-23, SERVICE WTR/ESW TRN A CROSS CONNECT
- EF HIS-25, SERVICE WTR/ESW TRN A CROSS CONNECT

5.1.5. Using EF HIS-37, ESW TRN A TO UHS, OPEN EFHV0037.

5.1.6. Using the following, CLOSE EFHV0039 and EFHV0041:

- EF HIS-39, ESW TRN A TO SERVICE WTR SYS
- EF HIS-41, ESW TRN A TO SERVICE WTR SYS

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Service Water	**Group #**	1		
	K/A #	076 K4.02		
	Importance Rating	2.9		
Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Automatic start features associated with SWS pump controls				

Question # 53

A reactor trip and safety injection have occurred.

How will the Essential Service Water (ESW) Pumps, PEF01A & B, respond to this event?

A. The "A" ESW Pump starts at 5 seconds and the "B" ESW Pump starts at 10 seconds.

B. Both ESW Pumps start at 15 seconds.

C. Train "A" ESW starts at 20 seconds, and Train "B" ESW starts at 25 seconds.

D. Train "B" ESW starts at 20 seconds, and Train "A" ESW starts at 25 seconds.

Answer: C

Explanation:
See E-22NF01 for the LOCA sequencer timer start points. The A starts @20 seconds and the B starts 5 seconds later at the 25 second point.

A. Incorrect – This is for the CCW pumps applied to a single train. If a train specific CCW punps fails to auto start at 5 seconds then the backup pump will start at the 10 second point
B. Incorrect – The containment spray pumps would autostart at the 15 second period if a Containment Spray Actuation signal was present.
C. Correct
D. Incorrect – these are the correct start times for the ESW pumps but not the correct pump at the correct time.

Technical Reference(s):
1. E-22NF01, Load Shedding and Emergency Load Sequencing Logic, Rev 8

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.0110, Systems, LP #7 Essential Service Water, Objective B: DESCRIBE the purpose and operation of the following Essential Service Water System components:

1. Essential Service Water Pump

Question Source: Bank # __X__L16653__
Modified Bank # ______
New _______

Question History: Last NRC Exam __N/A – 2011 Audit Exam__________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Modified 2 distractors from the bank question but still a bank question.

Revised choices per NRC comment of "after" to "at"

<QQ 16653(1410)><<Given the following:

- Reactor trip and safety injection have occurred.
- All equipment is operating as required.

Which ONE of the following describes the operation of Essential Service Water (ESW) Pumps for this event?>>

A. <QQ 16653(1482)><<Train "A" ESW starts after 20 seconds, and Train "B" ESW starts approximately 5 seconds later.>>

B. <QQ 16653(1480:0)><<BOTH trains of ESW start immediately when the LOCA sequencers are actuated.>>

C. <QQ 16653(1480:1)><<BOTH trains of ESW start simultaneously after 20 seconds as programmed by the LOCA sequencers.>>

D. <QQ 16653(1480:2)><<Train "B" ESW starts after 20 seconds, and Train "A" ESW starts approximately 5 seconds later.>>

Answer: <QQ 16653 (1419) ><<A >>

Question 1 Info	
Question Type:	<QQ 16653(1401)><<Multiple Choice>>
Status:	<QQ 16653(1405)><<Active>>
Always select on test?	<QQ 16653(1406)><<No>>
Authorized for practice?	<QQ 16653(1429)><<No>>
Points:	<QQ 16653(1441)><<1.00>>
Time to Complete:	<QQ 16653(1408)><<2>>
Difficulty:	<QQ 16653(1407)><<2.00>>
System ID:	<QQ 16653(1445)><<16653>>
User-Defined ID:	<QQ 16653(1404)><<L16653>>
Cross Reference Number:	
Topic:	<QQ 16653(1400)><<L16653 EF Operation of ESW pumps w/SIS>>
Num Field 1:	<QQ 16653(1414)><<2.9>>
Num Field 2:	
Text Field:	<QQ 16653(1413)><<076 K4.02>>
Comments:	<QQ 16653(1411)><<2011 ILT Audit Exam.>>

Question 1 History	
Exam Appearances:	<QQ 16653(1449)><<1>>
Student Encounters:	<QQ 16653(1448)><<12>>
Answered Right:	<QQ 16653(1452)><<12>>
Answered Wrong:	<QQ 16653(1453)><<0>>
Partially Correct:	<QQ 16653(1459)><<0>>
Answer Invalid:	<QQ 16653(1455)><<0>>
Unanswered:	<QQ 16653(1454)><<0>>
Ignore Response:	<QQ 16653(1460)><<0>>
Avg Points Awarded:	<QQ 16653(1450)><<1.00>>
... As % of Point Value:	100
Standard Deviation:	<QQ 16653(1456)><<0.00>>

Associated objective(s):

<OB 16130(1101)><< C DESCRIBE the operation of the Essential Service Water System under the following conditions:
1. Standby
2. Safety Injection Signal
3. Loss of Offsite Power
4. Low Suction Pressure to the Auxiliary Feedwater Pumps
5. Opposite train NB bus undervoltage with low flow to the CTM T coolers>>

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
078 Instrument Air	Group #	1		
	K/A #	078 G2.4.18		
	Importance Rating	3.3		
Knowledge of the specific bases for EOPs.				

Question # 54

Per procedure E-3, Steam Generator Tube Rupture, what is the PRIMARY reason for establishing instrument air to containment?

A. To prevent overfilling of the pressurizer.

B. To allow SG Blowdown to prevent SG over pressurization.

C. To return RCP seal flow parameters to normal to prevent RCP Seal damage.

D. To restore normal PZR pressure control in order to minimize the loss of reactor coolant during RCS Depressurization.

Answer: D

Explanation:

Per BD-E-3 the purpose of establishing instrument air to containment is "To restore a sustained compressed air supply to allow control of air-operated equipment inside containment (i.e., charging and letdown valves, PZR spray valves). Later in E-3, specifically step #16 (after the RCS Cooldown is completed) the RCS will be depressurized to minimize break flow and refill the PZR. The preferred method is to use normal PZR Spray (air operated valves). Step #16a checks to see if normal PZR spray is available and if not the RNO directs step #17 which is the use of the PZR PORVs. Per the BD of E-3 step #16, "The preferred means of RCS depressurization is normal PZR spray since this does not result in a loss of reactor coolant."

The distractor of "To prevent overfilling of the pressurizer" is plausible as letdown and charging valves are air operated valves and restoring air will allow control of these valves and allow PZR level control during normal operations. Specifically, in the BD of E-3, it states "In some cases, pressurizer level may approach the upper tap (top of the indicating range) before RCS pressure is reduced to the ruptured steam generator pressure. This may be a symptom of a smaller tube failure, voiding in the upper head during natural circulation conditions, injection of the SI accumulators, or ineffectiveness of the depressurization method." ….. "Depressurization of the RCS is terminated on high pressurizer level to prevent filling the pressurizer and loss of pressurizer pressure control."

EPPV0001, SG BD N2 Supply PCV is a AOV that fails closed inside containment (Attachment G). This is plausible as SG blowdown can lower SG level and therefore pressure. If the candidate *does not understand the priorities of E-3 and believes that the method to prevent SG overpressure (due to RCS leak) into the SG is to provide a SG blowdown path*

Each RCP has a seal #1 WTR Out ISO HV that fails open on a loss of air to containment (Attachment G). Additionally, E-3 step #1 is continuous action step to secure the RCPs when certain parameters apply. Returning seal parameters to normal is plausible if the candidate does not understand the priorities and steps of E-3 and incorrectly applies the importance of reestablishing air to containment as a method to prevent from having to perform the continuous action step, Step #1

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Correct – See above explanation

Technical Reference(s):

1. BD-E-3, Basis Document for E-3, SG Tube Rupture, Rev 8
2. E-3, Steam Generator Tube Rupture, Rev 17
3. OTO-KA-00001 Attachment G, AOVs Inside Containment, Rev 23

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #14, Service and Instrument Air, Objective B: DESCRIBE the purpose and operation of the following Service and Instrument Air components: 7. Containment Instrument Air Isolation Valve (KA-FV-29)

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

EOP STEP: 16 WOG ERG STEP: 16

STEP:

DEPRESSURIZE RCS To Minimize Break Flow And Refill PZR:

PURPOSE:

To lower RCS pressure to stop primary-to-secondary leakage and establish an indicated pressurizer level.

BASIS:

After the cooldown is completed, ECCS flow will pressurize the RCS to an equilibrium condition where break flow equals ECCS flow. The equilibrium pressure will be somewhere between the ruptured steam generator pressure and the shutoff head of the ECCS pumps and rises with SI capacity, as shown in the E-3 ERG Background Document Figure 26. A major objective of the E-3 procedure is to bring the plant from point A to point B where primary-to-secondary leakage will be stopped. However the path one takes is important.

The illustrated curve in the E-3 ERG Background Document represents equilibrium conditions where ECCS flow and break flow are equal. Hence, for points on the curve, reactor coolant inventory is constant. To the left of this curve RCS pressure is greater than equilibrium so that break flow exceeds ECCS flow. Therefore, in this region coolant inventory is lowering. Conversely, to the right of the curve, ECCS flow exceeds break flow so that coolant inventory is rising. The ideal path from point A to point B should raise coolant inventory and restore pressurizer level. Hence, the ideal path (see E-3 ERG Background Document Figure 27) requires a depressurization of the RCS.

In some cases, pressurizer level may approach the upper tap (top of the indicating range) before RCS pressure is reduced to the ruptured steam generator pressure. This may be a symptom of a smaller tube failure, voiding in the upper head during natural circulation conditions, injection of the SI accumulators, or ineffectiveness of the depressurization method. In that case, the preferred path from point A to point B is demonstrated in the E-3 Background Document Figure 28. Depressurization of the RCS is terminated on high pressurizer level to prevent filling the pressurizer and loss of pressurizer pressure control. Following SI termination, pressurizer level lowers which further reduces RCS pressure to equilibrium with the ruptured steam generator. In some cases, such as a small tube failure in a high pressure SI plant, the pressurizer may be sufficiently full such that no depressurization of the RCS is necessary prior to SI termination.

(Continued on next page)

EOP STEP: 16 WOG ERG STEP: 16

On the other hand, for multiple tube failures or reduced SI capacity for a smaller tube failure, it may be necessary to lower RCS pressure below that of the ruptured steam generator pressure in order to restore pressurizer level. This path is shown in E-3 ERG Background Document Figure 29. In that case reverse flow, i.e., secondary to primary leakage, will supplement ECCS flow to restore pressurizer level. If pressure continued to be reduced to saturation, voiding in the primary system may result in an unreliable pressurizer level indication and delay SI termination. To avoid this, depressurization of the RCS is terminated if minimum RCS subcooling is reached.

With PZR spray (normal and auxiliary) stopped, both pressurizer pressure and level should rise toward equilibrium conditions. If level continues to rise without a corresponding rise in pressure, leakage from the spray valves should be suspected. If this persists until filling of the pressurizer is imminent, appropriate measures to stop the leakage, such as stopping RCPs as necessary to terminate spray flow or isolating the auxiliary spray line should be performed. It may be necessary to stop two (or more) RCPs to terminate spray flow, depending on which spray valve is failed open and the existing pressurizer level. Spray effectiveness with different combinations of RCPs running will vary with plant design as discussed in the Plant-Specific Information Section of the E-3 ERG Background Document. Depressurization of the RCS due to leakage from the spray valves will stop once the pressurizer fills with water. Therefore, this condition should not prevent or delay termination of ECCS flow in subsequent steps when all the necessary criteria are satisfied.

The preferred means of RCS depressurization is normal PZR spray since this does not result in a loss of reactor coolant. If normal spray is not available, an alternative means of depressurizing the RCS, such as a pressurizer PORV or auxiliary spray must be used. However, the use of a PORV will result in an additional loss of reactor coolant which may rupture the PRT and lead to abnormal containment conditions. On the other hand auxiliary spray may cause excessive thermal stresses in the spray nozzle and may not be sufficient to rapidly lower RCS pressure. For these reasons, it is used only if normal spray and all pressurizer PORVs are unavailable.

(Continued on next page)

EOP STEP: 16 WOG ERG STEP: 16

KNOWLEDGE:

Maximum spray flow should be established to lower primary system pressure as rapidly as possible. The operator should be familiar with how rapidly pressure will lower with full spray to avoid overshooting the termination criteria. In addition, if pressure does not lower or lowers only slowly, the operator should proceed to the next step to select an alternative means of depressurizing the RCS to expedite recovery.

Voiding in the upper head region is not expected to occur if the reactor coolant pumps are running even with full spray flow. However, if the RCS is depressurized concurrently with the cooldown some voiding may occur. In that case, pressurizer level will rise rapidly as water is displaced from the upper head into the pressurizer.

If a subsequent SGTR is diagnosed by the operator while the RCS depressurization is in progress, although it does not impact the pressure in the newest ruptured steam generator, for the sake of simplicity it should be stopped and the plant stabilized by the operator until the newest ruptured steam generator is isolated.

High PZR water level with any combination of RCPs operating will raise spray effectiveness.

DEVIATIONS:

Added plant specific RNO substep b. to enhance procedure usage and meet the ERG intent by transitioning to the next high level step if normal PZR spray is not available (or effective).

Added plant specific means for RNO substep c.1) as required by the ERG.

Added plant specific actions for RNO substep c.2) to enhance procedure usage and assist the operator in meeting the ERG intent for isolating the auxiliary spray line.

Deviated from PWROG guidance due to Simulation validation not supporting PWROG recommendations. Did not add Steps as requested in DW-04-009 and DW-10-017 which established an initial depressurization, then allowed ECCS termination. However, when the subsequent depressurization was completed to allow RCS pressure to drop below ruptured steam generator pressure, the subcooling margin of 30°F was lost. This complicates the scenario by requiring operators to transition to ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Required. This is an undesirable condition for the given conditions.

(Continued on next page)

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Containment	Group #	1		
	K/A #	103 A2.03		
	Importance Rating	3.5		
Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation				

Question # 55

Reactor Power is 100%.

- The following spurious signals occur:
    - o CSAS (Containment Spray Actuation Signal)
    - o CIS-B (Phase B)
- The crew has entered OTO-SA-00002, Spurious Containment Spray and Containment Phase B Isolation Recovery, and has completed the immediate action.

(1) What is the impact on containment temperature?

And

(2) What action should the crew perform?

A. (1) Containment temperature will remain constant since service water to the containment coolers was not lost.
(2) Perform a rapid power reduction per OTO-MA-00008.

B. (1) Containment temperature will remain constant since service water to the containment coolers was not lost.
(2) Open, under administrative controls, the CCW bypass valves.

C. (1) Containment temperature will rise due to a loss of service water to the containment coolers until the ESW system is realigned and the ESW pumps are started.
(2) Perform a rapid power reduction per OTO-MA-00008.

D. (1) Containment temperature will rise due to a loss of service water to the containment coolers until the ESW system is realigned and the ESW pumps are started.
(2) Open, under administrative controls, the CCW bypass valves.

Answer: D

Explanation: *Per OTO-SA-00002, step #2 and 3 the operator is required to restore power and open containment isolation CCW bypass valves and then establish admin controls per Attachment A. Performing a rapid down is plausible containment spray has just actuated and electrical equipment may suffer a fault, especially the RCP motors, therefore lower reactor power below a protective interlock to prevent a scram due to a RCP trip will at full power is plausible.*

With a spurious containment spray signal, valve realignment (step 0 of the sequencer occurs and ESW isolates itself from service water) resulting in a loss of cooling water to the containment coolers. Containment temperatures will rise until ESW is started in step 4 of OTO-SA-00002 " CHECK ESW Has Isolated From Service Water, THEN PERFORM The Following: …b start the ESW pumps". ESW will be the containment cooler supply water until step #20 of OTO-SA-00002 when it will be shutdown and returned to a standby lineup.

No impact is plausible if the candidate does not understand that the valve realignment (step 0 of the sequencer will occur) and falsely believes that service water is supplying the containment coolers during this event. Additionally the candidate may ONLY associate service water and ESW with the containment coolers due to the fact that a CIS-A (Phase A) will cause system actuation affecting the containment coolers (fans shifting speed and flow rises on a phase A) and ESW isolating itself from service water.

A. Incorrect – both are wrong
B. Incorrect – wrong impact
C. Incorrect – wrong action
D. Correct

Technical Reference(s):

1. OTO-SA-00002, Spurious Containment Spray and Containment Phase B Isolation Recovery, Rev 12

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, Objective D and E:
D. Given a set of plant conditions or parameters indicating a required Spurious Containment Spray and Containment Phase B Isolation Recovery, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

E. DISCUSS the cautions and notes, in OTO-SA-00002, Spurious Containment Spray and Containment Phase B Isolation Recovery.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

Question Cognitive Level:

Memory or Fundamental Knowledge _____
Comprehension or Analysis ___X__

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Revised question per NRC comments.

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION

TRIP RCPs if Component Cooling Water is lost to the RCP motor's for greater than 10 MINUTES or IF upper or lower bearing temperatures reach 195°F.

NOTE

For RCP parameters use plant computer system display BB10.

2. RESTORE Power To And OPEN Containment Isolation CCW Bypass Valves (All on RL020):

Valve	Valve Bypassed
EG HIS-126	Bypass EGHV0071
EG HIS-127	Bypass EGHV0058
EG HIS-130	Bypass EGHV0060
EG HIS-131	Bypass EGHV0059
EG HIS-132	Bypass EGHV0062
EG HIS-133	Bypass EGHV0061

a. PLACE the associated NON-ISO Switches in the NON ISO position.

b. OPEN the Containment Isolation CCW Bypass Valve.

c. PLACE the associated NON-ISO Switch in the ISO position.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. DISPATCH An Operator To The North Piping Pen Room

a. Establish administrative controls for opening the Containment Isolation CCW Bypass valves per ATTACHMENT A, Administrative Controls For Containment Bypass Valves

4. CHECK ESW Has Isolated From Service Water, THEN PERFORM The Following:

a. ENSURE UHS RETURN Valves are OPEN:

- EF HIS-37
- EF HIS-38

b. START BOTH ESW Pumps:

- EF HIS-55A
- EF HIS-56A

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Control Rod Drive	Group #	2		
	K/A #	001 A2.14		
	Importance Rating	3.7		
Ability to (a) predict the impacts of the following malfunction or operations on the CRDS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Urgent failure alarm, including rod-out-of-sequence and motion-inhibit alarms.				

Question # 56

Reactor Power is 40%.

Turbine load is being adjusted when the following occurs:

- Annunciator 65D, T REF/T AUCT HI, alarms.

Subsequently:

- Annunciator 79A, ROD CTRL URG FAIL, alarms.
- It is determined that a Pulser Failure has occurred in the Rod Control System.

(1) What is the status of the Rod Control System?

And

(2) Per OTO-SF-00001, Rod Control Malfunctions, what action should the operator take to clear Annunciator 65D?

A. (1) ALL rod motion is inhibited.
(2) Turbine load must be RAISED.

B. (1) Rod motion is ONLY inhibited in the affected bank.
(2) Turbine load must be RAISED.

C. (1) Rod motion is ONLY inhibited in the affected bank.
(2) Turbine load must be LOWERED.

D. (1) ALL rod motion is inhibited.
(2) Turbine load must be LOWERED.

Answer: D

Explanation:

Per the Annunciator 79A, A Logic Cabinet Urgent Failure prevents all rod motion in manual and automatic. A Power Cabinet Urgent Failure prevents rod motion in manual and automatic for rods in the affected power cabinet. A Pulser failure is only associated with the Logic cabinet Urgent Failure and therefore ALL rod motion is prevented in manual and automatic.

OTO-SF-00001, Continuous action step #9, directs the operator to maintain RCS Tavg/Tref deviation within 1.5F using either adjusting turbine load or RCS boron concentration. Annunciator 65D alarms when the deviation is 3F. For the situation when Tref / Tavg (spoken as Tref as compared to Tavg) is HI, turbine load (Tref) shall be lowered to remove the deviation and met the requirements of the procedure. This is opposite of when Annun 65E (Tref / Tavg) is LO when raising turbine load would clear the alarm and met the step of the procedure. However, raising turbine load is plausible if the candidate does not correctly understand the relationship between actual plant conditions and / what the annunciator is comparing (i.e. believes it is saying that Tavg is too high for the given T ref and T ref should be raised by raising turbine load).

A. Incorrect. Turbine load would have to be lowered to clear Annun 65D.
B. Incorrect. Impact on the rod control system is incorrect and action to clear Annun 65D is also incorrect.
C. Incorrect. Correct action but impact on rod control is incorrect.
D. Correct. A pulser failure generates a logic cabinet urgent failure which inhibits all rod motion. To clear Annun 65D, turbine load needs to be lowered IAW OTO-SF-00001, Rod Control Malfunctions.

Technical Reference(s):
1. OTA-RK-00022, Addendum 79A, Rod Control Urgent Failure, Rev 2
2. OTO-SF-00001, Rod Control Malfunctions, Rev 15

References to be provided to applicants during examination: None

Learning Objective:

T61.0110, LP 26, Objective M, List the causes of a "Rod Control Urgent Failure" and Explain the effects on the system.

T61.003B, LP B-45, Objective D, Given a set of plant conditions or parameters indicating a Rod Control Malfunction, Analyze the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ___X L16779___
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2013_Question #65________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(6)

Comments:

Replaced question and K/A per NRC Comment.

Reworded stem for plausibility of Distractor of raising turbine load, removed refueling from stem and reordered the order of failures in the stem.

Rod Control Urgent Failure			79A
Initiating Device: 1. See below	**Setpoint:** 1. See below	**Reset:** 1. Condition cleared	**ROD CTRL URG FAIL**

1.0 AUTOMATIC ACTIONS:

- A Logic Cabinet Urgent Failure prevents all rod motion in manual and automatic.
- A Power Cabinet Urgent Failure prevents rod motion in manual and automatic for rods in the affected power cabinet.

2.0 IMMEDIATE ACTIONS:

2.1. None

3.0 OPERATOR ACTIONS:

3.1. Go To OTO-SF-00001, Rod Control Malfunction.

3.2. REQUEST I&C department to investigate.

Logic Cabinet Urgent Failure		Power Cabinet Urgent Failure	
Initiating Device	**Setpoint**	**Initiating Device**	**Setpoint**
Pulser	Failure of Pulser	Regulation Failure	Coil Current ≠ Demand
Slave Cycler	Input Incorrect	Phase Failure	Excessive Ripple on Voltage to Coils
Circuit Card	Loose or Removed	Logic Error	Current Command Signals to Stationary and Moveable Gripper Coils Lost at the Same Time
Shutdown Bank CDE	Circuit Failure	Multiplexing Error	Lift or Moveable Multiplexing Thyristor Failed, Could Energize More Than One Group of Mechanisims at the Same Time
		Circuit Card	Loose or Removed

4.0 SUPPLEMENTAL INFORMATION:

4.1. Manual M-763-0831, Full Length Rod Control System

1 **ID: L16779** **Points: 1.00**

<QQ 16779(1410)><<Given the following plant conditions:

- Turbine load is being raised following a refueling outage
- Annun 79A, ROD CTRL URG Fail, alarms
- Annun 65D, T REF/T AUCT HI, alarms
- The Balance of Plant Operator stops turbine loading
- It has been determined that a Pulser Failure occurred.

(1) What is the impact on the Rod Control System?

And

(2) What action is required to be taken?>>

A. <QQ 16779(1482)><<(1) ALL rod motion is inhibited.
(2) Turbine load must be lowered to clear Annun 65D.>>

B. <QQ 16779(1480:0)><<(1) ALL rod motion is inhibited.
(2) Turbine load must be raised to clear Annun 65D.>>

C. <QQ 16779(1480:1)><<(1) Rod motion is ONLY inhibited in the affected bank.
(2) Turbine load must be raised to clear Annun 65D.>>

D. <QQ 16779(1480:2)><<(1) Rod motion is ONLY inhibited in the affected bank.
(2) Turbine load must be lowered to clear Annun 65D.>>

Answer: <QQ 16779 (1419) ><<A >>

Question 1 Info	
Question Type:	<QQ 16779(1401)><<Multiple Choice>>
Status:	<QQ 16779(1405)><<Active>>
Always select on test?	<QQ 16779(1406)><<No>>
Authorized for practice?	<QQ 16779(1429)><<No>>
Points:	<QQ 16779(1441)><<1.00>>
Time to Complete:	<QQ 16779(1408)><<3>>
Difficulty:	<QQ 16779(1407)><<3.00>>
System ID:	<QQ 16779(1445)><<16779>>
User-Defined ID:	<QQ 16779(1404)><<L16779>>
Cross Reference Number:	<QQ 16779(1409)><<OTA-RK-00002, ADD 79A>>
Topic:	<QQ 16779(1400)><<L16779 SF Impact of pulser failure on rod control system>>
Num Field 1:	<QQ 16779(1414)><<3.7>>
Num Field 2:	<QQ 16779(1415)><<3.9>>
Text Field:	<QQ 16779(1413)><<001 A2.14>>
Comments:	<QQ 16779(1411)><<2013 NRC Exam OTO-SF-00001>>

Question 1 History	
Exam Appearances:	<QQ 16779(1449)><<16>>
Student Encounters:	<QQ 16779(1448)><<87>>
Answered Right:	<QQ 16779(1452)><<77>>
Answered Wrong:	<QQ 16779(1453)><<10>>
Partially Correct:	<QQ 16779(1459)><<0>>
Answer Invalid:	<QQ 16779(1455)><<0>>
Unanswered:	<QQ 16779(1454)><<0>>
Ignore Response:	<QQ 16779(1460)><<0>>
Avg Points Awarded:	<QQ 16779(1450)><<0.89>>
... As % of Point Value:	89
Standard Deviation:	<QQ 16779(1456)><<0.32>>

Question 1 Table-Item Links

<TB 5114(1301)><<OPS Procedures>>

<TB 8061(1305)><<OTO-SF-00001, ROD CONTROL MALFUNCTIONS>>

<TB 5818(1301)><<OPS Systems>>

<TB 5945(1305)><<SF, Reactor/Rod Control>>

<TB 5972(1301)><<OPS Question Catagory>>

<TB 5974(1305)><<LO Initial, Closed Book>>

Associated objective(s):

<OB 16401(1101)><< M LIST the causes of a "Rod Control Urgent Failure" and EXPLAIN the effects on the system.>>

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Rod Position Indication	Group #	2		
	K/A #	014 A1.02		
	Importance Rating	3.2		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: Control rod position indication on control room panels.				

Question # 57

A Reactor startup is in progress.

- Control Bank D at 102 steps.
- Annunciator 80B, RPI Non Urgent Alarm is LIT.
- A flashing General Warning LED is observed above the DRPI indication for Control Rod D-12.
- Data B failure is indicated.

Which of the following describes the MAXIMUM possible range of actual positions for Control Rod D-12?

A. 92 – 106 steps

B. 92 – 112 steps

C. 98 – 106 steps

D. 98 – 112 steps

Answer: A

Explanation:
Per Annunciator 80B, "Depending on the cause of the alarm, the RPI System may be in half-accuracy operation. The rod position in question will be denoted by a flashing General Warning LED. From the display panel, the range of any lighted LED at half-accuracy under "Error in A" condition is +10, -4 steps. The range of any lighted LED under "Error in B" condition is -10, +4 steps."

Therefore the possible range is 102 -10 and 102 + 4 steps which is 92 – 106 steps.

The distractors are a combination of data A failure number (+10 -4 steps) which would be 98 to 112 steps and a combination of data A and B failure numbers such as both are with 4 steps (98-106) or both are within a 10 step range (92-112 steps). These are plausible if the candidate only remember one of the accuracy number (i.e. 10 or 4)

A. Correct – See above explanation
B. Incorrect – This is the combination of ±10 steps to 102 for a range of 92 – 112 steps.
C. Incorrect – This is the combination of ±4 steps to 102 for a range of 98 – 106 steps.
D. Incorrect – These are the Data A Failure number range

Technical Reference(s):

1. OTA-RK-00022 ADD 80B, RPI Non Urgent Alarm, Rev 6

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #26, rod Control, Objective S: EXPLAIN how DRPI full and half accuracy are developed.

Question Source: Bank # ___X __R8582_
Modified Bank # ______
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

k/a/ match as the operator must predict the expected range of rod positions when given a RPI and MCB alarms associated with RPI. Knowing the rod position (or expected range of rod positions) is directly related to reactivity management and required such that the fuel design limits are not exceeded.

Question Cognitive Level was determined to be higher order. While the knowledge of Data B failure is -10 to +4 is lower order, applying these values to the given rod position to determine the correct range makes this a higher order question.

Revised stem to add MAXIMUM per NRC Comment

Rod Position Indication Non-Urgent Alarm			80B
Initiating Device: 1. See below	**Setpoint:** 1. See below	**Reset:** 1. See below	**RPI NON URG ALARM**

1.0 AUTOMATIC ACTIONS:

1.1. None

2.0 IMMEDIATE ACTIONS:

2.1. None

3.0 OPERATOR ACTIONS:

NOTE:

Depending on the cause of the alarm, the RPI System may be in half-accuracy operation.

The rod position in question will be denoted by a flashing General Warning LED.

From the display panel, the range of any lighted LED at half-accuracy under "Error in A" condition is +10, -4 steps. The range of any lighted LED under "Error in B" condition is -10, +4 steps.

Due to a failed B channel DRPI coil for Control Rod H10 in Control Bank A, DRPI position for this rod will indicate with half-accuracy from 162 to 180 steps. MP 14-0005 has defeated the Data B failure alarm for the failed coil. As the rod passes through the area of the failed coil, this annunciator may alarm momentarily due to rod deviation discrepancy.

3.1. IF more than one DRPI per group is INOPERABLE, for one or more groups, PERFORM the following:

3.1.1. ENSURE SE HS-9, ROD BANK AUTO/MAN SEL is in MANUAL.

3.1.2. STOP any evolutions that would require Control Rod motion.

3.1.3. MONITOR and RECORD RCS Tavg once per hour.

3.2. REQUEST I&C Department to investigate.

3.3. Refer To T/S 3.1.7, FSAR 16.1.3.1 and 16.3.3.8.

OPERATOR ACTIONS (Cont'd):

Initiating Device	Setpoint	Reset
A or B Rod Position Data	Parity Error, Grey Code Data Error, or Accuracy Mode Switch Not in A+B Position	No Error, Accuracy Mode Switch in A+B Position
Rod Deviation Discrepancy	RPI Rod Dev Monitor Cards Disagree	No Disagreement
Rod Deviation Card	Removed	Installed
Central Control Card	Removed Disagreement	Installed No Disagreement

4.0 SUPPLEMENTAL INFORMATION:

4.1. Manual M-763-0834, Digital Rod Position Indication System

4.2. SOER 84-02, Control Rod Mispositioning

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
015 Nuclear Instrumentation	Group #	2		
	K/A #	015 K1.01		
	Importance Rating	58		
Knowledge of the physical connections and/or cause/effect relationships between the NIS and the following systems: RPS				

Question # 58

Reactor Power is 6% and a plant shut down is in progress.

Intermediate Range Nuclear Instrument channel N36 fails high.

(1) What is the effect, if any, of this failure on the reactor?

And

(2) What is the effect, if any, of this failure on the Source Range Nuclear Instrument System?

A. (1) The Reactor Trips.
(2) Source Range Nuclear Instruments are NOT affected.

B. (1) The Reactor Trips.
(2) Source Range Nuclear Instruments must be manually energized.

C. (1) The Reactor does NOT Trip.
(2) Source Range Nuclear Instruments are NOT affected.

D. (1) The Reactor does NOT Trip.
(2) Source Range Nuclear Instruments must be manually energized.

Answer: B

Explanation:
At 6% Power (between P-6 and P-10) an IR channel failing high will result in a reactor scram. The logic for this trip is 1 out of 2; per E-0 section B. This is different if power was above P-10 (~10% power) as the IR trip would be blocked.

With one IR channel failed high P-6 will not extinguish (logic is 2 out of 2 less than 5E-11 amps) and the SR detectors will not auto energize during a reactor trip or reactor shutdown that is in progress.

Overall, due to the combination of logic requirements and power levels associated with these *setpoints, the distractors of not trip along with SR will auto energize are plausible.*

A. Incorrect – the SR detectors will not automatically energize as the logic is 2 out of 2 not 1 out of 2 for IR less than 5E-11 amps.
B. Correct
C. Incorrect – both are wrong. The logic for the reactor trip is 1 out of 2 not 2 out of 2. the SR detectors will not automatically energize as the logic is 2 out of 2 not 1 out of 2 for IR less than 5E-11 amps.
D. Incorrect - The logic for the reactor trip is 1 out of 2 not 2 out of 2.

Technical Reference(s):
1. OTO-SA-00001, EFSAS Verification and Restoration, Rev 39, Attachment AQ and AR
2. E-0, Reactor Trip or Safety Injection, Rev 16
3. 7250D64 S004, Rev 8 Functional diagram for NI Permissives and Blocks

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP -28, Excore Nuclear Instrumentation, Objective J: LIST the systems that interface with the Nuclear Instrumentation System and EXPLAIN how a loss of the interfacing system or a loss of the Nuclear Instrumentation System or component affects the other.

Question Source: Bank # __X R11935____
Modified Bank # ______
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

B. SYMPTOMS OR ENTRY CONDITIONS

1) The following are symptoms that require a reactor trip, if one has not occurred:

Reactor Trip	Logic, Interlock	Setpoint
SR high flux	(1/2, P-10 and P-6)	10^5 CPS
IR high flux	(1/2, P-10)	25% of RTP
PR high flux low level	(2/4, P-10)	25% of RTP
PR high flux high level	(2/4)	109%
PR positive rate trip	(2/4) (two seconds)	+4.25%
Overtemperature ΔT	(2/4)	122.6% ±
Overpower ΔT	(2/4)	110.73% -
PZR pressure - Low	(2/4, P-7)	1885 PSIG
PZR pressure - High	(2/4)	2385 PSIG
PZR water level - High	(2/3, P-7)	92%
Rx coolant flow - Low	(2/3, 2/4 P-7, 1/4 P-8)	90% Design
RCP bus undervoltage	(1/2, 2/2 P-7)	10584 VAC
RCP bus underfrequency	(1/2, 2/2 P-7)	57.2 Hz
SG NR level - Low-Low	(2/4, 1/4) Normal {EAM}	17% {21%}
Turb trip - Low oil press	(2/3, P-9)	598.94 PSIG
Turb trip - Stop valves	(4/4, P-9)	1% OPEN
Safety Injection ESFAS	(1/4 signals)	SI
SSPS General warning	(2/2)	N/A

2) The following are symptoms of a reactor trip:

- Any reactor trip annunciator lit.
- Rapid lowering of neutron flux on nuclear instrumentation.
- All shutdown and control rods are fully inserted.
- Rod bottom lights are lit.

3) The following are symptoms that require a reactor trip and safety injection, if one has not occurred:

Reactor Trip & Safety Injection	Logic, Interlock	Setpoint
PZR pressure - Low	(2/4, P-11)	1849 PSIG
Steamline pressure - Low	(2/3 on 1/4, P-11)	615 PSIG
Containment pressure - High-1	(2/3)	3.5 PSIG

4) The following are symptoms of a reactor trip and SI:

- Any SI annunciator lit.
- ECCS pumps running.

5) This procedure should also be entered any time a manual reactor trip or safety injection is actuated.

C. CONDITIONS FOR [ADVERSE CONTAINMENT]

- Containment Radiation — HAS BEEN GREATER THAN 10^5 R/HR

  OR
- Containment Pressure — GREATER THAN 3.5 PSIG

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Containment Iodine Removal	Group #	2		
	K/A #	00027 K2.01		
	Importance Rating	3.1		
Knowledge of bus power supplies to the following: Fans				

Question # 59

What are the power supplies to the Shutdown Purge Supply and Exhaust Fans?

A. NG01 and NG02

B. NG03 and NG04

C. PG19 and PG20

D. PG20 and PG25

Answer: C

Explanation: *Per the 2 electrical prints (E-23GT01 and 10), the power supplies to the supply and exhaust fan units are PG 19 and PG20 (non safety 480VAC MCCs). The distractors are either safety or non safety related 480VAC MCCs that power some type of containment ventilation equipment. The safety related choices of NG01 through NG04 are plausible as various components associated with containment and containment ventilation (isolation dampers and valves, containment cooler fans, some CRDM fans, hydrogen mixing fans, etc) are safety related and/or are powered from safety related 480 VACs MCC. Furthermore, as there is a CPIS (containment purge isolation signal) that isolates containment shutdown purge supply and exhaust air units (reference OTO-SA-00001 Attachment AD and AE), it is plausible that someone would believe that these fans are powered from a safety powered power supply*

A. Incorrect – These are the power supplies for the containment cooler fans A & B and the hydrogen mixing fans in combination of NG01 – A &C and NG02 – B &D
B. Incorrect – These are the power supplies for the containment cooler fans C & D
C. Correct
D. Incorrect – PG25 powers the mini purge exhaust fan

Technical Reference(s):
1. OTN-GT-00001, Containment Purge System, Rev 31
2. E-23GT01, CONTAINMENT PURGE SUPPLY AIR UNITS, Rev 4
3. E-23GT10, SHUTDOWN PURGE EXHAUST FAN, Rev 4
4. OTO-SA-00001, EFSAS Verification and Restoration, Rev 39

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References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#40, Containment Ventilation, Objective M: DESCRIBE function and operation of the following containment purge system components.
1. Mini Purge Supply Air Unit
2. Shutdown Purge Supply Air Unit
3. Containment Purge Filter Absorber Unit
4. Mini Purge Exhaust Fan
5. Shutdown Purge Exhaust Fan

Question Source: Bank # __X_L16679___
Modified Bank # ______
New _______

Question History: Last NRC Exam __N/A__________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

The supply unit is actually powered from PG19N but PG19N is feed from PG19 which is the correct answer. The use of PG19N instead of PG19 made this stand out as different and since it was the correct answer, it was decided to use the main MCC.

Revised explanation per NRC Comment on plausibility of safety related MCC.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Main Turbine Generator	Group #	2		
	K/A #	045 K4.02		
	Importance Rating	2.5		
Knowledge of MT/G system design feature(s) and/or interlock(s) which provide for the following: Automatic shut of reheat stop valves as well as main control valves when tripping turbine				

Question # 60

With the plant at full power, which of the following describes the operation of the Turbine Emergency Trip System (ETS) if the Main Turbine were manually tripped from the control room?

A. ONLY the Electrical Trip System actuates to provide turbine trip function; Intermediate Stop Valves AND Intercept Valves close.

B. ONLY the Electrical Trip System actuates to provide turbine trip function; Intermediate Stop Valves close, Intercept Valves remain open.

C. BOTH the Electrical Trip System AND the Mechanical Trip System actuate to provide turbine trip function; Intermediate Stop Valves AND Intercept Valves close.

D. BOTH the Electrical Trip System AND the Mechanical Trip System actuate to provide turbine trip function; Intermediate Stop valves close, Intercept Valves remain open.

Answer: C

Explanation:

Per OTN-AC-00001, attachment 2, the master trip PB feeds both the 24 VDC (electrical trip Solenoid) and the 125 VDC (mechanical trip solenoid). Only the electrical is plausible if the candidate does not recall the master trip PB also feeds the mechanical trip system i.e. only associates the PB trip with electrical trip system.

On a turbine trip all TSV (intermediate stop valves and main stop valves) and Intercept valves close. The distractor of the intercept valves remain open is plausible as the intercept stop valve are closed and the candidate may believe it is not necessary to close the intercept valves. Furthermore the candidate may recall chest warming or resetting the turbine has a combination of ISV open and IV closed and confuse these valve closures and then apply it to a turbine trip.

A. Incorrect – both trip systems actuate
B. Incorrect – both are wrong
C. Correct – see above explanation
D. Incorrect – the intercept valves close

Technical Reference(s):

1. OTN-AC-00001, Main Turbine and Generator Systems, Rev 50, Attachment 2
2. OTO-AC-00001, Turbine Trip Below P-9, Rev 3,

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #31, Main Turbine and Auxiliaries, Objective A: STATE the function and EXPLAIN the design criteria for the following systems:

1. Main Turbine

Question Source: Bank # _X_L16687 ____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

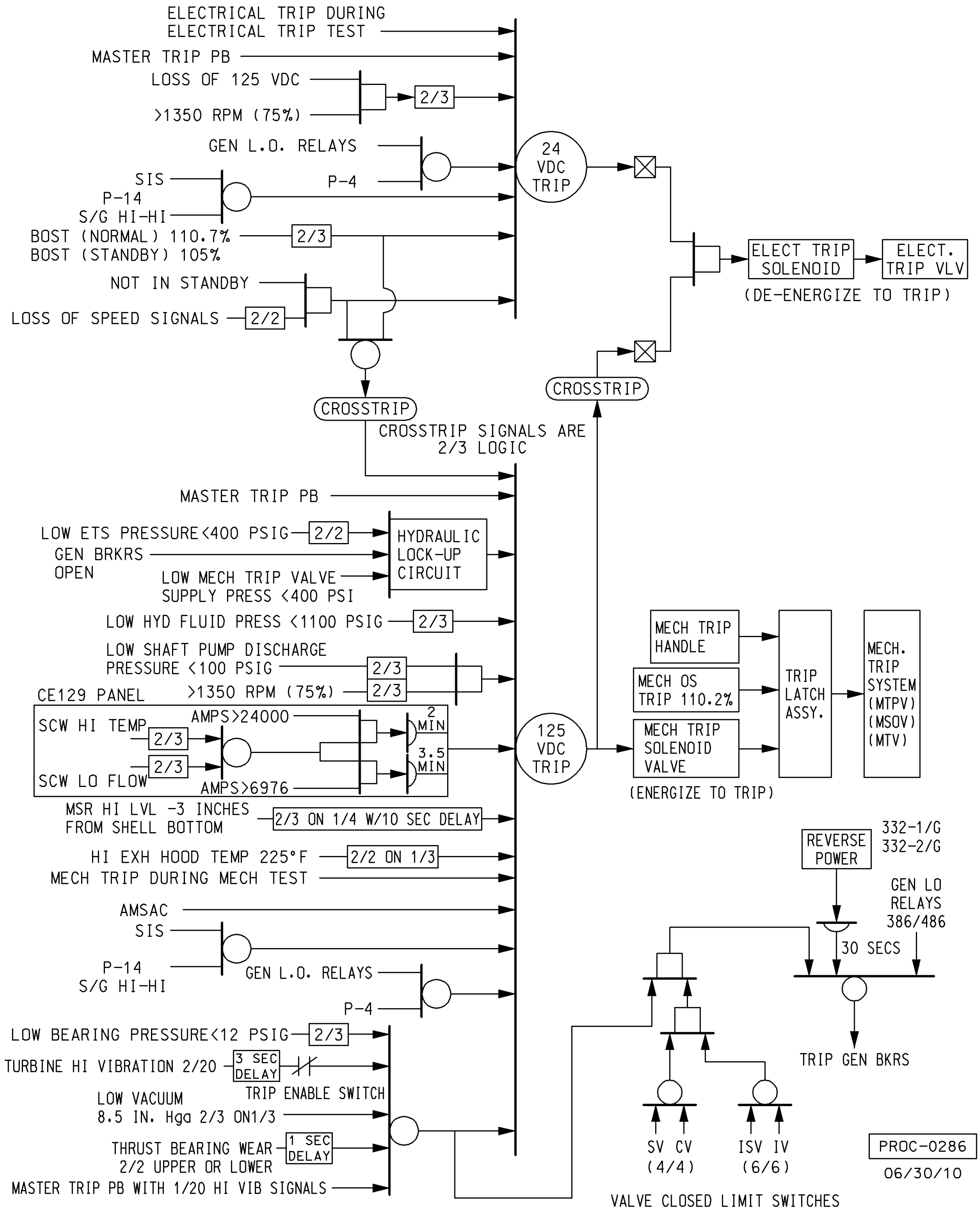
Comments:

Intermediate stop valves closing are included in each answer and does not make this a 3 part question. This was provided so the candidate understands what specific valves the question is referring to i.e intercept valves vice intercept stop valves.

Attachment 2

Turbine Trips

Sheet 1 of 1



CONTINUOUS USE

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Condenser Air Removal	Group #	2		
	K/A #	055 G2.1.28		
	Importance Rating	4.1		
Knowledge of the purpose and function of major system components and controls.				

Question # 61

What is the setpoint at which the standby condenser vacuum pump will automatically start?

A. 4.0 inches HgA

B. 5.6 inches HgA

C. 6.5 inches HgA

D. 8.5 inches HgA

Answer: B

Explanation:
Per the OTO-AD-00001, section B, the standby Condenser Air Removal Pump starts. The distractors are alarms, required component actuation values, or systems actuation due to condenser backpressure included in the OTO.

A. Incorrect – this is the value at which the operators are required to ENABLE the turbine setback on Circ Pump Lockout per OTN-DA-00001, step 5.10.2. This is also a backpressure operating limit when below 30% per OTG-ZZ-00004, Power Operations
B. Correct – this is the setpoint at which the standby Condenser Air Removal Pump starts.
C. Incorrect – this is the setpoint for the condenser vacuum Low Alarm which comes in after the auto start of the standby vacuum pump
D. Incorrect – this is the setting for the automatic main turbine trip

Technical Reference(s):
1. OTO-AD-00001, Loss of Condenser Vacuum, Rev 32
2. OTN-DA-00001, Circulating Water System, Rev 35

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #22, Objective C:DESCRIBE the function and operation of the following Condensate System components:

9. Condenser Air Removal Subsystem
   a. Condenser Vacuum Pumps

Question Source: Bank # __X__L4878__
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

All 3 of the bank question's distractor were modified to include setpoints of other equipment's actuation. The question was still considered a bank question as the stem and correct answer were unchanged.

k/a match as the question ask the auto start setpoint (function) of the standby condenser vacuum pump (component)

A. PURPOSE

1) This procedure provides instructions for the operator response to lowering or reduced Main Condenser vacuum (rising Main Condenser backpressure).

2) This procedure provides two levels of response:

   - Diagnostic process to identify and correct the cause of the loss of vacuum.

   - Monitoring condenser backpressure and reducing Main Turbine load if required.

B. SYMPTOMS OR ENTRY CONDITIONS

1) Automatic starting of the standby Condenser Vacuum pumps.

2) Rising Main Condenser backpressure.

3) Unexplained lowering in Main Turbine load.

4) Low Pressure Turbine Exhaust Hood Temperature rising.

Condenser Backpressure (in HgA)	Automatic Actuation
5.6	Standby Condenser Air Removal Pump starts
6.0	C-9, Steam Dump Block Permissive
6.5	Condenser Vacuum Low Alarm
8.5	Main Turbine Trip
15.6	Main Feed Pumps Trip

5) Any of the following Control Room annunciators in alarm:

   - Annunciator 116B, Cond A Vac Lo
   - Annunciator 117B, Cond B Vac Lo
   - Annunciator 118B, Cond C Vac Lo

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Steam Generator	Group #	2		
	K/A #	035 K6.01		
	Importance Rating	3.2		
Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: MSIVs				

Question # 62

Reactor Power is 25% with the main generator synchronized to the grid.

'A' MSIV, ABHV14, fails instantaneously CLOSED for unknown reasons.

'A' SG Pressure rises to 1200 psig.

Per Technical Specifications, what is the MAXIMUM number of Main Steam Safety Valves (MSSVs) that should be OPEN? (Ignore any lift setpoint drift.)

A. 1

B. 2

C. 3

D. 4

Answer: B

Explanation:

Per Technical Specification Table 3.7.1.2, 2 MSSVs should be open because their lift settings are 1185 psig and 1197 psig. The distractor of 1,3, and 4 MSSVs are plausible if the candidate does not correctly remember the lift setpoints in Technical Specifications Table 3.7.1.2.

A. Incorrect – see above explanation. Plausible if the candidate does not correctly remember or apply the 2nd lift setting of 1197 psig.
B. Correct – see above explanation.
C. Incorrect – see above explanation. Plausible if the candidate does not correctly remember or apply the 3rd lift setting of 1210 psig.
D. Incorrect – see above explanation. Plausible if the candidate does not correctly remember or apply the 3rd and 4th lift setting of 1210 psig and 1222 psig respectively.

Technical Reference(s):

1. Technical Specifications 3.7.1, Main Steam Safety Valves

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References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #20 Objective B: DESCRIBE the purpose and operation of the following Main Steam System components:

1. Main Steam Line (MSL) Safety and Atmospheric Steam Dumps (ASDs), including setpoints
2. Main Steam Isolation and Bypass Valves
3. MSL Low Point Drains
4. Steam Dumps and Isolation Valves
5. Turbine Driven Auxiliary Feedwater (TDAFW) Pump Steam Supply Valves
6. Steam Generator
   a. Feed ring with J-tubes
   b. Swirl vane separator
   c. Chevron separators
   d. Steam flow restrictor nozzle

Question Source: Bank # ______
Modified Bank # ______
New ___X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Question and k/a replaced per NRC Comments.

k/a match as the MSSVs are apart of the Main Steam system and the failure of A MSIV in the close position which results in a pressure rise (cause) will results in some of the MSSVs to open (effect).

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Steam Dump/Turbine Bypass Control	Group #	2		
	K/A #	041 A4.05		
	Importance Rating	3.1		
Ability to manually operate and/or monitor in the control room: Main steam header pressure				

Question # 63

During a cooldown the Control Room Supervisor directs the Reactor Operator to place the condenser steam dump cooldown valves in automatic to maintain RCS Tavg at 500°F.

The Reactor Operator should adjust AB-PK-507, Steam Header Pressure CTRL, to what value to accomplish this?

A. 3.33

B. 4.44

C. 4.54

D. 4.63

Answer: B

Explanation:

Per the OOA, ABPK507 is a 0-1500 psig controller with 150 psig per Turn making it a 10 turn controller.

First the saturated pressure at 500F must be found using steam tables. Per Table 1 of the reference provided to the students, 500F has a saturation pressure of 680.53 psia. To convert to psig subtract 14.7psi which equals 665.83 psig. To determine the setting on the controller, this value must be divided by the psig per turn i.e. 150 psig/ turn. 665.83 psig / (150 psig / turn) = 4.44 turns

For the distractors, if the candidate does not convert from absolute at all and uses 680.53 psia / (150 psig / turn), the setting on the controller would be 4.54 turns

If the conversion to absolute is done improperly by adding 14.7psi to the saturation pressure (instead of correctly subtracting) the resulting pressure would be 680.53 psia + 14.7 psi = 695.23 psi and then dividing by 150psi/ turn = 4.63 turns

If the candidate has a misconception that there is no need to convert to pressure or believes the controller is 150F per turn not 150 psig per turn and uses the saturation temperature of 500F /

(150 / turn) a setiing of 3.33 would be calculated.

A. Incorrect – using 500F without conversion to pressure
B. Correct – see above explanation
C. Incorrect – No conversion from absolute pressure
D. Incorrect – Incorrectly converted from absolute to guage

Technical Reference(s):

1. OOA-RL-00004, MCB Controllers and Potentiometers, Rev 13

References to be provided to applicants during examination:

1. ASME Steam Tables Compact Edition, Volume 83, 2006 edition

Learning Objective: T61.0110, Systems, LP #20, Main Steam, Objective D: IDENTIFY all Main Steam, Steam Dump and S/G controls, alarms and indications and DESCRIBE how each is used to predict, monitor or control the Main Steam, Steam Dump and S/G System.

Question Source: Bank # __X R8373____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Changed one bank question's distractor from 3.46 to 4.54

3.0 **RL005/6**

AB PK-507	STEAM HDR PRESS CTRL	7.28 TURNS (1092 psig)
AB PI-507	0-1500 psig	150 psig/TURN
NOTES:		

AB PIC-1A	SG A STEAM DUMP TO ATMS CTRL	1125 psig
N/A	0-1500 psig	N/A
NOTES:		

AB PIC-2A	SG B STEAM DUMP TO ATMS CTRL	1125 psig
N/A	0-1500 psig	N/A
NOTES:		

AB PIC-3A	SG C STEAM DUMP TO ATMS CTRL	1125 psig
N/A	0-1500 psig	N/A
NOTES:		

AB PIC-4A	SG D STEAM DUMP TO ATMS CTRL	1125 psig
N/A	0-1500 psig	N/A
NOTES:		

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Fire Protection	**Group #**	2		
	K/A #	086 A3.01		
	Importance Rating	2.9		
Ability to monitor automatic operation of the Fire protection System including: Starting mechanisms of fire water pumps				

Question # 64

A Fire Protection deluge valve actuated and caused a system pressure reduction to 122 psig and an associated alarm on Panel KC008.

What is the system response?

A. ONLY the Electric Fire Pump is running

B. ONLY the Electric Fire Pump and “A” Diesel Fire Pump are running

C. ONLY the Electric Fire Pump and “B” Diesel Fire Pump are running

D. The Electric Fire Pump, “A” Diesel Fire Pump, and “B” Diesel Fire Pump are running

Answer: B

Explanation:

Per OTN-KC-00001, section 4.1 the electric fire pumps starts at 130 psig, the “A” Diesel fire pump starts at 125 psig and the “B” Diesel fire pump starts at 120 psig. Therefore only the electric fire pump and A Diesel Fire Pump will be running.

A. Incorrect – see above explanation. Plausible if the candidate does not remember the setpoints for each fire pump auto starting.
B. Correct - see above explanation
C. Incorrect - see above explanation. Plausibe if the candidate does not remember the setpoints for each fire pump auto starting or confuses the setpoints of the A and B Diesel Fire pumps.
D. Incorrect - see above explanation. Plausible if the candidate does not remember the setpoints for each fire pump auto starting as there has been a fire protection deludge value actuation.

Technical Reference(s):

1. OTN-KC-00001, Fire Protection System, Rev 22
2. OTN-KC-01003, Simplex Fire Protection System, Rev 6
3. APA-ZZ-00703, Fire Protection Operability criteria and surveillance Requirements, Rev 24

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #35, Fire Protection, Objective B:
DESCRIBE the purpose, location and operation of the following Fire Protection System components:

2. Water Protection System

a. Fire Water Makeup Pump
b. Fire Water Storage Tanks
c. Freeze Protection Circulating Pumps and Heaters
d. Jockey Pump
e. Air Compressor
f. Accumulator
g. Electric Fire Pump
h. Diesel Fire Pumps
i. Fire Pump House Support Systems

Question Source: Bank # ____X R12076______
Modified Bank # ______
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Replaced question and K/A per NRC Comment.

Although randomly selected while maintain k/a spread and values >or = to 2.5, the same K/A appears on the Audit exam but the stem's pressure value and correct answer have been verified to be different when compared to the audit exam.

Audit exam question #65 provided for review along with bank question

4.0 PROCEDURE INSTRUCTIONS

4.1. General System Information

4.1.1. The Electric Fire Pump (PKC1001A) will start on any of the following:

- System pressure decreases to less than or equal to 130 psig.
- Remote start signal from KC008.
- Local start signal from CPKC1001.
- Local emergency start signal from CPKC1001.

4.1.2. The "A" Diesel Fire Pump (PKC1002A) will start on any of the following:

- System pressure decreases to less than or equal to 125 psig.
- Loss of power to the "A" Diesel Fire pump battery charger AND loss of power to the Electric Fire Pump.
- Remote start signal from KC008.
- Local start signal using the Manual Crank Pushbuttons.
- Local emergency start signal using the Manual Start Levers.

4.1.3. The "B" Diesel Fire Pump (PKC1002B) will start on any of the following:

- System pressure decreases to less than or equal to 120 psig.
- Loss of power to the "B" Diesel Fire Pump battery charger.
- "A" Diesel Fire Pump fails to start.
- Electric Fire Pump fails to start.
- Remote start signal from KC008.
- Local start signal using the Manual Crank Pushbuttons.
- Local emergency start signal using the Manual Start Levers.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Fire Protection	Group #	2		
	K/A #	086 A3.01		
	Importance Rating	2.9		
Ability to monitor automatic operation of the Fire protection System including: Starting mechanisms of fire water pumps				

Question # 65

A Fire Protection deluge valve actuated and caused a system pressure reduction to 126 psig and an associated alarm on Panel KC008.

What is the system response?

A. ONLY the Electric Fire Pump is running

B. Both the Electric Fire Pump and "A" Diesel Fire Pump are running

C. Both the Electric Fire Pump and "B" Diesel Fire Pump are running

D. The Electric Fire Pump, "A" Diesel Fire Pump, and "B" Diesel Fire Pump are running

Answer: A

Explanation:
Per OTN-KC-00001, section 4.1 the electric fire pumps starts at 130 psig, the "A" Diesel fire pump starts at 125 psig and the "B" Diesel fire pump starts at 120 psig. Therefore only the electric fire pump will be running.

A. Correct
B. Incorrect
C. Incorrect
D. Incorrect

Technical Reference(s):.
1. OTN-KC-00001, Fire Protection System, Rev 22
2. OTN-KC-01003, Simplex Fire Protection System, Rev 6
3. APA-ZZ-00703, Fire Protection Operability criteria and surveillance Requirements, Rev 24

References to be provided to applicants during examination: None

Audit Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.0110 Systems, LP #35, Fire Protection, Objective B:
DESCRIBE the purpose, location and operation of the following Fire Protection System components:

2. Water Protection System

a. Fire Water Makeup Pump
b. Fire Water Storage Tanks
c. Freeze Protection Circulating Pumps and Heaters
d. Jockey Pump
e. Air Compressor
f. Accumulator
g. Electric Fire Pump
h. Diesel Fire Pumps
i. Fire Pump House Support Systems

Question Source: Bank # ______
Modified Bank # __X R12076____
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41.7

Comments:

1	ID: R12076	Points: 1.00

<QQ 12076(1410)><<A Fire Protection deluge valve actuated and caused a system pressure reduction to 124 psig and an associated alarm on Panel KC008.

Which of the following describes the system response?>>

A. <QQ 12076(1482)><<The Electric and 'A' Diesel Driven pumps are running.>>

B. <QQ 12076(1480:0)><<Only the Electric pump is running.>>

C. <QQ 12076(1480:1)><<The 'A' and 'B' Diesel Driven pumps are running.>>

D. <QQ 12076(1480:2)><<All three fire pumps are running.>>

Answer: <QQ 12076 (1419) ><<A >>

Question 1 Info	
Question Type:	<QQ 12076(1401)><<Multiple Choice>>
Status:	<QQ 12076(1405)><<Active>>
Always select on test?	<QQ 12076(1406)><<No>>
Authorized for practice?	<QQ 12076(1429)><<No>>
Points:	<QQ 12076(1441)><<1.00>>
Time to Complete:	<QQ 12076(1408)><<3>>
Difficulty:	<QQ 12076(1407)><<3.00>>
System ID:	<QQ 12076(1445)><<12076>>
User-Defined ID:	<QQ 12076(1404)><<R12076>>
Cross Reference Number:	<QQ 12076(1409)><<OTN-KC-00001>>
Topic:	<QQ 12076(1400)><<R12076 KC Fire Pump starting pressures.>>
Num Field 1:	<QQ 12076(1414)><<3.3>>
Num Field 2:	<QQ 12076(1415)><<3.3>>
Text Field:	<QQ 12076(1413)><<086A4.01>>
Comments:	<QQ 12076(1411)><<2003 Biennial Exam Revised 11/01/06>>

Question 1 History	
Exam Appearances:	<QQ 12076(1449)><<7>>
Student Encounters:	<QQ 12076(1448)><<53>>
Answered Right:	<QQ 12076(1452)><<52>>
Answered Wrong:	<QQ 12076(1453)><<1>>
Partially Correct:	<QQ 12076(1459)><<0>>
Answer Invalid:	<QQ 12076(1455)><<0>>
Unanswered:	<QQ 12076(1454)><<0>>
Ignore Response:	<QQ 12076(1460)><<0>>
Avg Points Awarded:	<QQ 12076(1450)><<0.98>>
... As % of Point Value:	98
Standard Deviation:	<QQ 12076(1456)><<0.14>>

Question 1 Table-Item Links

<TB 5818(1301)><<OPS Systems>>

<TB 5900(1305)><<KC, Fire Protection>>

<TB 5972(1301)><<OPS Question Catagory>>

<TB 5975(1305)><<LO Requalification, Open Book>>

Associated objective(s):

<OB 16337(1101)><< E LIST the automatic system operation signals including pump starting, s pray actuation, trouble alarms and fire alarms.>>

<OB 27882(1101)><<C. Operate the Fire Detection Panel KC008>>

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Reactor Coolant	Group #	2		
	K/A #	002 K5.07		
	Importance Rating	3.3		
Knowledge of the operational implications of the following concepts as they apply to the RCS: Reactivity effects of RCS boron, pressure and temperature				

Question # 65

The Plant is in MODE 2 and the reactor is critical.

At 0800, a secondary plant failure occurs, resulting in the following parameters:

- RCS Tavg 548ºF and stable
- CVCS LETDOWN ISOLATED
- PZR LEVEL 15% and stable

What is/are the MINIMUM required action(s)?

A. Restore Tavg to $\geq$ 551ºF by 0830.
OR
Be in MODE 2 with Keff < 1.0 by 0830.

B. Restore Tavg to $\geq$ 551ºF by 0830.
AND
Be in MODE 2 with Keff < 1.0 by 0830.

C. Restore Tavg to $\geq$ 551ºF by 0900.
OR
Be in MODE 2 with Keff < 1.0 by 0900.

D. Restore Tavg to $\geq$ 551ºF by 0900.
AND
Be in MODE 2 with Keff < 1.0 by 0900.

Answer: A

Explanation: *LCO 3.4.2 Condition A (TAVG in one or more operating RCS Loops not within limit) is a 30 minute completion time. The required action is to be in MODE 2 with Keff < 1.0. The other method to restore compliance is to raise TAVG to greater than or equal to 551ºF to exit the* *mode of applicability. The distractors are from other reactivity related Technical Specification such as rod alignment groups in 3.1.4 which is 1 hour. Additionally T.S. 3.1.8 which is in Mode 2 during physics testing has a combination of 15 minutes and 1 hour completion time. The requirement to perform both is also used as a distractor. All of these are less than 1 hour completion times and are therefore RO knowledge.*

A. Correct – T.S 3.4.2 A is a 30 minute requirement and therefore the completion time is 0830 for one of the 2 actions.
B. Incorrect – both of the action are NOT required.
C. Incorrect – Plausible if the candidate falsely believes that the completion time in 1 hour vice 30 minutes.
D. Incorrect – Plausible if the candidate falsely believes that the completion time in 1 hour vice 30 minutes and that both actions are required per technical specification

Technical Reference(s):

1. Technical Specification 3.4.2, RCS Minimum Temperature for Criticality, Amendment No 202

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #9, RCS, Objective K: .STATE the LCOs for the RCS minimum temperature for Criticality Technical Specifications and IDENTIFY the RCS instruments that these Technical Specifications are based on.

Question Source: Bank # ______
Modified Bank # __ X R8317____
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(6)

Comments:

k/a/ match as the candidate must have knowledge of the RCS Temperature limits and the actions required (i.e. operation implications) if these limits are not met. Furthermore determining the time by which a required action is due is an operational implication. The reason for these temperature limits and their short completion times are due to the reactivity effects of a colder than analyzed RCS when the reactor is critical.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each operating RCS loop average temperature (T_{avg}) shall be ≥ 551°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more operating RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each operating loop ≥ 551°F.	In accordance with the Surveillance Frequency Control Program

<QQ 8317(1410)><<The Callaway Plant is in MODE 2 at 10E-8 AMPS following a Reactor Startup.

A Steam Generator Atmospheric Steam Dump fails OPEN, resulting in the following parameters:

RCS T AVG	548 DEG F STABLE
CVCS LETDOWN	ISOLATED
PZR LEVEL	15% STABLE

15 minutes later, these parameter values STILL EXIST.

Which ONE of the following actions should be taken?>>

A. <QQ 8317(1482)><<Restore T AVG to ≥ 551 DEG F or be in MODE 2 with Keff < 1.0 within the next 15 minutes.>>

B. <QQ 8317(1480:0)><<Restore T AVG to ≥ 551 DEG F within 30 minutes or be in MODE 2 with Keff < 1.0 within the next 15 minutes.>>

C. <QQ 8317(1480:1)><<Restore T AVG to ≥ 551 DEG F within 15 minutes or be in MODE 2 with Keff < 1.0 within the next 30 minutes.>>

D. <QQ 8317(1480:2)><<Restore T AVG to ≥ 551 DEG F or be in MODE 2 with Keff < 1.0 within the next 30 minutes.>>

Answer: <QQ 8317(1419)><<A>>

Question 1 Info	
Question Type:	<QQ 8317(1401)><<Multiple Choice>>
Status:	<QQ 8317(1405)><<Active>>
Always select on test?	<QQ 8317(1406)><<No>>
Authorized for practice?	<QQ 8317(1429)><<No>>
Points:	<QQ 8317(1441)><<1.00>>
Time to Complete:	<QQ 8317(1408)><<4>>
Difficulty:	<QQ 8317(1407)><<4.00>>
System ID:	<QQ 8317(1445)><<8317>>
User-Defined ID:	<QQ 8317(1404)><<R8317>>
Cross Reference Number:	
Topic:	<QQ 8317(1400)><<R8317 BB Min temp for criticality>>
Num Field 1:	<QQ 8317(1414)><<3.3>>
Num Field 2:	<QQ 8317(1415)><<3.6>>
Text Field:	<QQ 8317(1413)><<002K5.07>>
Comments:	<QQ 8317(1411)><<TS 3.4.2 INPO OE 5239 LOCT 05-2 FUND 2005 BIENNIAL EXAM - MODIFIED>>

Question 1 History	
Exam Appearances:	<QQ 8317(1449)><<1>>
Student Encounters:	<QQ 8317(1448)><<8>>
Answered Right:	<QQ 8317(1452)><<5>>
Answered Wrong:	<QQ 8317(1453)><<3>>
Partially Correct:	<QQ 8317(1459)><<0>>
Answer Invalid:	<QQ 8317(1455)><<0>>
Unanswered:	<QQ 8317(1454)><<0>>
Ignore Response:	<QQ 8317(1460)><<0>>
Avg Points Awarded:	<QQ 8317(1450)><<0.63>>
... As % of Point Value:	63
Standard Deviation:	<QQ 8317(1456)><<0.52>>

Question 1 Table-Item Links

<TB 5114(1301)><<OPS Procedures>>

<TB 5811(1305)><<Technical Specifications>>

<TB 5818(1301)><<OPS Systems>>

<TB 5832(1305)><<BB, Reactor Coolant>>

<TB 5972(1301)><<OPS Question Catagory>>

<TB 5975(1305)><<LO Requalification, Open Book>>

Associated objective(s):

<OB 16202(1101)><< K STATE the LCOs for the RCS minimum temperature for Criticality Technic al Specifications and IDENTIFY the RCS instruments that these Technica l Specifications are based on.>>

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Conduct of Operations	Group #	N/A		
	K/A #	G2.1.13		
	Importance Rating	2.5		
Knowledge of facility requirements for controlling vital / controlled access.				

Question # 66

Per ODP-ZZ-00001, Operations Department – Code of Conduct, who would grant permission for a system engineer to enter the "AT THE CONTROLS AREA"? (No declared Emergency exists.)

A. Shift Manager

B. Reactor Operator

C. Shift Technical Advisor

D. Work Control Center Supervisor

Answer: A

Explanation:

Per ODP-ZZ-00001, Section 4.3.4 step c states that " The On-Shift/On-Coming Operations Crew and Resident NRC Inspectors may enter the "AT THE CONTROLS AREA" without obtaining permission from the SM/CRS. Additional permission is required to enter the "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA". Radiation Protection Technicians, Radiation Protection Tech Support, RTOs, Chemistry Technicians and I&C Technicians are considered to be part of the On-Shift Operations Crew." Therefore a system Engineer is not apart of the Operations crew and is required to obtain permission per step 4.3.4.e.

Step 4.3.4.e states "All other personnel needing access to the "AT THE CONTROLS AREA" must obtain permission from the SM or CRS. This access is on an as needed basis only. Additional permission is required to enter the "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA".

A. Correct – See explanation above
B. Incorrect – plausible as the person grants permission to the "IMMEDIATELY ADJACENT TO THE CONTROL ROOM PANELS AREA" per step 4.3.4.f
C. Incorrect. Plausible as this person is apart of the operations crew and performs administrative duties plus assists the CRS and SM in managing plant operations as directed.
D. Incorrect – plausible as the work control center supervisor authorizes work, surveillances, processes clearance (tagout / hold offs) and performs other administrative tasks during the shift.

Furthermore, the candidate may believe that a person is required to get approval prior to entering the control room access door.

Technical Reference(s):

1. ODP-ZZ-00001, Operations Department – Code of Conduct, Rev 97, section 4.3.4

References to be provided to applicants during examination: None

Learning Objective: None specific for section 4.3.4 of procedure. ODP-ZZ-00001 is covered per T61.0110, Systems, LP #66

Question Source: Bank # ______
Modified Bank # __________
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Replaced question and k/a per NRC Comment

4.3.4. Control Room Access
[Ref: 5.2.38]

NOTE

Transient combustibles are limited within the Control Room in accordance with APA-ZZ-00741, Control Of Combustible Materials.

a. Department Heads, Nuclear Function Managers, Nuclear Oversight Operations personnel, Security department personnel, I&C department personnel, Radiation Protection Technicians, Radiation Protection Tech Support, RTOs, Chemistry Technicians, On-Shift Emergency Response personnel, and other personnel authorized specifically by the Manager, Operations - Shift may enter the Control Room without obtaining permission from the SM/CRS. Additional permission is required to enter the "AT THE CONTROLS AREA" and "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA". (Attachment 1)

b. All other personnel needing access to the Control Room enter the missile door into the Control Room foyer and wait outside the "AT THE CONTROLS AREA" to be acknowledged by the SM/CRS. This permission is only for the Control Room. Additional permission is required to enter the "AT THE CONTROLS AREA" and "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA". (Attachment 1)

c. The On-Shift/On-Coming Operations Crew and Resident NRC Inspectors may enter the "AT THE CONTROLS AREA" without obtaining permission from the SM/CRS. Additional permission is required to enter the "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA". Radiation Protection Technicians, Radiation Protection Tech Support, RTOs, Chemistry Technicians and I&C Technicians are considered to be part of the On-Shift Operations Crew.

d. During a declared emergency, On-Shift Emergency Response personnel are granted permission to enter the "AT THE CONTROLS AREA".

e. All other personnel needing access to the "AT THE CONTROLS AREA" must obtain permission from the SM or CRS. This access is on an as needed basis only. Additional permission is required to enter the "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA".

Step 4.3.4 Cont'd

f. Except for the on-shift Control Room watchstanders, all personnel who need to enter the "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA" must obtain permission from the Unit Reactor Operator (URO), Balance of Plant Operator (BOP), or the CRS. The "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA" is delineated on Attachment 1, Control Room.

g. The SM/CRS has the authority and responsibility to clear the Control Room of all nonessential personnel anytime it is deemed necessary.

h. In an emergency situation, access to the Control Room is limited by the SM to the Operating Shift Complement; Emergency Coordinator, Director, Nuclear Operations, one NRC Representative, and additional management or support personnel deemed necessary to effectively handle the situation.

4.3.5. Logkeeping

a. For Logkeeping requirements, Refer To ODP-ZZ-00006, Operations Department Narrative Logs.

4.3.6. Operator Rounds

a. For guidance on Operator Rounds, Refer To ODP-ZZ-00016, Reactor Operator Watchstation Practices And Logs, ODP-ZZ-0016E, Operations Technician Watchstation Practices And Rounds, and ODP-ZZ-00020, Instrumentation Channel Deviations.

4.3.7. Operation of Equipment

a. Only personnel specifically authorized by Operations department administrative procedures may operate Control Room equipment. Control Room supervisors monitor and coach operators who perform control board manipulations, and they maintain oversight of integrated plant operations.

- Exceptions of 4.3.7.a include:
  - Routine maintenance may be performed on RM11 or SP010 by Count Room Technicians or I&C Technicians.
  - RM11 System Engineer may perform routine maintenance on RM11.
  - During emergency situations the RM11 may be operated by the Dose Assessment Technician or Engineer for trends/logtaking.

b. Appropriate human performance tools are to be used while manipulating plant controls and components.

c. Reinforce the expectation for deliberately performing manipulations, and avoiding time pressure traps and haste.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Conduct of Operations	**Group #**	N/A		
	K/A #	G 2.1.36		
	Importance Rating	3.0		
Knowledge of procedures and limitations involved in core alterations.				

Question # 67

Core loading is in progress.

Core Alterations may CONTINUE if ….?

A. Containment Purge system is not in service and the equipment hatch is closed and held in place with 2 bolts.

B. Reactor vessel boron concentration has lowered from 2200 ppm to 2190 ppm over the last 12 hours.

C. The operable RHR pump will be secured for 2 hours this shift and the other train of RHR is inoperable.

D. All source range nuclear instrument readings increased from 5 cps to 15 cps after the addition of a single fuel assembly to the core.

Answer: B

Explanation: *OTG-ZZ-00007, step 5.9.9 directs loading fuel into the reactor from the spent fuel pool per ETP-ZZ-00035, Refueling Performance (IPTE).*

Per OTG-ZZ-00007, step 5.12.4 states "ENSURE two RHR trains are operable with one train in operation during MODE 6 with less than 23 feet of water above the flange." But with core alterations in progress, RPV level will be more than 23 ft above the flange. The required RHR train may be secured for up to one hour per eight hour period as long as no operations that would results in a reduction in RCS boron concentration occur. This is stated in T.S 3.9.5 and OTG-ZZ-00007 notes.

T.S, 3.9.4, Containment Penetrations is the penetrations that are required to allow core alterations. Per the LCO, the equipment hatch if closed is required to be held closed by 4 or more bolts.

ETP-ZZ-00035 section 5 precautions and limitations, step 5.8 provides direction for when to secure core alterations.

"IF any of the following conditions occur during core loading, SUSPEND core alterations pending the evaluation by the SRO in charge of core alterations and the Reactor Engineer:

- *An increase in count rate by a factor of five occurs on any of the responding nuclear monitoring channels after the addition of any single fuel assembly, except initial load of the sources.*
- *An increase in count rate by a factor of two occurs simultaneously on all responding nuclear monitoring channels after the addition of any single fuel assembly, except initial load of the sources.*
- *Reactor vessel water boron concentration changes by 20 ppm or more from the previous value.*
- *The evacuation horn, coupled to one of the plant source range channels is activated other than for an announced test.*
- *Less than two of the responding nuclear channels are in service when both sources are loaded.*
- *Less than two of the responding nuclear channels have counting rates greater than or equal to 0.5 counts per second (after both source bearing assemblies are loaded in the vessel).*
- *A fuel assembly is damaged.*
- *Loss of communications. [Ref: 9.2.8]*
- *Any other condition occurs which the SRO or Reactor Engineer feels warrants suspending core loading.*

A. Incorrect – T.S 3.9.4 requires 4 bolts for the equipment hatch when it is held closed.
B. Correct – Boron Concentration lowered by 10 ppm (i.e. less than 20 ppm) and per the highlighted reason above, core alterations can continue. These values of ppm were chosen such that the final was still greater than T.S. 3.7.16 SFP concentration value of 2165 ppm.
C. Incorrect – the running RHR pump may be secured for 1 hour per 8 hour period as long as RPV level is high and no dilution activities are in progress. (T.S 3.9.5 Note allows the RHR pump to be secured less than or equal to one hour per 8 hours.)
D. Incorrect – all SR NI have increased by more than a factor of 2 and core alteration must be secured per ETP-ZZ-00035 (see highlighted step above)

Technical Reference(s):
1. ETP-ZZ-00035, Refueling Performance (IPTE), Rev 39
2. Technical Specification 3.9.5, RHR and Coolant Circulation - High Water Level
3. Technical Specification 3.9.4, Containment Penetrations
4. OSP-SF-00003, Pre-Core Alteration Verifications, Rev 28, section 6.3

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off normal Operations, LP #60, Objective A &B:

A. STATE the purpose and scope of OTG-ZZ-00007, Refueling Preparation, Performance and Recovery.

B. STATE the Precautions and Limitations for OTG-ZZ-00007 Refueling Preparation, Performance, and Recovery.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question History: Last NRC Exam ___N/A_________

Question Cognitive Level:

Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised question per NRC comments. Change the correct answer, replaced one distractor and provided actual SR readings. Reordered choices based on length.

5.2. Glass containers or other breakable items are NOT to be used over the open reactor vessel or in the reactor cavity.

5.3. Anticipate criticality at any time when positive reactivity is being added to the core.

5.4. The response from at least one responding detector must be displayed continuously on a strip chart recorder when fuel is being added to the reactor vessel or moved from a temporary core location adjacent to one or more assemblies.

5.5. During CORE ALTERATIONS, at least two source range neutron flux monitors are required to be OPERABLE per T/S LCO 3.9.3.

5.6. When a fuel assembly has been inserted in the reactor, the refueling machine must remain connected to the assembly until the count rates have stabilized and physics monitoring personnel have signaled permission to unlatch.

5.7. During fuel assembly insertion, monitor neutron multiplication. Count rates should stabilize within approximately ten seconds after a fuel assembly has reached bottom. If count rates should continue to increase, indicating possible criticality, withdraw the fuel assembly and move it to the Rod Control Cluster (RCC) change fixture or the spent fuel pool. Suspend core alterations until the situation is evaluated.

5.8. IF any of the following conditions occur during core loading, SUSPEND core alterations pending the evaluation by the SRO in charge of core alterations and the Reactor Engineer:

- An increase in count rate by a factor of five occurs on any of the responding nuclear monitoring channels after the addition of any single fuel assembly, except initial load of the sources.
- An increase in count rate by a factor of two occurs simultaneously on all responding nuclear monitoring channels after the addition of any single fuel assembly, except initial load of the sources.
- Reactor vessel water boron concentration changes by 20 ppm or more from the previous value.
- The evacuation horn, coupled to one of the plant source range channels is activated other than for an announced test.
- Less than two of the responding nuclear channels are in service when both sources are loaded.
- Less than two of the responding nuclear channels have counting rates greater than or equal to 0.5 counts per second (after both source bearing assemblies are loaded in the vessel).
- A fuel assembly is damaged.
- Loss of communications. [Ref: 9.2.8]
- Any other condition occurs which the SRO or Reactor Engineer feels warrants suspending core loading.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Conduct of Operations	Group #	N/A		
	K/A #	G2.1.3		
	Importance Rating	3.7		
Knowledge of shift or short-term relief turnover practices				

Question # 68

You are the on-coming Unit Reactor Operator for the February 7th AM Shift (0700). Your last shift was finished at 0700 on February 5th.

(1) Per ODP-ZZ-00003, Shift Relief and Turnover, what describes the MINIMUM amount of RO logs that you must review prior to watch relief?

And

(2) Per ODP-ZZ-00003, PRIOR to relief you should ….?

A. (1) RO logs back to February 5th at 0700.
(2) Perform a Control Room annunciator test including the EFSAS status panel horns.

B. (1) RO logs back to February 5th at 0700.
(2) Check the Main Control Board equipment, indicating lights and inspect chart recorders for proper operation.

C. (1) RO logs back to February 6th at 0700.
(2) Perform a Control Room annunciator test including the EFSAS status panel horns.

D. (1) RO logs back to February 6th at 0700.
(2) Check the Main Control Board equipment, indicating lights and inspect chart recorders for proper operation.

Answer: C

Explanation:

Per ODP-ZZ-00003, step 4.1.8 " On-coming watchstanders shall review watchstation narrative log for the previous 24 hours, or back to their last watch, whichever is shorter." Therefore Feb 6th @0700 is correct. Back a period of 48 hours to Feb 5th @0700 is plausible as this is the last time the individual had the watch. The minimum requirement per the procedure is back to Feb 6th @0700 making it correct.

Per Section 4.5 of ODP-ZZ-00003, step 4.5.1 lists the requirement prior to relief "Perform a Control Room annunciator test, which should include testing ESFAS status panel horns. (Either the on-coming RO or BOP may perform the test, only one test is required)". This is the correct answer. Step 4.5.5 states that shortly after relief the RO and BOP should "Check the Main Control Board equipment, indicating lights and inspect all chart recorders for proper operation."

This is a plausible distractor as it is discussed in the correct section of the procedure and an action that will be taken but the candidate must know which is required prior to relieving the off going watch stander.

A. Incorrect – see above explanation
B. Incorrect – see above explanation
C. Correct - see above explanation
D. Incorrect – see above explanation

Technical Reference(s):
1. ODP-ZZ-00003, Shift Relief and Turnover, Rev 36

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #73 Objective A: In accordance with ODP-ZZ-00003, "Shift Relief and Turnover"
1. EXPLAIN the General Notes applicable to Shift Relief/Turnover.
2. EXPLAIN the specific requirements to complete the following Relief/Turnover:
   a. SM/CRS/FS/STA/SE
   b. URO/BOP

Question Source: Bank # ______
Modified Bank # ___X_R8352__
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Replaced K/A and question per NRC Comment

<QQ 8352(1410)><<You are the on-coming Unit Reactor Operator for the February 7th AM Shift (0700). Your last shift was finished at 0700 on February 5th.

Which of the following describes the minimum amount of RO logs that you must review prior to watch relief?>>

A. <QQ 8352(1482)><<Back to February 6th at 0700.>>

B. <QQ 8352(1480:0)><<Back to February 6th at 2300.>>

C. <QQ 8352(1480:1)><<Back to February 5th at 2300.>>

D. <QQ 8352(1480:2)><<Back to February 5th at 0700.>>

Answer: <QQ 8352(1419)><<A>>

Question 1 Info	
Question Type:	<QQ 8352(1401)><<Multiple Choice>>
Status:	<QQ 8352(1405)><<Active>>
Always select on test?	<QQ 8352(1406)><<No>>
Authorized for practice?	<QQ 8352(1429)><<No>>
Points:	<QQ 8352(1441)><<1.00>>
Time to Complete:	<QQ 8352(1408)><<3>>
Difficulty:	<QQ 8352(1407)><<2.00>>
System ID:	<QQ 8352(1445)><<8352>>
User-Defined ID:	<QQ 8352(1404)><<R8352>>
Cross Reference Number:	<QQ 8352(1409)><<ODP-ZZ-00003>>
Topic:	<QQ 8352(1400)><<R8352 ODPZZ03 RO logs that you must review prior to watch relief>>
Num Field 1:	<QQ 8352(1414)><<3.0>>
Num Field 2:	<QQ 8352(1415)><<3.4>>
Text Field:	<QQ 8352(1413)><<2.1.3>>
Comments:	

Question 1 History	
Exam Appearances:	<QQ 8352(1449)><<2>>
Student Encounters:	<QQ 8352(1448)><<15>>
Answered Right:	<QQ 8352(1452)><<15>>
Answered Wrong:	<QQ 8352(1453)><<0>>
Partially Correct:	<QQ 8352(1459)><<0>>
Answer Invalid:	<QQ 8352(1455)><<0>>
Unanswered:	<QQ 8352(1454)><<0>>
Ignore Response:	<QQ 8352(1460)><<0>>
Avg Points Awarded:	<QQ 8352(1450)><<1.00>>
... As % of Point Value:	100
Standard Deviation:	<QQ 8352(1456)><<0.00>>

Question 1 Table-Item Links

<TB 5114(1301)><<OPS Procedures>>

<TB 5162(1305)><<ODP-ZZ-00003, Shift Relief and Turnover>>

<TB 5818(1301)><<OPS Systems>>

<TB 5822(1305)><<ADMIN, Operations Administration>>

<TB 5972(1301)><<OPS Question Catagory>>

<TB 5975(1305)><<LO Requalification, Open Book>>

Associated objective(s):

<OB 16539(1101)><< A In accordance with ODP-ZZ-00003, "Shift Relief And Turnover"

1.EXPLAIN the General Notes applicable to Shift Relief/Turnover.

2.EXPLAIN the specific requirements to complete the following Relief/Turnover:
a.SS/CRS/FS/STA
b.URO>>

<OB 26220(1101)><< A In accordance with ODP-ZZ-00003, "Shift Relief And Turnover"

1.EXPLAIN the General Notes applicable to Shift Relief/Turnover.

2.EXPLAIN the specific requirements to complete the following Relief/Turnover:
a.SS/CRS/FS/STA
b.URO>>

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Equipment Control	Group #	N/A		
	K/A #	G2.2.14		
	Importance Rating	3.9		
Knowledge of the process for controlling equipment configuration or status.				

Question # 69

A component will be placed in an off normal position in support of maintenance.

Per ODP-ZZ-00035, Plant Status Control, what is the LATEST time the 10CFR50.59 evaluation MUST be performed?

A. Prior to repositioning the component

B. Prior to the End of Shift in which the component is repositioned

C. 30 days after the component is repositioned

D. 60 days after the component is repositioned

Answer: D

Explanation:
Per ODP-ZZ-00035, step 4.2.3 "IF the components are placed out-of-normal position in support of Maintenance or while waiting for WPA tags to be placed, GENERATE a Business Tracking CAR requiring a 10CFR50.59 evaluation to be performed within 60 days."

A. Incorrect, Plausible as it can be assumed that the evaluation must be performed before the component is repositioned thereby changing the plant configuration
B. Incorrect, Plausible as this is the requirement to place the plant status control tags in step 4.1.7.
C. Incorrect, Plasuible as this is the requirement for the work control supervisor to perform an audit every 30 days to ensure a 10CFR50.59 evaluation will be performed within 60 days per step #4.1.12.
D. Correct - See above explanation

Technical Reference(s):

1. ODP-ZZ-00035, Plant Status Control, Rev 16

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, LP #8, Objective A:

A. DESCRIBE the following as they pertain to ODP-ZZ-00035, “Plant Status Control”
   1. Purpose and Scope of procedure
   2. Responsibilities of the CRS/SM
   3. Responsibility of Work Control Supervisor with regards to Plant Status Control Tag audits.
   4. Station personnel authorized to manipulate plant equipment and exceptions
   5. Requirement for component manipulation as part of watchstation rounds
   6. Use of form CA2789, Plant Status Control Form.
   7. Use of Plant Status Control Tags
   8. When 10CFR50.59 evaluations are required to be performed including time restrictions.

Question Source: Bank # __X _L15416___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Replaced question per NRC comment.

4.1.6. CA2789, Plant Status Control Form provides a method of documentation of equipment manipulations that are not covered by any other plant process or program. A CA2789 form is not required if Plant Status Control Tags are being created at the time of manipulation.

4.1.7. Plant Status Control Tags are generated in accordance with ODP-ZZ-00310, WPA and Caution Tagging. Tags may be placed at any time during the shift, but MUST be placed prior to the end of shift or immediately after turnover.

4.1.8. Plant Status Control Tags can be used for Equipment Protection. [Ref: 5.2.6]

4.1.9. Work documents tied to Plant Status Control Tags are for tracking purposes only. Sign On is NOT required.

4.1.10. When work instructions do not document restoration, Maintenance tracks component positioning via a CA2884, Maintenance Plant Status Control Log per APA-ZZ-00320, Work Execution.

4.1.11. USE Attachment 2 Flowchart as necessary.

4.1.12. *Work Control Supervisor*, PERFORM an audit of Plant Status Control tags every 30 days to verify their requirement and to ensure a 10CFR50.59 evaluation will be performed within 60 days

4.2. <u>10CFR50.59 Evaluation</u>

4.2.1. REFER to Attachment 2 for flowchart for a visual depiction of when to apply the 50.59 Review process to PSC tagging.

4.2.2. DETERMINE if the PSC tag is for a Maintenance Activity or for an Operations Activity. This is critical to the applicability of 10 CFR 50.59.

a. Section 7 contains definitions for Maintenance Activity and Operations Activity.

4.2.3. IF the components are placed out-of-normal position in support of **<u>Maintenance</u>** or while waiting for WPA tags to be placed, GENERATE a Business Tracking CAR requiring a 10CFR50.59 evaluation to be performed within 60 days.

a. Component manipulation in support of maintenance is used anytime the equipment must be manipulated out of its normal position and should not be returned until maintenance has been performed.

Example of Maintenance Need:

- Isolating a valve that is not operating correctly or is broken, and requires a rebuild.

 INFORMATION USE

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Equipment Control	Group #	N/A		
	K/A #	G2.2.23		
	Importance Rating	3.1		
Ability to track Technical Specification limiting conditions for operations.				

Question # 70

Per ODP-ZZ-00002, Equipment Status Control, the reactor operator should initiate an EOSL for which of the following situations?

A. An Emergency Diesel Generator surveillance is started and completed on the same shift.

B. A Safety Injection Pump is out of service solely due to the INOPERABILITY of a support system.

C. A large oil leak is discovered and repaired on an RHR pump motor during surveillance testing.

D. A CCP is placed in pull-to-lock to support surveillance testing for 30 minutes during the shift and an entry is made into the Control Room Supervisors Log.

Answer: C

Explanation:

ODP-ZZ-00002 step 2.6 and the Caution before it state when and EOSL is not required:

"CAUTION - EOSL entries are made for ALL discovered inoperability of systems, subsystems, trains, components or devices regardless of the duration of the inoperability. The ONLY exception allowed is if ALL the requirements of Step 2.6 are met.

Step 2.6. When all of the following conditions are met, EOSL entries are NOT required:

- *The items are entered in the Control Room Supervisors Log*
- *They are out of service for surveillance, testing, or for Rad Monitor filter replacement.*
- *The out of service equipment is under the control of an operator or technician which allows the work to be suspended, and the equipment restored to full OPERABILITY in a timely manner."*

Furthermore, a note prior to step 4.2.2 states "Components which are out of service solely due to a support system inoperability need NOT have separate EOSLs."

Step 2.7 states "Track out of service time when a periodic surveillance is suspended and the equipment remains out of service condition."

A. Incorrect – all of the items in step 2.6 would be met. The EDG surveillance would be entered into the log, it's a surveillance, and under control of control room and local equipment operators, so NO EOSL is required. Plausible as it is an ECCS system with testing in progress.
B. Incorrect – Per the note prior to step 4.2.2, an EOSL is not required as it is solely due to the inoperability of a support system. Plausible as it is an ECCS system that is out of service.
C. Correct – per step 2.7 as stated above
D. Incorrect – all of the items in step 2.6 would be met, so NO EOSL is required. Plausible as this action places an ECCS pumps in a condition in which it would NOT automatically start when required.

Technical Reference(s):

1. ODP-ZZ-00002, Equipment Status Control, Rev 81, Section 4.5

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #74, ODP-ZZ-00002, Objective B: DESCRIBE the following as it pertains to ODP ZZ 00002, 'Equipment Status Control": 4. The process for:

i. Initiating an EOSL entry
ii. Clearing an EOSL entry
iii. Maintaining the EOSL

Question Source: Bank # __X L16697____
Modified Bank # ______
New _______

Question History: Last NRC Exam __N/A______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Replaced question per NRC comment and facility rep feedback

NOTE

Fire Protection out of service time is tracked by the Fire Protection Impairment Program in accordance with APA-ZZ-00701, Control of Fire Protection Impairments.

The Non-Safety Related Maintenance Rule Risk Significant SSCs, referenced in the following step as requiring an Equipment Out of Service Log (EOSL), are limited to the following:

- Condensate Storage Tank out of service
- Service Water Pumps out of service
- Switchyard work making one or both offsite sources unavailable to the NB busses
- Startup Transformer out of service including its output breaker PA0201

2.3. Track out of service time for the following:

- Maintenance activities, except those excluded by Step 2.6, which render a Safety Related and Non-Safety Related Maintenance Rule Risk Significant SSC inoperable *(excluding Fire Protection).*
- Corrective maintenance performed in conjunction with normal surveillance's on Safety Related and Non-Safety Related Maintenance Rule Risk Significant SSC *(excluding Fire Protection).*
- A component that fails its surveillance, or passes its late date.

2.4. Track out of service time for Plant Shutdown Action Statements as described in APA-ZZ-00703, Fire Protection Operability and Surveillance Requirements. Otherwise, Fire Protection components are tracked as described in APA-ZZ-00701, Control of Fire Protection Impairments.

2.5. Track impairments to components required to be operable in accordance with FSAR 16.11 and Technical Specifications.

CAUTION

EOSL entries are made for ALL discovered inoperability of systems, subsystems, trains, components or devices regardless of the duration of the inoperability. The ONLY exception allowed is if ALL the requirements of Step 2.6 are met.

2.6. When all of the following conditions are met, EOSL entries are NOT required:

- The items are entered in the Control Room Supervisors Log
- They are out of service for surveillance, testing, or for Rad Monitor filter replacement.
- The out of service equipment is under the control of an operator or technician which allows the work to be suspended, and the equipment restored to full OPERABILITY in a timely manner.

2.7. Track out of service time when a periodic surveillance is suspended and the equipment remains out of service condition.

2.8. Track time a component is out of service in accordance with an OTO attachment, even for a short duration.

2.9. At the discretion of the Operations Department, "Information Only" EOSLs may be written to track any out of service plant component or to create an on line list of work activities as a means of tracking work. Information Only EOSLs are labeled as discussed in Section 4.0.

3.0 RESPONSIBILITIES

3.1. Director, Nuclear Operations or Designated Alternate

3.1.1. Ensures Operations personnel are familiar with this procedure.

3.1.2. Reviews those EOSL System Shift Turnover Reports which have time tracked components.

3.1.3. Ensures EOSL System Shift Turnover Reports, which track time associated with inoperable equipment, are processed as QA Records in accordance with APA-ZZ-00220, Records Management.

3.2. Shift Manager (SM) or Designee

3.2.1. Ensures the EOSL System is maintained in accordance with this procedure.

3.2.2. Determines operability.

3.2.3. Reviews and sign the Shift Turnover Report. [Ref. 5.2.14]

3.2.4. Reviews EOSL System prior to plant mode changes to ensure equipment required in the entered mode is operable. [Ref. 5.2.23]

3.2.5. Determines required testing in order to declare a component operable.

3.3. Construction Supervisor/Design Engineer

3.3.1. Provides the Shift Manager with information, as requested, regarding the scope and description of any modification work.

3.3.2. Notifies the Shift Manager when modification related work and documentation updates have been completed satisfactorily.

-END OF SECTION-

4.0 COMPUTERIZED EOSL SYSTEM

4.1. Accessing EOSL System

NOTE

Most plant personnel may log on in the Inquiry mode to obtain EOSL information. Typically, only NRC licensed personnel and STA qualified personnel may log on in the Update mode to change or update system data.

4.1.1. LOG on to the EOSL System by logging on to Work Management application and CHOOSE the menu OPERATIONS and SELECT EOSL.

4.1.2. PERFORM inquiries in the EOSL using Criteria Option under EOSL menu in Work Management application.

-END OF SECTION-

4.2. Creating an EOSL

NOTE

Attachment 3 contains system/component Operability Evaluations listed by Technical Specification number.

4.2.1. REVIEW the following:

- Attachment 2 for Preparation Guidelines
- Attachment 3 for applicable Operability Evaluations.

NOTE

Components which are out of service solely due to a support system inoperability need NOT have separate EOSLs.

4.2.2. WHEN a component needs to be tracked in the EOSL, CREATE an EOSL record using the computerized EOSL application.

4.2.3. IF using a Master EOSL record, PERFORM the following:

a. CLICK MORE on the details page of an EOSL record.

b. SELECT ADD T/S and Location from Master.

NOTE

Master EOSL Records allow the user to import technical specifications and location information including SFDP that is already available on Master EOSL Records.

c. COMPLETE remaining information after Master information is imported.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Radiation Control	**Group #**	N/A		
	K/A #	G2.3.11		
	Importance Rating	3.8		
Ability to control radiation releases.				

Question # 71

A liquid radwaste release is in progress.

1 sample was taken and analyzed prior to the release.

Annunciator 61B, Process Radiation Hi, is LIT. HB-RE-0018, RW BLD Disch Line Gamma DET, is in Alert (Yellow Alarm).

Subsequently, Radiation Protection reports that HB-RE-0018 has failed AS IS.

The Reactor Operator MUST …?

A. Direct Chemistry to obtain and analyze effluent samples.

B. Suspend release of radioactive effluents via this pathway.

C. Verify the liquid radwaste discharge automatically terminates.

D. Verify that process radiation readings have not changed on HB-RE-0018's RM-23 unit.

Answer: B

Explanation:
Note: A high alarm does not automatically terminate this release, a hi hi does; nor does a rad monitor failure "as is" terminate a release in progress per Attachment 20 of OTA-SP-RM011. A Low dilution water flow would also automatically terminate the release.

Step 3.f of Attachment 20 of OTA-SP-RM011 directs the reactor operator to refer to OOA-SP-00002 PRM T.S. / FSAR actions. This OOA then list the FSAR section of 16.11.1.3 table 16.11-2 item 1a.

In FSAR 16.11.1.3 "RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION LIMITING CONDITION FOR OPERATION" and per Table 16.11-2 Radioactive Liquid Effluent Monitoring Instrumentation with the 1 required instrument inoperable, action 31 applies which has actions PRIOR to initiating a release but since a release is in progress the action of "otherwise suspend release of radioactive effluents via this pathway" is correct. This is considered an immediate action (i.e 1 hour or less required action) making it RO knowledge.

A. Incorrect – This is step 3.c of Attachment 20 of OTA-SP-RM011 and applies if the alarm was RED (i.e. Hi Hi). Furthermore, per the actions of the FSAR, 16.11-2 Radioactive Liquid Effluent Monitoring Instrumentation with the 1 required instrument inoperable, at least 2 independent samples can be analyzed to allow effluent release via this pathway PRIOR to initiating the release. As a release is in progress this does not apply making this plausible but incorrect and only 1 sample was taken and analyzed.
B. Correct – see above explanation
C. Incorrect – The release will automatically on terminate Hi Hi not Hi nor does it terminate a release in progress when a rad monitor fails
D. Incorrect – Per OTA-SP-RM011 steps 3.4.2 explains that HB-RE-0018 does not have a RM23 unit but is required per the FSAR. This is plausible as several required rad monitors have RM 23 units as shown in step 3.4.1. Furthermore per Attachment 20 of OTA-SP-RM0111, the first action is to check trends to validate the alarm per release permit which adds to the plausibility to verify the process radiation readings are stable.

Technical Reference(s):
1. HTP-ZZ-02006, Liquid Radwaste Release Permit, Rev 88
2. OTA-RK-00020, Annunciators
    a. 61C, Rev 2
    b. 61B, Rev 0
3. OTA-SP-RM011, RM11 Control Panel, Rev 40 Attachment 20
4. OOA-SP-00002, PRM Tech Spec / FSAR Actions, Rev 19
5. FSAR Table 16.11-2 Radioactive Liquid Effluent Monitoring Instrumentation, OL-14 Rev 12/04

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #16, Radwaste systems, Objective E: DESCRIBE the purpose and operation of the following Liquid Radwaste components / subsystems:
7. Liquid Release and Isolation
8. Liquid Radwaste Sampling / Monitoring

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(11)

Comments:
k/a match "Generic – ability to control rad releases" as it places the operator in the situation of a liquid rad release in progress with a subsequent rad monitor failure and asks what action will the operator take. Generic as it does not as system setpoints, interlocks etc. Only a specific rad monitor noun name and annunicator titles are provided for operational reference for the candidate.

TABLE 16.11-2 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT			MINIMUM CHANNELS OPERABLE	ACTION
1.	Radioactivity Monitors Providing Alarm and Automatic Termination of Release			
	a.	Liquid Radwaste Discharge Monitor (HB-RE-18)	1	31
	b.	Steam Generator Blowdown Discharge Monitor (BM-RE-52)	1	32
2.	Flow Rate Measurement Devices			
	a.	Liquid Radwaste Blowdown Discharge Line (HB-FE-2017)	1	34
	b.	Steam Generator Blowdown Discharge Line (BM-FE-0054)	1	34
	c.	Cooling Tower Blowdown and Bypass Flow Totalizer (FYDB1017A)	1	34
3.	Discharge Monitoring Tanks (DMT's) Level			
	a.	DMT A(HB-LI-2004)	1	33
	b.	DMT B(HB-LI-2005)	1	33

ACTION STATEMENTS

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Section 16.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Radiation Control	Group #	N/A		
	K/A #	G2.3.14		
	Importance Rating	3.4		
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.				

Question # 72

An Operator is reviewing a Radiation Work Permit (RWP) for an upcoming surveillance in the Auxiliary building. The RWP has the following information:

- General area dose rates are 200 mrem/hr at 30 centimeters.
- Total Airborne Particulate (including tritium, halogens, and noble gases) is 0.5 DAC.
- Loose surface contamination of 5,000dpm/100 cm^2 beta-gamma.

This operator has an accumulated dose for the current year of 800 mrem.

(1) What are the radiological hazards associated with this surveillance?

And

(2) What is the MAXIMUM time the operator can stay in this area without exceeding their administrative limit?

A. (1) Contaminated Area
(2) 1 hour

B. (1) Contaminated Area
(2) 6 hours

C. (1) Airborne Radioactivity Area
(2) 1 hour

D. (1) Airborne Radioactivity Area
(2) 6 hours

Answer: B

Explanation:

Per HDP-ZZ-01500 section 6.10 and 6.11, this area is a contaminated area because " Loose contamination > 1000 dpm/100 cm² beta-gamma or > 20 dpm/100 cm² alpha on any normally accessible surface," It is not a Airborne Radioactivity Area as defined in section 6.11 of this procedure because total airborne must be greater than 1DAC. .5 DAC is plausible as there is an level of 0.3DAC for alpha, particulate, tritium and halogens combined.

The Callaway admin limit per year is 2000mrem. At 200 mrem/hr the operator can stay for 6 hours. The distractor of 1 hour is if it is believed that the admin limit is 1000mrem. 1000 mrem/hr at 30 cm is the minimum limit for an area to be classified as a Locked High Radiation Area per HDP-ZZ-01500 section 6.3.

A. Incorrect – wrong time
B. Correct
C. Incorrect – both are wrong
D. Incorrect – wrong hazard

Technical Reference(s):
1. HDP-ZZ-01500, Radiological Postings, Rev 44

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #75, RB Entry, Objective E: HDP-ZZ-01500, Radiological Posting

1. DEFINE and DISCUSS the posting requirements of the following areas:
   a. Radiation Area
   b. High Radiation Area
      - HRA
      - LHRA
   c. Very High Radiation Area (VHRA)
   d. Contaminated Area
   e. Radioactive Materials Area
   f. Airborne Radioactivity Area
   g. Hot Spot

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(12)

Comments:

Updated explanation per NRC Comments

6.10. Contaminated Area

NOTE

A Contaminated Area (CA) is any area in which there exists the following:

- Loose contamination > 1000 dpm/100 cm^2 beta-gamma or > 20 dpm/100 cm^2 alpha on any normally accessible surface,
- Surveys have NOT been taken and contamination is suspected,
- Activities, such as opening of a system containing radioactive material without positive containment, which could release loose surface contamination to the surrounding area.

6.10.1. POST a Standard Radiological Warning Sign so it is visible from on all sides and entrances to the Contaminated Area with the words:

CAUTION

CONTAMINATED AREA

6.10.2. ESTABLISH postings and boundaries:

- in the lowest dose rate area as practical;
- to prevent inadvertent contact with contaminated surfaces

6.10.3. ENSURE the barricade meets the requirements of Section 6.1.

NOTE

Contaminated Area boundaries with durations of less than one shift do NOT require Contaminated Area Tape below rope boundary.

Contaminated Area Tape may be continuous or segmented.

6.10.4. PLACE Contaminated Area Tape on floor, marking Contaminated Area boundary.

6.10.5. CONSIDER the following:

a. Contaminated Area Tape is placed so that the wording can be read when approaching the area from the uncontaminated side.

b. The barricade does NOT have to extend across the entrance of a Contaminated Area as long as:

- Area is NOT posted as HRA, LHRA or VHRA
- Area is NOT Contact RP for Survey Prior to Entry
- Entrance has a Step Off Pad and Contaminated Area Tape between the Step Off Pad and Contaminated Area
- Entrance has a stanchion on each side with radiological postings on each stanchion

6.10.6. USE Contaminated Area Tape or equivalent (such as pre-printed stickers) as the sole boundary for any area where:

- There is a raised pedestal (e.g., pump pedestals, sump pedestals, counters, sinks, or sample points)
- Valve stems/packing areas, and pump seal areas/housings

6.10.7. For frequently entered Contaminated Areas, INSTALL a Step Off Pad for exits.

6.10.8. For infrequently entered Contaminated Areas, PLACE a Step Off Pad if desired.

NOTE

Room 3101 is exempt from Radioactive Material Area around Contaminated Areas since room is inside of RCA where workers are trained to work around Contaminated Areas.

6.10.9. IF Contaminated Area is outside Auxiliary, Fuel Handling, Radwaste, or Reactor Buildings, PLACE a Radioactive Material Area (RMA) around Contaminated Area to prevent inadvertent removal of contaminated items.

-END OF SECTION-

6.11. Airborne Radioactivity Area

NOTE

An Airborne Radioactivity Area (ARA) is any room, enclosure, or area in which airborne radioactive materials exist such that the sum of the ratios of the concentration to Derived Air Concentrations (DAC) is in excess of:

- Noble gases - 1 DAC
- Alpha, particulate, tritium and halogens combined - 0.3 DAC
- Total of particulate, tritium and halogens and noble gases combined - 1 DAC

PLACE a Standard Radiological Warning Sign so it is visible from on all sides and entrances to the Airborne Radioactivity Area with the words:

CAUTION

AIRBORNE RADIOACTIVITY AREA

-END OF SECTION-

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	Group #	N/A		
	K/A #	2.4.8		
	Importance Rating	3.8		
Knowledge of how abnormal operating procedures are used in conjunction with EOPs.				

Question # 73

An event has occurred and the crew entered the appropriate abnormal procedure.

Several minutes later, the reactor was tripped in accordance with the abnormal procedure.

The crew has entered E-0, Reactor Trip Or Safety Injection.

In accordance with ODP-ZZ-00025, EOP/OTO User's Guide, what describes the CONCURRENT procedure use requirements of the abnormal operating procedure?

A. NOT allowed at anytime during performance of E-0.

B. May be used concurrently with E-0 anytime it is deemed necessary.

C. May ONLY be used concurrently with E-0 with the approval of the SM AND CRS.

D. May be used concurrently with E-0 after step 1 of E-0 and if directed by abnormal procedure.

Answer: D

Explanation: *Per ODP-ZZ-00025, step 4.12.1 " Concurrent procedure use is not permitted during performance of immediate action steps. The only exception is the performance of action(s) that should be performed promptly after a reactor trip in several OTOs. The specified action(s) may be performed after verifying the reactor trip in E-0 Step 1 and concurrently with the remaining immediate action steps of E-0. This requirement will be identified by written instructions in the associated OTO."*

A. Incorrect – Concurrent use may be permitted as described above but plausible as the EOPs are a higher level procedure than OTO (abnormal operating procedures)
B. Incorrect – Plausible as the CRS "determines how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress." But wrong as it is not permitted as described above i.e. "The specified action(s) may be performed after verifying the reactor trip in E-0 Step 1" therefore anytime is not correct.

C. Incorrect – Plausible as section 4.27, Inadequate Procedure, requires both the CRS with the concurrence of the SM, prior to action being taken. The candidate may falsely assume that this is the situation when trying to use an OTO during EOP usage. Also Plausible as section 4.15 of ODP-ZZ-00025, section 4.15 states that OTOs are entered based on "Direction from the SM/CRS" but incorrect as that does not apply to the concurrent procedure use specific situation that the stem is asking about. The use of AND "SM and CRS" was included because it could be argued that 'CRS only' may be correct because of a statement in step 4.12 "The CRS determines how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress."
D. Correct – See above explanation

Technical Reference(s):
1. ODP-ZZ-00025, EOP/OTO User's Guide, Rev 27, Section 4.12

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #1, ERG Introduction and User' s Guide, Objective AA: DESCRIBE the General Procedural Guidance provided by ODP-ZZ-00025, EOP/OTO User's Guide.

Question Source: Bank # ____X L16375____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised per NRC comments, added standard in the question stem and revised distractor B.

4.12. Concurrent Procedure Use

The CRS determines how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress. The following general rules should be observed:

4.12.1. Concurrent procedure use is not permitted during performance of immediate action steps. The only exception is the performance of action(s) that should be performed promptly after a reactor trip in several OTOs. The specified action(s) may be performed after verifying the reactor trip in E-0 Step 1 and concurrently with the remaining immediate action steps of E-0. This requirement will be identified by written instructions in the associated OTO.

4.12.2. If implementing E-0, concurrent procedure use should be avoided until exiting E-0 or until monitoring of CSFSTs begins within E-0.

4.12.3. ROs may be given the responsibility to implement the lower priority procedure when procedures are being performed concurrently.

4.12.4. While performing EOPs, plant conditions may indicate the need to correct problems not directly related to the event mitigation strategy. The operator may perform OTOs, OTNs and OTAs which address these problems as long as the actions do not interfere with performance of the EOPs.

4.13. Notes and Cautions

4.13.1. A Note contains administrative or advisory information which supports operator action contained in the procedure step.

4.13.2. A Caution contains information about potential hazards to personnel or equipment. They also advise on actions or transitions that may become necessary depending on changes in plant conditions.

4.13.3. Notes and Cautions apply to the step which they precede.

4.13.4. Both Notes and Cautions are introduced by their descriptor, in bold letters, followed by the text extending across both columns.

4.13.5. A Note or Caution which precedes the first operator action step may also apply to the entire procedure.

4.13.6. One type of Note deserves special mention:

Several procedures contain steps which are designated as "immediate actions." These steps are intended to be performed, if necessary, without the written procedure being available. These procedures contain a Note advising which steps are "immediate action" steps.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	Group #	N/A		
	K/A #	G2.4.29		
	Importance Rating	3.1		
Knowledge of the emergency plan				

Question # 74

What is the LOWEST emergency classification level at which the Field Supervisor or designee is REQUIRED to perform Attachment 2 of EIP-ZZ-00102, Operations Personnel Emergency Actions?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

Explanation:

Per Attachment 5 of EIP-ZZ-00102, the flowchart asks if the emergency is at the ALERT level or Higher and if yes the Field Supervisor or other personnel in the Field Office will be assigned Attachment 2.

A. Incorrect – Plausible as this is an Emergency classification level but incorrect because it is not required per EIP-ZZ-00102.
B. Correct – see explanation above
C. Incorrect – Accountability is performed at a site area emergency or higher. Plausible distractor as a candidate may confuse accountability with the requirement to perform Attachment 2.
D. Incorrect – PARs (Protective Action Recommendation) are required for General Emergencies but plausible distractor as the candidate may confuse PARs with this or believe PARs is apart of Attachment 2.

Technical Reference(s):
1. EIP-ZZ-00102, Emergency Implementing Actions, Rev 59
2. EIP-ZZ-00230, Accountability, Rev 34

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T68.1020 RERP, Objective D: Explain the actions to be taken to respond to an emergency classification following an event declaration, per EIP-ZZ-00102 including required paperwork.

Question Source: Bank # _____
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

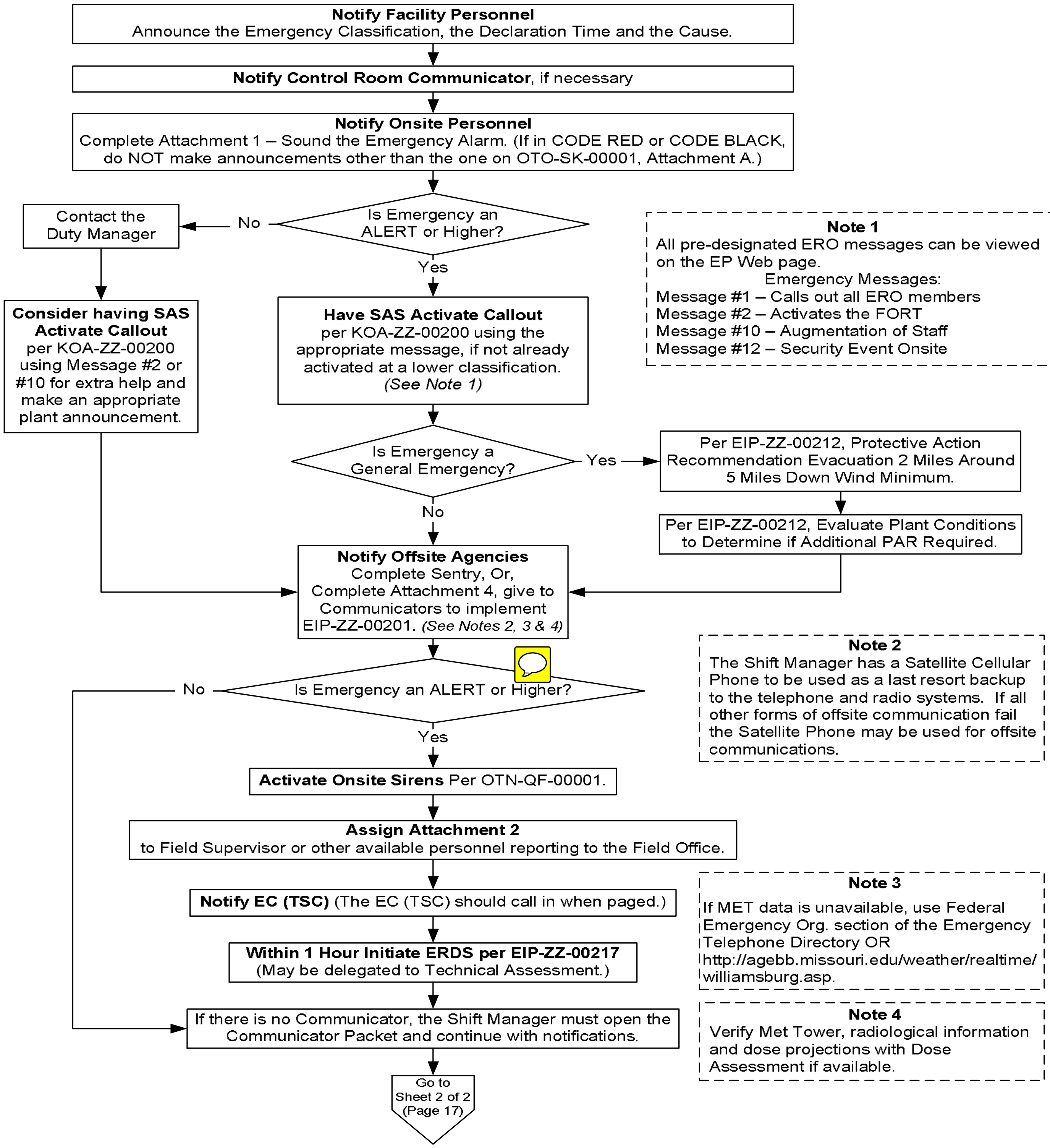
Comments:

Revised question based on NRC comment about general employee training to a task specific to operations personnel and the emergency plan.

This question is at the RO level as the Field Supervisor is either a RO or lead Operations Technician (OT) and the question is asking about when this person would be performing an action during a declared emergency.

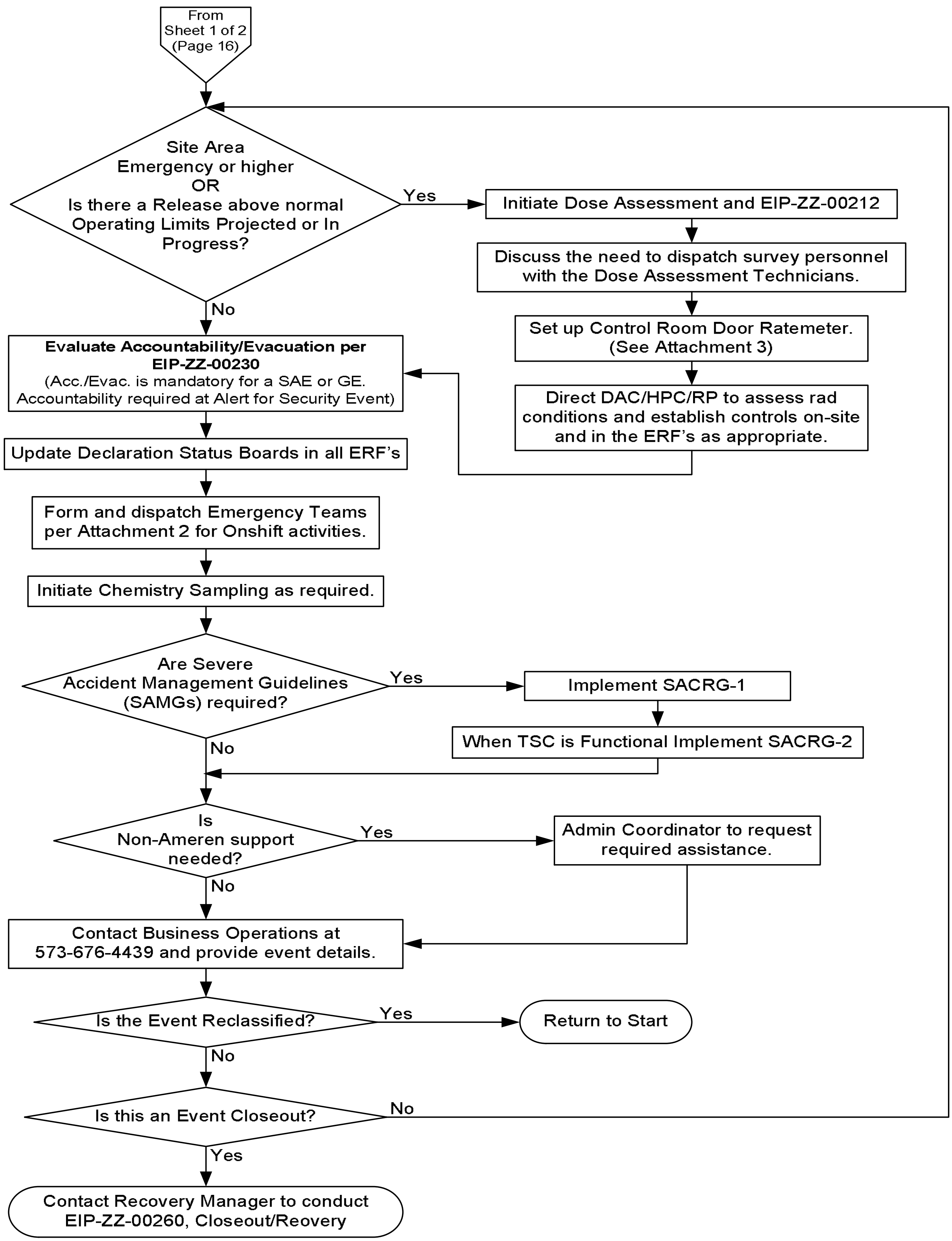
Attachment 5
Emergency Coordinator Flowchart for Declared Emergencies
Sheet 1 of 2

> If there is an Imminent Threat or Attack against the station, an accelerated NRC call is required within approximately 15 minutes of notification of the Threat or Attack. Refer to the "Accelerated NRC Call" Section of EIP-ZZ-00201.



Attachment 5 (Cont'd.)

Sheet 2 of 2



Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	Group #			
	K/A #	G2.4.42		
	Importance Rating	2.6		
Knowledge of emergency response facilities.				

Question # 75

A General Emergency has been declared.

Which emergency response facility will coordinate the OFFSITE radiological monitoring during emergencies and recovery operations after all facilities have assumed their prescribed function(s)?

A. Control Room

B. Technical Support Center

C. Emergency Operation Facility

D. Joint Public Information Center

Answer: C

Explanation:

The recovery manager, person in charge of the EOF facility, will have taken the responsibility from the control room for making Protective Action Recommendations per EIP-ZZ-C0010 step 3.1.3a.

A. Incorrect - Plausible as the control room is charge in during UE.
B. Incorrect – this facility is required to be activated during an alert but will have been transferred to the EOF and the recovery manager from the EC.
C. Correct
D. Incorrect - Plausible as this is the last Emergency Response facility and location for the backup EOF, but no information was given about the need to man the back EOF so therefore it is incorrect.

Technical Reference(s):

1. EIP-ZZ-C0010, Emergency Operations Facility Operations, Rev 39

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: None

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Revised question stem per NRC Comment

EMERGENCY OPERATIONS FACILITY OPERATIONS

1.0 PURPOSE

The purpose of this procedure is to provide guidance to Emergency Response Personnel who report to the Emergency Operations Facility (EOF) and Backup EOF (BEOF).

2.0 SCOPE

2.1. Only those Emergency Response Personnel who report to the EOF or the BEOF are covered by this procedure.

2.2. On-Shift, TSC and JPIC guidance is contained in other Emergency Implementing Procedures.

3.0 RESPONSIBILITIES

3.1. Recovery Manager (RM)

3.1.1. Ensures that the EOF and the BEOF become operational after notification of an ALERT, SITE or GENERAL EMERGENCY classification.

3.1.2. Has overall command and control of the entire Callaway Plant Emergency Response Organization.

3.1.3. Additional duties include the following: [Ref: 5.2.3, 5.2.4]

a. The following responsibilities may only be performed by the Recovery Manager:

- Authorizing notifications to off-site agencies.
- Assuming responsibility from the Control Room for making Protective Action Recommendations.
- Authorizing personnel exposure in excess of 10CFR20 limits (the Emergency Coordinator also has this authority).

Step 3.1.3 Cont'd

b. The following responsibilities may be delegated by the Recovery Manager:

- Establishing and maintaining communications with the Emergency Coordinator.
- Requesting off-site support (e.g., NSSS, A/E, INPO, Federal, State and Local).
- Ensuring responsibility for notifications and communications with off-site agencies is transferred from the Control Room to the EOF Emergency Response Organization (excluding NRC ENS communications).
- Maintaining command and control over personnel in the EOF and providing considerations necessary for their safety.
- Ensuring coordinated emergency response among Callaway Plant and off-site agencies.

3.2. Protective Measures Coordinator (PMC)

3.2.1. Reports to the RM.

3.2.2. Responsible for formulating Protective Action Recommendations (PARs).

3.2.3. Assists the RM, State, and Federal Officials in the interpretation of any plant related data.

3.3. Plant Assessment Coordinator (PAC)

3.3.1. Reports to the PMC.

3.3.2. Reviews plant conditions and EALs to verify the adequacy of the existing PARs.

3.3.3. Assists in formulating new PARs, when necessary.

3.4. Plant Assessment Staff

3.4.1. Reports to the PAC.

3.4.2. Knowledgeable in plant equipment, systems, and operations.

3.4.3. May provide additional technical expertise while maintaining status boards displaying plant conditions.

3.5. Dose Assessment Coordinator (DAC)

3.5.1. Reports to the PMC (or the RM if the PMC has not arrived).

3.5.2. Responsible for providing dose projection calculations based on radiological effluent monitors and field data.

3.5.3. Directs Field Monitoring Teams (FMTs).

3.5.4. Reviews effluent based EALs and assists the PMC in formulating PARs. [Ref: 5.2.5]

3.6. Assistant DAC

3.6.1. Reports to the DAC.

3.6.2. Assists the DAC as directed.

3.7. Dose Assessment Staff

3.7.1. Reports to the DAC.

3.7.2. Responsible for FMT communications.

3.7.3. Responsible for updating radiological status boards. [Ref: 5.2.6]

3.8. Field Monitoring Teams (FMTs)

3.8.1. Dispatched by the DAC.

3.8.2. Responsible for taking direct radiation measurements and collecting air samples.

3.8.3. Responsible for collecting environmental sample media (soil, water and vegetation samples) in cooperation with the State Department of Health.

3.9. Logistical Support Coordinator (LSC)

3.9.1. Reports to the RM.

3.9.2. Responsible for contracting with vendors for engineering services, materials, and services needed for emergency mitigation and restoration.

3.9.3. Provides administrative and logistical support to the Emergency Response Organization (ERO).

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
Loss of Rx Coolant Makeup	Group #	1		
	K/A #	022AA2.01		
	Importance Rating	3.8		
Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exists				

Question # 76

Reactor Power is 100%.

- All control systems are in normal alignment.
- Letdown flow is 75 gpm on BG FI-132, LTDN HX OUTLET FLOW.

The following parameters are now noted on the CVCS system:

- Seal Return Flows are 3 gpm per Reactor Coolant Pump.
- Charging flow is 94 gpm and rising.
- BG TI-130, LTDN HX OUTLET TEMP, has risen 25°F from its steady state value.
- Volume Control Tank level is 55% and lowering.
- Pressurizer level is 53% and lowering slowly.
- Reactor Coolant System (RCS) average temperature is 585°F and stable.

Which of the following describes the correct leak location AND associated leakage monitoring requirements?

The Leakage is from the ....

A. letdown line between the letdown isolation valves and the orifices valves. This leakage must be monitored as post accident recirculation flowpath leakage, as required by TS 5.5.2, Primary Coolant Sources Outside Containment.

B. charging line between the flow indicator and Containment. This leakage is required to be monitored as a radiological effluent, as required by TS 5.5.4, Radioactive Effluent Controls Program.

C. letdown line between the letdown isolation valves and the orifices valves. This leakage is required to be monitored as a radiological effluent, as required by TS 5.5.4, Radioactive Effluent Controls Program.

D. charging line between the flow indicator and Containment. This leakage must be monitored as post accident recirculation flowpath leakage, as required by TS 5.5.2, Primary Coolant Sources Outside Containment.

Answer: D

Explanation:

A. INCORRECT. Wrong location for leak. This leakage would cause a rise in letdown flow through the Regen HX and a lowering on Regen HX Outlet Temp - Letdown. However, this leakage would be inside Containment and would not apply to the TS 5.5.2.
B - INCORRECT. Correct location for leak but leakage outside containment would be TS 5.5.2.
C - INCORRECT. Wrong location for leak. Leak is in the charging header. This leakage is required to be monitored as per TS 5.5.2.
D - CORRECT. Correct location for leak. This leakage would cause a temperature rise on Regen HX Outlet Temp - Letdown due to less charging flow to cool the letdown exiting the RHX. Since charging flow increased from a normal value of 87 GPM to 94 GPM with no other changes the leak rate is approximately 7 GPM and will required isolation of the charging header to isolate the leak.

TS 5.5.4 is plausible due to a leak being present. The SRO candidate must be able to determine that even though a leak is present it is not being tracked and monitored as a radiological effluent since it is contained within the Aux Building.

Technical Reference(s):
1. M-22BG01, 02, 03, and 04. P&ID for CVCS Sheets 1-4. Rev 33,29,56,21 respectively
2. TS 5.5.2, Primary Coolant Sources Outside Containment

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #11 Chemical & Volume Control System / Reactor Makeup Water System – BG Objective, U. LIST the LCOs associated with Charging Pumps from the FSAR.

Question Source: Bank # ______
Modified Bank # ______
New ___X Based on Q 85 on 2012 DC Cook exam____

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(b)(1) - The required actions for not meeting administrative controls listed in Technical Specification (TS) Section 5

Comments:

k/a match as the question requires operator to determine where the leak exists (RO knowledge) and then interpret the leakage data and determine the required monitoring is per Section 5.5.2 of Tech Specs.(SRO knowledge).

Replaced k/a and question based on NRC Comment

5.5 Programs and Manuals (continued)

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the recirculation portion of the Containment Spray, Safety Injection, Chemical and Volume Control, and Residual Heat Removal. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: A dose rate of $\leq$ 500 mrem/yr to the whole body and a dose rate of $\leq$ 3000 mrem/yr to the skin, and

2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: A dose rate of $\leq$ 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents released to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190;

k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section 3.9(N).1.1, "Design Transients", cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

(continued)

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	1		
Loss of Component Cooling Water	**Group #**	1		
	K/A #	00026 AA2.01		
	Importance Rating	3.5		
Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: Location of a leak in the CCWS				

Question # 77

Reactor Power is 100%.

- "B" Component Cooling Water (CCW) train is running supplying the Service Loop.
- "B" CCW surge tank is at 20% and lowering.
- "A" CCW surge tank indicates 55% and stable.
- VCT level is rising.

The Crew has entered OTO-EG-00001, “CCW System Malfunction”

What is the location of the leak and what procedure attachment will the CRS direct the crew to use?

A. The leak is in the Letdown Heat Exchanger and the CRS will direct use of Attachment B, “CCW Train B Leak”

B. The leak is in the Seal Water Heat Exchanger and the CRS will direct use of Attachment B, “CCW Train B Leak”

C. The leak is in the Letdown Heat Exchanger and the CRS will direct use of Attachment D, “Transferring Service Loop From Train B to Train A”

D. The leak is in the Seal Water Heat Exchanger and the CRS will direct use of Attachment D, “Transferring Service Loop From Train B to Train A”

Answer: B

Explanation:

With the VCT level going up and the B CCW surge tank going down the leak is from the CCW system in the Seal Water Heat Exchanger. A leak in the Letdown Heat Exchanger would cause the CCW surge tank to rise. Attachment D would be used if the leak was not in the service loop and the leak was in the B CCW loop. Attachment B contains direction on how to isolate a leak in the Seal Water HX even though the seal water heat exchanger is in the Service Loop.

A. Incorrect, Plausible in the operator incorrectly assumes a leak in the Letdown HX would cause CCW surge tank level to lower and VCT level to rise. The action is correct.
B. Correct, See above
C. Incorrect, Plausible in the operator incorrectly assumes a leak in the Letdown HX would cause CCW surge tank level to lower and VCT level to rise. The action is plausible if the operator incorrectly identifies the location of the leak and takes action to maintain CCW flow to the RCPs
D. Incorrect, The location of the leak is correct, The action is plausible if the operator incorrectly identifies the location of the leak and takes action to maintain CCW flow to the RCPs

Technical Reference(s):
1. OTO-EG-00001, "CCW System Malfunction", Rev 14

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, Lesson B-17, OTO-EG-00001 CCW System Malfunction, Objective E, Given a set of plant conditions or parameters indicating a CCW System Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ____NA________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

SRO Justification

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? NO

Can the question be answered solely by knowing immediate operator actions? NO

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? NO

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? NO

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES**

OTO-EG-00001

CCW SYSTEM MALFUNCTION

Revision 014

CONTINUOUS USE

A. PURPOSE

This procedure provides instructions for a leak or loss of flow in the Component Cooling Water (CCW) system.

B. SYMPTOMS OR ENTRY CONDITIONS

1) CCW Leak Symptoms:

   a. Lowering CCW Surge Tank Level.

   b. Lowering CCW flow.

   c. Rising temperatures on components supplied by CCW.

   d. Any of the following Control Room annunciators in alarm:

   - Annunciator 51D, CCW Srg Tk A Lev HiLo
   - Annunciator 52F, CCW To Aux Comp Flow Hi
   - Annunciator 53D, CCW Srg Tk B Lev HiLo

2) CCW Pump/Flowpath Problem Symptoms:

   a. Rising temperatures on components supplied by CCW.

   b. Any of the following Control Room annunciators in alarm:

   - Annunciator 51B, CCW Pmp A/C Trouble
   - Annunciator 51C(53C,52C,54C), CCW Pmp A(B,C,D) Flow Lo
   - Annunciator 52B, CCW Pmp A/C Press Lo
   - Annunciator 53B, CCW Pmp B/D Trouble
   - Annunciator 53F, CCW To Aux Comp Flow Lo
   - Annunciator 54B, CCW Pmp B/D Press Lo
   - Annunciator 54F, Seal Hx Flow HiLo

C. REFERENCES

1) Implementing:

   a. ODP-ZZ-00001 Addendum 13, Shift Manager Communications

   b. OTN-EG-00001, Component Cooling Water System

2) Developmental:

   a. M-22EG01, P&ID Component Cooling Water System

   b. M-22EG02, P&ID Component Cooling Water System

   c. M-22EG03, P&ID Component Cooling Water System

   d. CARS 200301950, EGHV0061 Failed OSP-EG-V002A

   e. RFR 016805D, Clarify Results Of Several EGHV0132 Evaluations

   f. RFR 016805E, Change To Administrative Controls For EGHV0132

   g. CAR 201007678, Technical Specification (T/S) Bases 3.6.3 are being met

   h. MP 10-0009 New RCP Seals

   i. Westinghouse DW 09-010 RCP Seals

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- OTO-BB-00002, Reactor Coolant Pump Offnormal contains instructions for a loss of CCW to the RCPs.
- RCPs that lose Seal Injection AND CCW To Thermal Barrier Heat Exchanger must have at least one restored within 6 minutes or the RCP MUST be secured.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1.	**CHECK One CCW Pump Running For Each Operating Train:**
	• Train A:
	  • EG HIS-21 (CCW Pump A)
	  • EG HIS-23 (CCW Pump C)
	• Train B:
	  • EG HIS-22 (CCW Pump B)
	  • EG HIS-24 (CCW Pump D)
		PERFORM the following:
		a. ENSURE at least one CCW pump running in each operating train.
		b. If unable to start a pump in the train supplying the Service Loop, THEN PERFORM the following:
		1) TRANSFER the Service Loop to alternate Train using one of the following:
		• Attachment C, Transferring Service Loop From Train A to Train B
		• Attachment D, Transferring Service Loop From Train B to Train A
		2) Do NOT proceed until service loop has been transferred.

2.	**CHECK CCW Flow - REDUCED OR LOST** • EG FI-55A (Radwaste & Containment) • EG FI-128 (Containment) • EG FI-129 (Containment)	Go To Step 8 for indication of CCW leak.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**RECORD Time CCW Lost To The RCPs:** Time _______________	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

4. **VERIFY At Least ONE Method of RCP Seal Cooling To All RCPs In Progress**

- Seal Injection

  OR

- CCW to Thermal Barrier Heat Exchanger

RESPONSE NOT OBTAINED:

RESTORE ONE method of RCP Seal Cooling to all RCPs WITHIN 6 minutes OR PERFORM the following while continuing with this procedure:

a. TRIP the Reactor

b. TRIP All Affected RCPs

c. PERFORM E-0, Reactor Trip or Safety Injection

d. CONTINUE actions of this procedure:

1) IF A or B RCP is secured, THEN CLOSE the Pressurizer Spray Valve for the affected RCP

   - BB PK-455B (A RCP)
   - BB PK-455C (B RCP)

2) PLACE Steam Dumps in Steam Pressure Mode.

   a) PLACE Steam Dump Select Switch in STM PRESS position:

      - AB US-500Z

   b) PLACE Steam Header Pressure Controller in AUTO:

      - AB PK-507

   c) If only one RCP is affected DEFEAT Tavg and ΔT for Idle RCS Loop:

      - BB TS-412T for Tavg
      - BB TS-411F for ΔT

3) If only one RCP is affected DEFEAT Tavg and ΔT for Idle RCS Loop:

   - BB TS-412T for Tavg
   - BB TS-411F for ΔT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
# 5.	**CHECK CCW Lost to RCPs - GREATER THAN 10 MINUTES**	WHEN time since CCW was lost to RCPs is greater than 10 minutes, THEN PERFORM Steps 5.a through 5.d. CONTINUE with Step 6.
	a. TRIP the Reactor	
	b. TRIP all RCPs	
	c. PERFORM E-0, Reactor Trip Or Safety Injection	
	d. CONTINUE actions of this procedure	
	1) CLOSE the Pressurizer Spray Valve for A and B RCPs • BB PK-455B (A RCP) • BB PK-455C (B RCP)	
	2) PLACE Steam Dumps in Steam Pressure Mode.	
	a) PLACE Steam Dump Select Switch in STM PRESS position: • AB US-500Z	
	b) PLACE Steam Header Pressure Controller in AUTO: • AB PK-507	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**CHECK CCW Flow To Containment - NORMAL OR HIGH FOR PLANT CONDITIONS** • EG FI-128 • EG FI-129	PERFORM the following: a. ENSURE all CCW to Containment Inner and Outer isolation valves are open: • EG HIS-58 • EG HIS-59 • EG HIS-60 • EG HIS-71 • EG HIS-61 • EG HIS-62 b. IF any valve(s) fail to open, THEN OPEN the associated bypass valve using Attachment E, CCW Containment Isolation Valves. c. ENSURE CCW Flow To Containment is restored.
7.	**CHECK CCW To RW & RCS Flow - NORMAL OR HIGH FOR PLANT CONDITIONS** • EG FI-55A	ENSURE the Radwaste Building Supply and Return Headers are open: • EG HS-69 • EG HS-70 IF valves can NOT be opened, THEN BALANCE CCW flow using OTN-EG-00001, Component Cooling Water System.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	**CHECK CCW Surge Tank Level(s) - LOWERING** • EG LI-1 (Tank A) • EG LI-2 (Tank B)	PERFORM the following: a. IF level is stable, THEN Go To Step 13. b. IF CCW Surge Tank A is rising AND makeup is NOT required, THEN PERFORM the following: 1) ENSURE EGLV0001, DI Water To CCW Surge Tank A is closed: • EG HIS-1 2) IF EGLV0001 does not close, THEN locally CLOSE EGV0145, DI Water to CCW Surge Tank EGLV0001 Upstream Isolation. c. IF CCW Surge Tank B is rising AND makeup is NOT required, THEN PERFORM the following: 1) ENSURE EGLV0002, DI Water To CCW Surge Tank B is closed: • EG HIS-2 2) IF EGLV0002 does not close, THEN locally CLOSE EGV0148, DI Water To CCW Surge Tank B EGLV0002 Upstream Isolation. d. IF level continues to rise, THEN Go To OTO-BB-00003, RCS Excessive Leakage.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. CHECK CCW Surge Tank Level - GREATER THAN 44%

- EG LI-1 (Tank A)
- EG LI-2 (Tank B)

PERFORM the following:

a. IF CCW Surge Tank A is low,
THEN PERFORM the following:

1) ENSURE EGLV0001, DI Water To CCW Surge Tank A is open.

   - EG HIS-1

2) IF EGLV0001 does not open,
THEN locally OPEN EGV0147, DI Water To CCW Surge Tank A EGLV0001 Bypass Isolation.

b. IF CCW Surge Tank B is low,
THEN PERFORM the following:

1) ENSURE EGLV0002, DI Water To CCW Surge Tank B is open.

   - EG HIS-2

2) IF EGLV0002 does not open,
THEN locally OPEN EGV0150, DI Water To CCW Surge Tank B EGLV0002 Bypass Isolation.

c. IF CCW Surge Tank is still lowering AND an emergency conditions exist,
THEN ALIGN ESW Makeup to CCW for the applicable train:

- EG HIS-11/EG HIS-13 (Train A)
- EG HIS-12/EG HIS-14 (Train B)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
# 10.	**CHECK CCW Surge Tank Level - GREATER THAN 10% IN TRAIN SUPPLYING SERVICE LOOP** • EG LI-1 (Tank A) • EG LI-2 (Tank B)	PERFORM the following: a. Manually TRIP the Reactor. b. TRIP all RCPs. c. PERFORM E-0, Reactor Trip Or Safety Injection. d. CONTINUE actions of this procedure using one of the following: • Attachment A, CCW Train A Leak • Attachment B, CCW Train B Leak
11.	**DIRECT Operators To Walkdown CCW To Determine Source Of Leakage**	
12.	**Go To The Following Attachment As Appropriate:** • Attachment A, CCW Train A Leak • Attachment B, CCW Train B Leak	
13.	**REVIEW Technical Specifications 3.6.3 and 3.7.7**	
14.	**PERFORM Notifications Per ODP-ZZ-00001 Addendum 13, Shift Manager Communications**	
15.	**Go To Appropriate Plant Procedure As Directed By The Shift/Control Room Supervisor**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 1 of 7)
CCW Train A Leak

NOTES

- When the leak is isolated, it is no longer necessary to continue with the remainder of this attachment.
- If the location of the CCW leak is known, the CRS may go to step that isolates that component.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A1.	**CHECK Location Of The Leak - KNOWN** • Radwaste - Step A3 • RCDT Hx - Step A9 • Seal Water Hx - Step A10 • Service Loop - Step A11 • Train A Safety Loop - Step A13	Go To Step A2 of this Attachment.
A2.	**CHECK Service Loop Is Being Supplied From Train A:** • EG ZL-53 - OPEN • EG ZL-15 - OPEN	Go To Step A13 of this Attachment.
A3.	**ISOLATE The Radwaste Building Supply And Return Headers:** • EG HS-69 • EG HS-70	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 2 of 7)
CCW Train A Leak

A4. CHECK For Indications That Leak - STILL PRESENT

PERFORM the following:

a. DISPATCH Radwaste Operator to walkdown system to determine leak location:

- Catalytic Hydrogen Analyzer
- Waste Gas Compressors
- Aux Steam Rad Monitor
- Waste Evaporator
- Recycle Evaporator
- Secondary Waste Evaporator
- Reverse Osmosis Unit

b. DISPATCH Equipment Operator to Nuclear Sample Coolers (SJ coolers) to determine if leak exist.

c. WHEN the source of the leak has been identified, THEN PERFORM the following:

1) CLOSE isolation valves immediately upstream and downstream of leak.

2) Go To Step 13 of the procedure.

A5. RESTORE Lineup For Radwaste Building, As Time Permits

A6. TRANSFER The Service Loop To Train B Using Attachment C, Transferring Service Loop From Train A to Train B

A7. Do NOT Proceed Until Service Loop Has Been Transferred

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 3 of 7)
CCW Train A Leak

NOTE

If the leak is no longer present in Train A, it may have been transferred to Train B.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A8.	**CHECK For Indications That Leak Has Been Transferred To Train B Service Loop:** • EG LI-2 (Tank B)	PERFORM the following: a. DISPATCH Equipment Operator to walkdown system to determine leak location. b. WHEN the source of the leak has been identified, THEN CLOSE isolation valves immediately upstream and downstream of leak. c. Go To Step A13 of this Attachment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 4 of 7)
CCW Train A Leak

A9. ISOLATE The RCDT Heat Exchanger By Performing The Following:

a. CLOSE the PRT To RCDT Valve:

- BB HIS-8031

b. CLOSE RCDT HX Outlet Outer Containment Isolation Valve:

- HB HIS-7136

c. DIRECT Radwaste Operator to place RCDT Transfer Pumps in PULL TO LOCK:

- HB HS/1003A
- HB HS/1003B

d. DIRECT Radwaste Operator to close the following valves:

- RCDT Outlet Header HBHV7127 Hand Switch:
  - HBHS/1003C
- RCDT Pumps Discharge Hx Recirc Hand Control Valve Hand Switch:
  - HBHS/1003F
- RCS RCDT Hx To PRT BBHV7141 Hand Switch:
  - BBHS/1003D

e. Do NOT Proceed Until RCDT Heat Exchanger is isolated

f. CHECK For Indications That Leak - STILL PRESENT

RESPONSE NOT OBTAINED: f. Go To Step 13 of the procedure.

g. RESTORE Lineup For RCDT Heat Exchanger, As Time Permits

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 5 of 7)
CCW Train A Leak

A10. BYPASS And ISOLATE CCW To The Seal Water Heat Exchanger:

a. DISPATCH EO to perform the following:

1) OPEN BG8400, CVCS Seal Water HX Bypass Valve

2) CLOSE BG-8398A, CVCS Seal Water HX Inlet Isolation

3) CLOSE BG-8398B, CVCS Seal Water HX Outlet Upstream Isolation

4) CLOSE BGV0206, CVCS Seal Water HX Outlet CCW Return Isolation

5) CLOSE EGV0085, Seal Water HX CCW Inlet Isolation

6) Do NOT Proceed Until Seal Water Heat Exchanger is isolated

b. CHECK For Indications That Leak - STILL PRESENT

RESPONSE NOT OBTAINED: b. Go To Step 13 of the procedure.

c. RESTORE Lineup For CCW To The Seal Water Heat Exchanger

A11. ISOLATE The CCW Service Loop By Closing The CCW Train Supply/Return Valves:

- EG HS-15
- EG HS-16

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 6 of 7)
CCW Train A Leak

A12. PERFORM All Of The Following:

a. Manually TRIP the Reactor

b. TRIP all RCPs

c. PERFORM E-0, Reactor Trip Or Safety Injection

A13. TRANSFER CCW Train A Safety Loop Loads To CCW Train B As Required:

- CCP Oil cooler
- SI Pump Oil cooler
- Fuel Pool Hx
- RHR Hx
- RHR Pump

A14. ISOLATE CCW Train A By Performing The Following:

a. CHECK charging flow is supplied from NCP or CCP B:

- BG HIS-3
- BG HIS-2A

RESPONSE NOT OBTAINED: a. PERFORM the following:

1) START the NCP or CCP B:

- BG HIS-3
- BG HIS-2A

2) STOP CCP A:

- BG HIS-1A

b. STOP CCW pump(s) on leaking CCW train AND PLACE in PTL:

- EG HIS-21 (CCW Pump A)
- EG HIS-23 (CCW Pump C)

c. CLOSE isolation valves immediately upstream and downstream of leak

d. CLOSE EGV0145, DI Water To CCW Surge Tank A EGLV0001 Upstream Isolation

e. CHECK For Indications That Leak - STILL PRESENT

RESPONSE NOT OBTAINED: e. Go To Step 13 of the procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 7 of 7)
CCW Train A Leak

A15. Go To Step 13 Of The Procedure

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 1 of 7)
CCW Train B Leak

NOTES

- When the leak is isolated, it is no longer necessary to continue with the remainder of this attachment.
- If the location of the CCW leak is known, the CRS may go to step that isolates that component.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B1.	**CHECK Location Of The Leak - KNOWN** • Radwaste - Step B3 • RCDT Hx - Step B9 • Seal Water Hx - Step B10 • Service Loop - Step B11 • Train B Safety Loop - Step B13	Go To Step B2 of this Attachment.
B2.	**CHECK Service Loop Is Being Supplied From Train B:** • EG ZL-54 - OPEN • EG ZL-16 - OPEN	Go To Step B13 of this Attachment.
B3.	**ISOLATE The Radwaste Building Supply And Return Headers:** • EG HS-69 • EG HS-70	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 2 of 7)
CCW Train B Leak

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B4. CHECK For Indications That Leak - STILL PRESENT	PERFORM the following: a. DISPATCH Radwaste Operator to walkdown system to determine leak location: • Catalytic Hydrogen Analyzer • Waste Gas Compressors • Aux Steam Rad Monitor • Waste Evaporator • Recycle Evaporator • Secondary Waste Evaporator • Reverse Osmosis Unit b. DISPATCH Equipment Operator to Nuclear Sample Coolers (SJ coolers) to determine if leak exist. c. WHEN the source of the leak has been identified, THEN PERFORM the following: 1) CLOSE isolation valves immediately upstream and downstream of leak. 2) Go To Step 13 of the procedure.
B5. RESTORE Lineup For Radwaste Building, As Time Permits	
B6. TRANSFER The Service Loop To Train A Using Attachment D, Transferring Service Loop From Train B to Train A	
B7. Do NOT Proceed Until Service Loop Has Been Transferred	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 3 of 7)
CCW Train B Leak

NOTE

If the leak is no longer present in Train B, it may have been transferred to Train A.

B8.	**CHECK For Indications That Leak Has Been Transferred To Train A Service Loop:** • EG LI-1 (Tank A)	PERFORM the following: a. DISPATCH Equipment Operator to walkdown system to determine leak location. b. WHEN the source of the leak has been identified, THEN CLOSE isolation valves immediately upstream and downstream of leak. c. Go To Step B13 of this Attachment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 4 of 7)
CCW Train B Leak

B9. ISOLATE The RCDT Heat Exchanger By Performing The Following:

a. CLOSE the PRT To RCDT Valve:

- BB HIS-8031

b. CLOSE RCDT HX Outlet Outer Containment Isolation Valve:

- HB HIS-7136

c. DIRECT Radwaste Operator to place RCDT Transfer Pumps in PULL TO LOCK:

- HB HS/1003A
- HB HS/1003B

d. DIRECT Radwaste Operator to close the following valves:

- RCDT Outlet Header HBHV7127 Hand Switch:
  - HBHS/1003C
- RCDT Pumps Discharge Hx Recirc Hand Control Valve Hand Switch:
  - HBHS/1003F
- RCS RCDT Hx To PRT BBHV7141 Hand Switch:
  - BBHS/1003D

e. Do NOT Proceed Until RCDT Heat Exchanger is isolated

f. CHECK For Indications That Leak - STILL PRESENT

RESPONSE NOT OBTAINED: f. Go To Step 13 of the procedure.

g. RESTORE Lineup For RCDT Heat Exchanger, As Time Permits

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 5 of 7)
CCW Train B Leak

B10. BYPASS And ISOLATE CCW To The Seal Water Heat Exchanger:

a. DISPATCH EO to perform the following:

1) OPEN BG8400, CVCS Seal Water HX Bypass Valve

2) CLOSE BG-8398A, CVCS Seal Water HX Inlet Isolation

3) CLOSE BG-8398B, CVCS Seal Water HX Outlet Upstream Isolation

4) CLOSE BGV0206, CVCS Seal Water HX Outlet CCW Return Isolation

5) CLOSE EGV0085, Seal Water HX CCW Inlet Isolation

6) Do NOT Proceed Until Seal Water Heat Exchanger is isolated

b. CHECK For Indications That Leak - STILL PRESENT

RESPONSE NOT OBTAINED: b. Go To Step 13 of the procedure.

c. RESTORE Lineup For CCW To The Seal Water Heat Exchanger

B11. ISOLATE The CCW Service Loop By Closing The CCW Train Supply/Return Valves:

- EG HS-15
- EG HS-16

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 6 of 7)
CCW Train B Leak

B12. PERFORM All Of The Following:

a. Manually TRIP the Reactor

b. TRIP all RCPs

c. PERFORM E-0, Reactor Trip Or Safety Injection

B13. TRANSFER CCW Train B Safety Loop Loads To CCW Train A As Required:

- CCP Oil cooler
- SI Pump Oil cooler
- Fuel Pool Hx
- RHR Hx
- RHR Pump

B14. ISOLATE CCW Train B By Performing The Following:

a. CHECK charging flow is supplied from NCP or CCP A:

- BG HIS-3
- BG HIS-1A

RESPONSE NOT OBTAINED:

a. PERFORM the following:

1) START the NCP or CCP A:

- BG HIS-3
- BG HIS-1A

2) STOP CCP B:

- BG HIS-2A

b. STOP CCW pump(s) on leaking CCW train AND PLACE in PTL:

- EG HIS-22 (CCW Pump B)
- EG HIS-24 (CCW Pump D)

c. CLOSE isolation valves immediately upstream and downstream of leak

d. CLOSE EGV0148, DI Water To CCW Surge Tank B EGLV0002 Upstream Isolation

e. CHECK For Indications That Leak - STILL PRESENT

RESPONSE NOT OBTAINED:

e. Go To Step 13 of the procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 7 of 7)
CCW Train B Leak

B15. Go To Step 13 Of The Procedure

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C
(Page 1 of 2)
Transferring Service Loop From Train A to Train B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C1.	**CHECK If CCW Aligned To The RCPs**	PERFORM the following: a. RECORD Time CCW lost to the RCPs: Time _______________ b. IF CCW is Lost to RCPs for greater than 10 minutes, THEN PERFORM the following: 1) TRIP the Reactor. 2) TRIP all RCPs. 3) PERFORM E-0, Reactor Trip Or Safety Injection. 4) CONTINUE actions of this procedure.

C2.	**CHECK ESW Cooling Water To The CCW Heat Exchanger - ALIGNED FOR CURRENT PLANT CONDITIONS**
	
	• Train A:
	
	  • EF HIS-51
	  • EF HIS-59
	
	• Train B:
	
	  • EF HIS-52
	  • EF HIS-60
		ALIGN ESW cooling water to CCW heat exchanger as required.

C3.	**CHECK CCW Pump B or D - RUNNING** • EG HIS-22 (CCW Pump B) • EG HIS-24 (CCW Pump D)	PERFORM the following: a. ENSURE CCW Surge Tank B level is greater than 50%. b. START CCW Pump B or D.
C4.	**CLOSE Both CCW Surge Tank Vent Control Valves:** • EG HIS-9 • EG HIS-10	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C
(Page 2 of 2)
Transferring Service Loop From Train A to Train B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C5.	**OPEN CCW Train B Supply/Return Valves:** • EG HS-16	
C6.	**CLOSE CCW Train A Supply/Return Valves:** • EG HS-15	
C7.	**OPEN Both CCW Surge Tank Vent Control Valves:** • EG HIS-9 • EG HIS-10	
C8.	**CHECK CCW Pump A Or C Required For The Operating Safety Loop Loads:** • CCP Oil cooler • SI Pump Oil cooler • Fuel Pool Hx • RHR Hx • RHR Pump	PERFORM the following: a. STOP CCW Pump A or C: • EG HIS-21 (CCW Pump A) • EG HIS-23 (CCW Pump C) b. PLACE affected handswitch in AUTO.
C9.	**CHECK Both Trains Of CCW Remain - IN SERVICE**	PERFORM the following: a. TRANSFER Train A Safety Loop loads to Train B, as required. b. ENSURE all Safety Loads are being supplied by the operating train, as required. c. Go To Step C11 of this attachment.
C10.	**CHECK CCW To RHR Heat Exchanger A Isolation - OPEN** • EG HIS-101	OPEN CCW To RHR Heat Exchanger A Isolation.
C11.	**NOTIFY Shift Chemistry Technician That CCW Train B Is In Service**	

-END-

STEP / ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D
(Page 1 of 2)
Transferring Service Loop From Train B to Train A

STEP / ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
D1. CHECK If CCW Aligned To The RCPs	PERFORM the following: a. RECORD Time CCW lost to the RCPs: Time _______________ b. IF CCW is Lost to RCPs for greater than 10 minutes, THEN PERFORM the following: 1) TRIP the Reactor. 2) TRIP all RCPs. 3) PERFORM E-0, Reactor Trip Or Safety Injection. 4) CONTINUE actions of this procedure.

D2. CHECK ESW Cooling Water To The CCW Heat Exchanger - ALIGNED FOR CURRENT PLANT CONDITIONS
 	
• Train A:
  • EF HIS-51
  • EF HIS-59
 	
• Train B:
  • EF HIS-52
  • EF HIS-60
	ALIGN ESW cooling water to CCW heat exchanger as required.

D3. CHECK CCW Pump A or C - RUNNING • EG HIS-21 (CCW Pump A) • EG HIS-23 (CCW Pump C)	PERFORM the following: a. ENSURE CCW Surge Tank A level is greater than 50%. b. START CCW Pump A or C.
D4. CLOSE Both CCW Surge Tank Vent Control Valves: • EG HIS-9 • EG HIS-10	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT D
(Page 2 of 2)
Transferring Service Loop From Train B to Train A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
D5.	**OPEN CCW Train A Supply/Return Valves:** • EG HS-15	
D6.	**CLOSE CCW Train B Supply/Return Valves:** • EG HS-16	
D7.	**OPEN Both CCW Surge Tank Vent Control Valves:** • EG HIS-9 • EG HIS-10	
D8.	**CHECK CCW Pump B Or D Required For The Operating Safety Loop Loads:** • CCP Oil cooler • SI Pump Oil cooler • Fuel Pool Hx • RHR Hx • RHR Pump	PERFORM the following: a. STOP CCW Pump B or D: • EG HIS-22 (CCW Pump B) • EG HIS-24 (CCW Pump D) b. PLACE affected handswitch in AUTO.
D9.	**CHECK Both Trains Of CCW Remain - IN SERVICE**	PERFORM the following: a. TRANSFER Train B Safety Loop loads to Train A, as required. b. ENSURE all Safety Loads are being supplied by the operating train, as required. c. Go To Step D11 of this attachment.
D10.	**CHECK CCW To RHR Heat Exchanger B Isolation - OPEN** • EG HIS-102	OPEN CCW To RHR Heat Exchanger B Isolation.
D11.	**NOTIFY Shift Chemistry Technician That CCW Train A Is In Service**	

-END-

ATTACHMENT E
(Page 1 of 3)
CCW Containment Isolation Valves

E1. PLACE Administrative Controls for any OPEN Containment Isolation CCW Bypass Valve:

- Dedicated operators must be briefed and able to CLOSE the open Ctmt Iso CCW Bypass Valve upon receipt of a valid CIS 'B' Signal.
- A dedicated Control Room operator able to CLOSE the open Ctmt Iso CCW Bypass Valve or notify the local dedicated operator.
- A local dedicated operator able to CLOSE the OPEN Ctmt Iso CCW Bypass Valve.
- The local dedicated operator is in communication with the Control Room.
- The local dedicated operator is stationed near (in a low dose area if possible), the OPEN Ctmt Iso CCW Bypass Valve.

ATTACHMENT E
(Page 2 of 3)
CCW Containment Isolation Valves

E2. Use The Tables Below For Additional Containment Isolation CCW Valve Information

- EGHV0058 (EG HIS-58) [PEN 74] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0127	EG HIS-127	EG HIS-127A	MCB and Local at EGHV0127

- EGHV0059 (EG HIS-59) [PEN 75] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0131	EG HIS-131	EG HIS-131A	MCB and Local at EGHV0131

- EGHV0060 (EG HIS-60) [PEN 75] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0130	EG HIS-130	EG HIS-130A	MCB and Local at EGHV0059

If inner containment valve EGHV0060 has failed CLOSED, allowing the CCW flowpath through EGHV0130 and EGHV0059, station Operator near EGHV0059 to isolate the flowpath. Both valves are powered from Separation Group 1.

(Step 2. continued on next page)

ATTACHMENT E
(Page 3 of 3)
CCW Containment Isolation Valves

Step 2. (continued from previous page)

- EGHV0061 (EG HIS-61) [PEN 76] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0133	EG HIS-133	EG HIS-133A	MCB and Local at EGHV0133

- EGHV0062 (EG HIS-62) [PEN 76] {CISB and EGFSH0062}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0132	EG HIS-132	EG HIS-132A	MCB and Local at EGHV0061

If inner containment valve EGHV0062 has failed CLOSED, allowing the CCW flowpath through EGHV0132 and EGHV0061, station Local operator near EGHV0061 to isolate the flowpath. Both valves are power from Separation Group 1.

With EGHV0132 OPEN, the dedicated Control Room Operator is required to CLOSE EGHV0132 and ENSURE RCP thermal barrier isolations BB HIS-13, BB HIS-14, BB HIS-15 and BB HIS-16 are CLOSED during a high flow condition, as indicated by MCB Annunciator 74C, RCP THERM BAR CCW FLOW.

- EGHV0071 (EG HIS-71) [PEN 74] {CISB}:

Bypass	Switch	Bypass Iso/ Non Iso Switch	If Bypass Open, Operators At
EGHV0126	EG HIS-126	EG HIS-126A	MCB and Local at EGHV0126

-END-

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
Steam Gen. Tube Rupture	**Group #**	1		
	K/A #	00038 G2.4.41		
	Importance Rating	4.6		
Knowledge of the emergency action level thresholds and classifications.				

Question # 78

(REFERENCE PROVIDED)

Given the following plant conditions:

- 0800 the crew identifies a SG tube leak on “A” SG
- 0810 the crew maximizes charging
- 0812 the crew isolates letdown
- 0817 the crew trips the reactor and inserts a manual safety injection (SI)
- 0823 a unisolable leak from “A” SG develops in Area 5
- 0826 GT-RE-21B reads 5E+6 µCi/sec

What is the HIGHEST Emergency Plan Action Level that applies?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Explanation: *With the conditions given a loss of the Containment barrier has occurred (a leaking or RUPTURED SG is FAULTED outside of Containment) and a loss of the RCS boundary has occurred (an automatic or manual ECCS (SI) actuation required by either:…. or SG tube RUPTURE). There is no loss or potential loss of the Fuel cladding. The EAL would be a FS1.1.*

A. Incorrect, Plausible in the operator incorrectly could apply the requirements of EAL SU5.1 for RCS boundary leakage. This EAL is not the HIGHEST EAL level that is applicable
B. Incorrect, Plausible if the operator does not recognize the loss or potential loss of a second barrier or the operator only applies the GT-RE-21B read that is higher than the Alert level reading but lower than the site area emergency.
C. Correct, See above

D. Incorrect, plausible if the operator incorrectly applies Table R1 or applies the GT RE 21B reading to the containment radiation monitors (GT-RE-59 & 60) which would be a loss of the Fuel Clad Barrier for a total of three losses.

Technical Reference(s):

1. EIP-ZZ-00101, Addendum 1 EAL Classification Matrix, Rev 5

References to be provided to applicants during examination:

1. EIP-ZZ-00101, Addendum 1 EAL Classification Matrix, Rev 5

Learning Objective: Lesson T68.1020.6 (.8), Obj B, Determine the emergency classification for given indications and/or symptoms, per EIP-ZZ-00101.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ____NA________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

SRO Only due to:

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES**

• Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
• Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

F Fission Product Barrier Degradation	FG1.1 1 2 3 4	FS1.1 1 2 3 4	FA1.1 1 2 3 4
	Loss of **any** two barriers. **AND** Loss or potential loss of third barrier (Table F-1).	Loss or potential loss of **any** two barriers (Table F-1).	**Any** loss or **any** potential loss of either Fuel Clad or RCS (Table F-1).

Table F-1 Fission Product Barrier Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CMT) Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A RCS or SG Tube Leakage			1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage. • SG tube RUPTURE.	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage. • SG tube leakage. 2. CSFST Integrity-**RED** Path conditions met.	1. A leaking or RUPTURED SG is FAULTED outside of containment.	
B Inadequate Heat Removal	1. CSFST Core Cooling-**RED** Path conditions met.	1. CSFST Core Cooling-**ORANGE** Path conditions met. 2. CSFST Heat Sink-**RED** Path conditions met. **AND** Heat sink required.		1. CSFST Heat Sink-**RED** Path conditions met. **AND** Heat sink required.		1. CSFST Core Cooling-**RED** Path conditions met. **AND** Restoration procedures not effective within 15 min. (Note1)
C CMT Radiation / RCS Activity	1. Containment radiation > 2.80E+03 R/hr on GT-RE-59 (591) or GT-RE-60 (601). 2. Dose equivalent I-131 coolant activity > 300 µCi/cc. 3. CVCS letdown radiation > 2.50E+01 µCi/ml on SJ-RE-01 (016).		1. Containment radiation > 6.40E+00 R/hr on GT-RE-59 (591) or GT-RE-60 (601).			1. Containment radiation > 8.06E+04 R/hr on GT-RE-59 (591) or GT-RE-60 (601).
D CMT Integrity or Bypass					1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment. • UNISOLABLE pathway from containment to the environment exists. 2. Indications of RCS leakage outside of containment.	1. CSFST Containment-**RED** Path conditions met. 2. Containment hydrogen concentration ≥ 4%. 3. Containment pressure > 27 psig with < one full train of Containment depressurization equipment operating per design for ≥ 15 min. (Note 1, 9)
E Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier.	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier.	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier.	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier.	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier.	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
Small Break LOCA / 3	Group #	1		
	K/A #	009EA2.36		
	Importance Rating	4.6		
Ability to determine or interpret the following as they apply to a small break LOCA: Difference between overcooling and LOCA indications.				

Question # 79

A small LOCA occurred

- During the performance of E-0, Reactor Trip or Safety Injection The Control Room staff identified a fault in ‘A’ Steam Generator outside of containment.
- The crew responded by isolating the fault per E-2, Faulted Steam Generator Isolation.
- The crew transitioned to E-1, Loss of Reactor or Secondary Coolant.

The following conditions exist:

- ‘A’ Steam Generator pressure - 100 psig and lowering
- ‘B’, ‘C’ and ‘D’ Steam Generator Pressure - 680 psig and slowly lowering
- RCS pressure is 960 psig and stable
- RCS temperature is 500 °F and slowly lowering
- Intact Steam Generator level is 10% narrow range and rising
- Total Aux Feedwater Flow is 175,000 Lbm/Hr
- Pressurizer Level is 10% and rising

The crew should transition to…

A. ES-1.1, SI Termination

B. ES-1.2, Post LOCA Cooldown And Depressurization

C. ES-1.4, Transfer To Hot Leg Recirculation after 13 hours have elapsed since event initiation

D. ECA-2.1, Uncontrolled Depressurization Of All Steam Generators

Answer: A

Explanation:

E-1 Step 6CHECK If ECCS Flow Should Be Reduced:
RCS subcooling - GREATER THAN 30°F [50°F] – YES
Secondary heat sink:

- *Narrow range level in at least one intact SG - GREATER THAN 7% [25%] – YES*

RCS pressure - STABLE OR RISING – YES
PZR level - GREATER THAN 9% [29%] – YES

If the answer to all of these questions in Step 6 (a continuous action step) is YES the transition should be to ES-1.1, SI Termination

A. Correct, Per above,
B. Incorrect, per above this transition occurs in Step 13. Plausible if the applicant does not correctly remember the questions in step six and goes to step 7 (then eventually 13) prior to transitioning to ES-1.1
C. Incorrect, per above. Plausible as this is the transition that would occur for a larger break LOCA
D. Incorrect, per above. Plausible if the candidate incorrectly applies the information from the stem to Step 2 of E-1 and believes that the SGs are lowering in an uncontrolled manner

Technical Reference(s):

1. E-1, Loss of Reactor or Secondary Coolant, Rev 17

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, D-08 E-1 LOSS OF REACTOR OR SECONDARY COOLANT Objective J OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of E-1

Question Source: Bank # __L16017____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

Replaced question and k/a per NRC Comments.

The question meets the KA by making the candidate differentiate between if a small LOCA is occurring or if overcooling is still occurring with and isolable SG.

SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**
• Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures **YES**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
# 6.	**CHECK If ECCS Flow Should Be Reduced:**	
	a. RCS subcooling — GREATER THAN 30°F [50°F]	a. Go To Step 7.
	b. Secondary heat sink: • Narrow range level in at least one intact SG — GREATER THAN 7% [25%] OR • Total feed flow to intact SGs — GREATER THAN 285,000 LBM/HR	b. Go To Step 7.
	c. RCS pressure — STABLE OR RISING	c. Go To Step 7.
	d. PZR level — GREATER THAN 9% [29%]	d. PERFORM the following: 1) TRY to stabilize RCS pressure with normal PZR spray. 2) Go To Step 7.
	e. Go To ES—1.1, SI Termination, Step 1	

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
Loss of DC Power	**Group #**	1		
	K/A #	00058 G2.2.44		
	Importance Rating	4.4		
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.				

Question # 80

Reactor Power is 100% when:

- Annunciator 25B, NN11 INV TRBL/XFR, alarms.
- Annunciator 25C, NK01 TROUBLE, alarms.
- NK EI-I, 125V DC BUS NK01 VOLT, indicates 0 volts.

The following parameters are observed:

- Containment Pressure is 0 psig and steady.
- Steam Line Header Pressure is 1115 psig and slowly rising.
- PZR Pressure lowered to 2000 psig and is slowly returning to NOP.

(1) What is the status of the A train of EFSAS?

And

(2) What will the CRS direct to verify plant response?

A. (1) SA066X indications will be white
(2) Attachment A of E-0, Automatic Action Verification

B. (1) SA066X indications will be white
(2) ES-0.1, Reactor Trip Response

C. (1) SA066X indications will be red
(2) Attachment A of E-0, Automatic Action Verification

D. (1) SA066X indications will be red
(2) ES-0.1, Reactor Trip Response

Answer: D

Explanation:

Note: Per OTO-NK-00002, attachment A, the MSIVs will slowly drift close along with other FWIVs drifting which will result in a plant trip and entry into E-0 due to the initial mode. But with no control power and per the note prior to step #1 of Attachment A, A Train EFSAS will not automatically realign which will show as a red indication on the SA066X panel (red = NOT in its safety function position)

With the conditions given SA066X indications will be red and no SI will have occurred. NO SI has occurred so the operator is kicked out of E-0 at step 4 RNO which directs you to ES-0.1. This is prior to Step 5 which directs Attachment A.

A. Incorrect, Plausible if the operator incorrectly assumes that these indications will NOT have an effect on SA066X and does not correctly transition out of E-0 prior to step 5.
B. Incorrect, Plausible if the operator incorrectly assumes that these indications will NOT have an effect on SA066X. The second part is correct
C. Incorrect, The first part is correct for the indications given. The second part is plausible if the operator does not correctly transition out of E-0 prior to step 5.
D. Correct, correct indication and procedure attachment to enter for conditions given

Technical Reference(s):
1. E-0, Reactor Trip or Safety Injection, Rev 16
2. OTO-NK-00002, Loss of Vital 125 VDC Bus, Rev 14

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #-46, OTO-SA-00001, ENGINEERED SAFETY FEATURE ACTUATION VERIFICATION AND RESTORATION OBJ C, Given a set of plant conditions or parameters indicating an Engineered Safety Feature Actuation Verification and Restoration is required, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ___X R17671___
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Comments:

k/a match as the operator is given a set of control room indications conditions (indicating a loss of DC) and must interpret those to verify the status of a train of EFSAS. Additionally the operator is given another set of plant conditions, (indicating that no SI conditions exist) and understand the plant response and provide direction of which procedure to perform in this situation thereby affecting plant and system conditions.

SRO Justification

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **NO**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES**
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures **NO**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures **NO**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Steps 1 through 4 are immediate action steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**CHECK Reactor Trip:** • Rod Bottom Lights — ALL LIT • Reactor Trip and Bypass Breakers — OPEN • Neutron Flux — LOWERING	Manually TRIP Reactor. IF Reactor Power is greater than or equal to 5% OR Intermediate Range SUR is positive, THEN Go To FR—S.1, Response To Nuclear Power Generation/ATWS, Step 1.
2.	**CHECK Turbine Trip:** a. All Turbine Stop valves — CLOSED	a. Manually TRIP Turbine. IF Turbine will NOT trip, THEN FAST CLOSE all MSIVs and Bypass valves: • AB HS—79 • AB HS—80

FOLDOUT PAGE FOR E—0

1. **RCP TRIP CRITERIA**
   IF BOTH conditions listed below occur,
   THEN TRIP all RCPs:
   - CCPs or SI Pumps — AT LEAST ONE RUNNING
     AND
   - RCS pressure — LESS THAN 1425 PSIG

2. **FAULTED SG ISOLATION CRITERIA**
   IF any SG pressure is lowering in an uncontrolled manner OR is completely depressurized,
   THEN PERFORM the following as desired:
   - FAST CLOSE MSIVs.
   - Manually CLOSE or locally ISOLATE any failed open ASD(s).
   - ISOLATE feed flow to faulted SG(s).
   - MAINTAIN total feed flow greater than 285,000 lbm/Hr until narrow range level is greater than 7% [25%] in at least one SG.

3. **RUPTURED SG ISOLATION CRITERIA**
   IF BOTH conditions listed below occur,
   THEN ISOLATE feed flow to affected SG(s) as desired:
   - Level in any SG rises in an uncontrolled manner OR any SG has abnormal radiation.
     AND
   - Narrow range level in affected SG(s) — GREATER THAN 7% [25%].

4. **COLD LEG RECIRCULATION CRITERIA**
   IF RWST level lowers to less than 36%,
   THEN Go To ES—1.3, Transfer To Cold Leg Recirculation, Step 1.

5. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

6. **SPENT FUEL POOL COOLING**
   IF SFP Cooling pumps have tripped,
   THEN monitor SFP level and temperature and implement the following as resources permit:
   - OTO—EC—00001, Loss of SFP/Refuel Pool Level
   - OTO—EC—00002, Spent Fuel Pool High Temperature

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**CHECK Power To AC Emergency Buses:**	
	a. AC emergency buses — AT LEAST ONE ENERGIZED • NB01 <u>OR</u> • NB02	a. Perform the following: 1) Depress START/RESET pushbutton for any stopped Diesel Generator: • KJ HS-8A • KJ HS-108A 2) IF DG started AND output breaker did NOT close, THEN CLOSE DG output breaker: • NE HS-25 • NE HS-26 3) IF neither AC emergency bus is energized, THEN go to ECA-0.0, Loss Of All AC Power, Step 1.
	b. AC emergency buses — BOTH ENERGIZED	b. TRY to restore power to deenergized AC emergency bus as time permits: 1) Depress START/RESET pushbutton for any stopped Diesel Generator: • KJ HS-8A • KJ HS-108A 2) If DG started AND output breaker did NOT close, THEN close DG output breaker: • NE HS-25 • NE HS-26

FOLDOUT PAGE FOR E—0

1. **RCP TRIP CRITERIA**
   IF BOTH conditions listed below occur,
   THEN TRIP all RCPs:
   - CCPs or SI Pumps — AT LEAST ONE RUNNING
     AND
   - RCS pressure — LESS THAN 1425 PSIG

2. **FAULTED SG ISOLATION CRITERIA**
   IF any SG pressure is lowering in an uncontrolled manner OR is completely depressurized,
   THEN PERFORM the following as desired:
   - FAST CLOSE MSIVs.
   - Manually CLOSE or locally ISOLATE any failed open ASD(s).
   - ISOLATE feed flow to faulted SG(s).
   - MAINTAIN total feed flow greater than 285,000 lbm/Hr until narrow range level is greater than 7% [25%] in at least one SG.

3. **RUPTURED SG ISOLATION CRITERIA**
   IF BOTH conditions listed below occur,
   THEN ISOLATE feed flow to affected SG(s) as desired:
   - Level in any SG rises in an uncontrolled manner OR any SG has abnormal radiation.
     AND
   - Narrow range level in affected SG(s) — GREATER THAN 7% [25%].

4. **COLD LEG RECIRCULATION CRITERIA**
   IF RWST level lowers to less than 36%,
   THEN Go To ES—1.3, Transfer To Cold Leg Recirculation, Step 1.

5. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

6. **SPENT FUEL POOL COOLING**
   IF SFP Cooling pumps have tripped,
   THEN monitor SFP level and temperature and implement the following as resources permit:
   - OTO—EC—00001, Loss of SFP/Refuel Pool Level
   - OTO—EC—00002, Spent Fuel Pool High Temperature

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**CHECK SI Status:**	
	a. CHECK if SI is actuated: • Any SI annunciator 88A through 88D - LIT OR • SB069 SI Actuate RED light - LIT OR • LOCA Sequencer annunciators 30A or 31A - LIT	a. CHECK if SI is required: • PZR pressure less than or equal to 1849 PSIG OR • Any SG pressure less than or equal to 615 PSIG OR • Containment pressure greater than or equal to 3.5 PSIG IF SI is required, THEN manually ACTUATE SI: • SB HS-27 • SB HS-28 IF SI is NOT required, THEN Go To ES-0.1, Reactor Trip Response, Step 1.
	b. CHECK both Trains of SI - ACTUATED • LOCA Sequencer annunciator 30A - LIT • LOCA Sequencer annunciator 31A - LIT • SB069 SI Actuate RED light - LIT SOLID (NOT blinking)	b. Manually ACTUATE SI: • SB HS-27 • SB HS-28
5.	**PERFORM Attachment A, Automatic Action Verification, While Continuing With This Procedure**	

1 ID: R17671 Points: 1.00

<QQ 17671(1410)><<Given the following plant conditions:

? Reactor Power is 100%.
? Annunciator 25B, NN11 INV TRBL/XFR, alarms.
? Annunciator 25C, NK01 TROUBLE, alarms.
? NK EI-I, 125V DC BUS NK01 VOLT indicates 0 volts.

The following parameters were observed during the transient:

- Containment Pressure remained at 0 psig.
- Steam Line Header Pressure slowly lowered to 900 psig and has stabilized.
- RCS Pressure lowered to 2000 psig and is now returning to NOP.

(1) What is the status of the A train of EFSAS?

And

(2) The CRS will direct which of the following procedures to verify proper alignment of ESF systems?>>

A. <QQ 17671(1480:0)><<(1) SA066X indications will be red
(2) E-0, Attachment A, Automatic Action Verification>>

B. <QQ 17671(1482)><<(1) SA066X indications will be red
(2) OTO-SA-00001, Attachment AH, AFAS/LSP Train A Verification>>

C. <QQ 17671(1480:1)><<(1) SA066X indications will be white
(2) E-0, Attachment A, Automatic Action Verification>>

D. <QQ 17671(1480:2)><<(1) SA066X indications will be white
(2) OTO-SA-00001, Attachment AH, AFAS/LSP Train A Verification>>

Answer: <QQ 17671 (1419) ><<B >>

Answer Explanation:

<QQ 17671(1412)><<NO SI has occurred so the operator is kicked out of E-0 prior to Step 5 which directs Attachment A.>>

Question 1 Info	
Question Type:	<QQ 17671(1401)><<Multiple Choice>>
Status:	<QQ 17671(1405)><<Active>>
Always select on test?	<QQ 17671(1406)><<No>>
Authorized for practice?	<QQ 17671(1429)><<No>>
Points:	<QQ 17671(1441)><<1.00>>
Time to Complete:	<QQ 17671(1408)><<0>>
Difficulty:	<QQ 17671(1407)><<0.00>>
System ID:	<QQ 17671(1445)><<17671>>
User-Defined ID:	<QQ 17671(1404)><<R17671>>
Cross Reference Number:	<QQ 17671(1409)><<OTO-NK-00002>>
Topic:	<QQ 17671(1400)><<R17671 SRO ONLY Loss of DC and procedure selection>>
Num Field 1:	<QQ 17671(1414)><<3.7>>
Num Field 2:	<QQ 17671(1415)><<4.2>>
Text Field:	<QQ 17671(1413)><<013 A2.05>>

Comments:	<QQ 17671(1411)><<**Technical Reference(s)**:. 1. OTO-NK-00002, Loss of Vital 125 VDC Bus, Rev 13 2. OTO-SA-00001, ESFAS Verification and Restoration, Rev 39 3. OOA-SA-C066X, Engineered Safety Feature (ESF) Status Panel SA066X Alarm Information, Rev 14 4. E-0, Reactor Trip or Safety Injection, Rev 16 **Technical Reference(s)**:. 5. OTO-NK-00002, Loss of Vital 125 VDC Bus, Rev 13 6. OTO-SA-00001, ESFAS Verification and Restoration, Rev 39 7. OOA-SA-C066X, Engineered Safety Feature (ESF) Status Panel SA066X Alarm Information, Rev 14 8. E-0, Reactor Trip or Safety Injection, Rev 16 **Technical Reference(s)**:. 9. OTO-NK-00002, Loss of Vital 125 VDC Bus, Rev 13 10. OTO-SA-00001, ESFAS Verification and Restoration, Rev 39 11. OOA-SA-C066X, Engineered Safety Feature (ESF) Status Panel SA066X Alarm Information, Rev 14 12. E-0, Reactor Trip or Safety Injection, Rev 16 OTO-NKOOA-SA-0C066X ***Per OTO-NK-00002 Attachment A Loss of power to NK01, "Loss of control power to ESFAS Cabinet SA036A results in loss of ESFAS Train A automatic and manual actuation". Furthermore, "Loss of control power to ESFAS Train A Solid State Load Sequencer Panel NF039C results in loss of Train A load shed and sequencing capability."*** *Based on the initial plant conditions and a loss of NK01, which will cause all 4 FWIVs to close, a reactor trip will be required. The crew will enter E-0 at at step 4 the crew will implement the RNO since a SI did not occur or is required. Therefore, a transition to ES-0.1 will happen before step 5, which would direct performing E-0 Attachment A, making E-0 Attachment an incorrect choice* *OTO-SA-00001 Attachment AH is correct as a TDAFW actuation occurred on low low SG levels and due to the loss of indication of A Train EFSAS (SA066X) this procedure attachment is correct to verify A Train EFSAS.* >>

Question 1 History	
Exam Appearances:	<QQ 17671(1449)><<1>>
Student Encounters:	<QQ 17671(1448)><<7>>
Answered Right:	<QQ 17671(1452)><<1>>
Answered Wrong:	<QQ 17671(1453)><<6>>
Partially Correct:	<QQ 17671(1459)><<0>>
Answer Invalid:	<QQ 17671(1455)><<0>>
Unanswered:	<QQ 17671(1454)><<0>>
Ignore Response:	<QQ 17671(1460)><<0>>
Avg Points Awarded:	<QQ 17671(1450)><<0.14>>
... As % of Point Value:	14
Standard Deviation:	<QQ 17671(1456)><<0.38>>

<u>Question 1 Table-Item Links</u>

<u><TB 5114(1301)><<OPS Procedures>></u>

<TB 8551(1305)><<OTO-SA-00001, Engineered Safety Feature Actuation Verification and Restoration>>

<TB 8653(1305)><<OTO-NK-00002, Loss of Vital 125VDC Bus>>

<u><TB 5972(1301)><<OPS Question Catagory>></u>

<TB 8859(1305)><<SRO Only Question>>

Associated objective(s):

<OB 17379(1101)><< C Given a set of plant conditions or parameters indicating an Engineered Safety Feature Actuation Verification and Restoration is required, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.>>

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	Group #	1		
	K/A #	W/E 05 G 2.4.6		
	Importance Rating	4.7		
Knowledge of EOP mitigation strategies.				

Question # 81

A Reactor Trip and Safety Injection have occurred.

- The crew transitioned to FR-H.1, Response to Loss of Secondary Heat Sink
- ‘A’ condensate pump is running with discharge pressure of 320 psig

Per FR-H.1, what EOP Addendum(s) should the CRS direct to restore MFW?

A. ONLY EOP Addendum 30, Establishing Main Feedwater Flow

B. ONLY EOP Addendum 28, Placing the Condensate System in Service AND EOP Addendum 30, Establishing Main Feedwater Flow

C. ONLY EOP Addendum 29, FWIS Bypass Operation AND EOP Addendum 30, Establishing Main Feedwater Flow

D. EOP Addendum 28, Placing the Condensate System in Service, EOP Addendum 29, FWIS Bypass Operation, AND EOP Addendum 30, Establishing Main Feedwater Flow

Answer: C

Explanation:
Per FR-H.1 Step 7 “Try to Establish Main Feedwater flow to at Least one SG”
The first sub set is to check condensate system in service. Based on the conditions in the stem of the question this is met so the CRS should not direct the performance of EOP Add 28. The candidate need to understand what it means “Check condensate system in service. This is met with a condensate pump running and discharge pressure greater 300 psig. Per the procedure a bypass of the FWIS is required per EOP Addendum 29 (This is a plant specific exception to the EOP due to the design of the FWIS). After the FWIS is bypassed Main Feedwater is Established using EOP Addendum 30, Establishing Main Feedwater Flow.

A. Incorrect, Per above, Plausible if the candidate does not understand the plant specific design the FWIS must be bypassed prior to restoring MFW
B. Incorrect, per above. Plausible if the candidate incorrectly believes that since the condensate pump discharge pressure is below the annunciator setpoint of 325 psig that condensate is not

considered "in service" and believes that EOP addendum 28 must be completed prior to restarting the MFW system. Only 300 psig is required for the Condensate system to be considered "in service" Also the FWIS needs to be bypassed prior to restoring MFW.
C. Correct, per above
D. Correct, per above. Plausible if the candidate incorrectly believes that since the condensate pump discharge pressure is below the annunciator setpoint of 325 psig that condensate is not considered "in service" and believes that EOP addendum 28 must be completed prior to restarting the MFW system.

Technical Reference(s):
1. FR-H.1, Response to Loss of Secondary Heat Sink, Rev 16
2. OTA-RK-00027 Add 120B, MFP A Suction Pressure Low, Rev 1

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #-26 FR-H.1/FR-H.2/FR-H.3/FR-H.4/FR-H.5, FRG HEAT SINK (H) SERIES Objective I. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of:
1. FR-H.1, Response To Loss Of Secondary Heat Sink.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

Replaced question per NRC Comment.

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**
• Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
• Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES, The SRO candidate must select the correct EOP addendum.**
• Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
• Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Main Feedwater Pump A Suction Pressure Low	120B
Initiating Device: 1. AEPSL0006 **Setpoint:** 1. 325 psig **Reset:** 1. 332 psig	**MFP A SUCT PRESS LO**

1.0 AUTOMATIC ACTIONS:

1.1. None

2.0 IMMEDIATE ACTIONS:

2.1. None

3.0 OPERATOR ACTIONS:

3.1. IF an additional Condensate Pump is available, Refer To OTN-AD-00001, Condensate System, AND START a Condensate Pump.

3.2. IF a Heater Drain Pump is available, Refer To OTN-AF-00001, High Pressure and Low Pressure Feedwater Heater System, AND START a Heater Drain Pump.

> **NOTE:**
> During operation with one Condensate Pump or one Heater Drain Pump secured, sufficient NPSH is available to the MFP(s) as long as suction pressure remains above 240 psig for two MFPs and above 300 psig for one MFP.

3.3. IF two Main Feedwater pumps are in service AND computer point AEP0006, MFP A SUCT PRESS, is less than 240 psig, Go To OTO-AE-00001, Feedwater System Malfunction.

3.4. IF 'A' MFP is the only MFP running AND computer point AEP0006, MFP A SUCT PRESS, is less than 300 psig, Go To OTO-AE-00001, Feedwater System Malfunction.

3.5. ENSURE Reactor Power is $\leq$ 100%.

3.6. IF desired, RUNBACK Main Turbine load to recover suction pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**TRY To Establish Main Feedwater Flow To At Least One SG:**	
	a. Check Condensate System - IN SERVICE	a. TRY to place Condensate System in service: • Refer To EOP Addendum 28, Placing Condensate System In Service. IF Condensate System can NOT be placed in service, THEN Go To Step 11.
	b. RESET SI if necessary: • SB HS-42A • SB HS-43A	
	c. RESET FWIS: • SB HS-17 • SB HS-18	
	d. BYPASS the FWIS using EOP Addendum 29, FWIS Bypass Operation	
	e. OPEN at least one Feedwater Isolation Valve: • AE HIS-39 (SG A) • AE HIS-40 (SG B) • AE HIS-41 (SG C) • AE HIS-42 (SG D)	e. IF NO Feedwater Isolation Valve can be opened, THEN Go To Step 11.
	f. ESTABLISH Main Feedwater flow: • Refer To EOP Addendum 30, Establishing Main Feedwater Flow	f. IF Main Feedwater flow can NOT be established, THEN Go To Step 9. OBSERVE CAUTION and NOTES prior to Step 9.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
Inoperable/Stuck Control Rod	Group #	2		
	K/A #	005 G2.1.20		
	Importance Rating	4.6		
Ability to interpret and execute procedure steps.				

Question # 82

(REFERENCE PROVIDED)

Reactor Power is 80%.

- Control D4 is misaligned by 18 steps.
- Operators are performing step A.5 of OTO-SF-00001, Check AFD within limits of the Curve Book.
- Axial Flux Difference Values are as follows:

N41	-18%
N42	-16%
N43	-18%
N44	-16%

(1) Is Axial Flux Difference (AFD) within limits?

And

(2) What are the bases for the Rod Group Alignment Technical Specification?

A. (1) The AFD limits are met.
(2) To ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

B. (1) The AFD limits are NOT met.
(2) To ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

C. (1) The AFD limits are met.
(2) To ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met.

D. (1) The AFD limits NOT met.
(2) To ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met.

Answer: B

Explanation:
The operator will be interpreting and executing OSP-NE-00002 step 3.3 which states "The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits" by comparing the data given in the stem to the procedural step. The applicant is given information and must interpret this data along with knowledge of the number of channels outside of the limits which constitute AFD is outside of the limits and then use Curve Book Figure 1-1 to determine if AFD limits are met.

A. Incorrect, Per the conditions given the AFD limit is NOT met. The second part is correct per TS 3.1.4. Plausible if the applicant does not understand that the AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.
B. Correct, Per the conditions given the AFD limit is NOT met. The bases is correct per TS 3.1.4.
C. Incorrect, Per the conditions given the AFD limit is NOT met. The second part is the bases for the AFD limits.(TS 3.2.3). Plausible if the applicant does not understand that the AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limit. The TS bases is for a related TS.
D. Incorrect, Per the conditions given the AFD limit is NOT. The second part is the bases for the AFD limits.(TS 3.2.3) which is plausible because the TS bases is for a related TS.

Technical Reference(s):
1. OSP-SE-00002, AFD Monitor Alarm Inoperable, Rev 11
2. Curve Book Figure 1-1, AFD Limits cycle 21, Rev 394
3. TS Bases 3.1.4

References to be provided to applicants during examination:
1. Curve Book Figure 1-1

Learning Objective: T61.0110, Systems, LP #26, ROD CONTROL - SF Obj U, STATE the Technical Specification limiting conditions for operations (LCOs) applicable to the rod control system.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

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

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

10 CFR Part 55 Content:

10 CFR 55.43(b)(2)

Comments:

Revised explanation per NRC comments and updated stem.

SRO Justification:

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action? **NO**

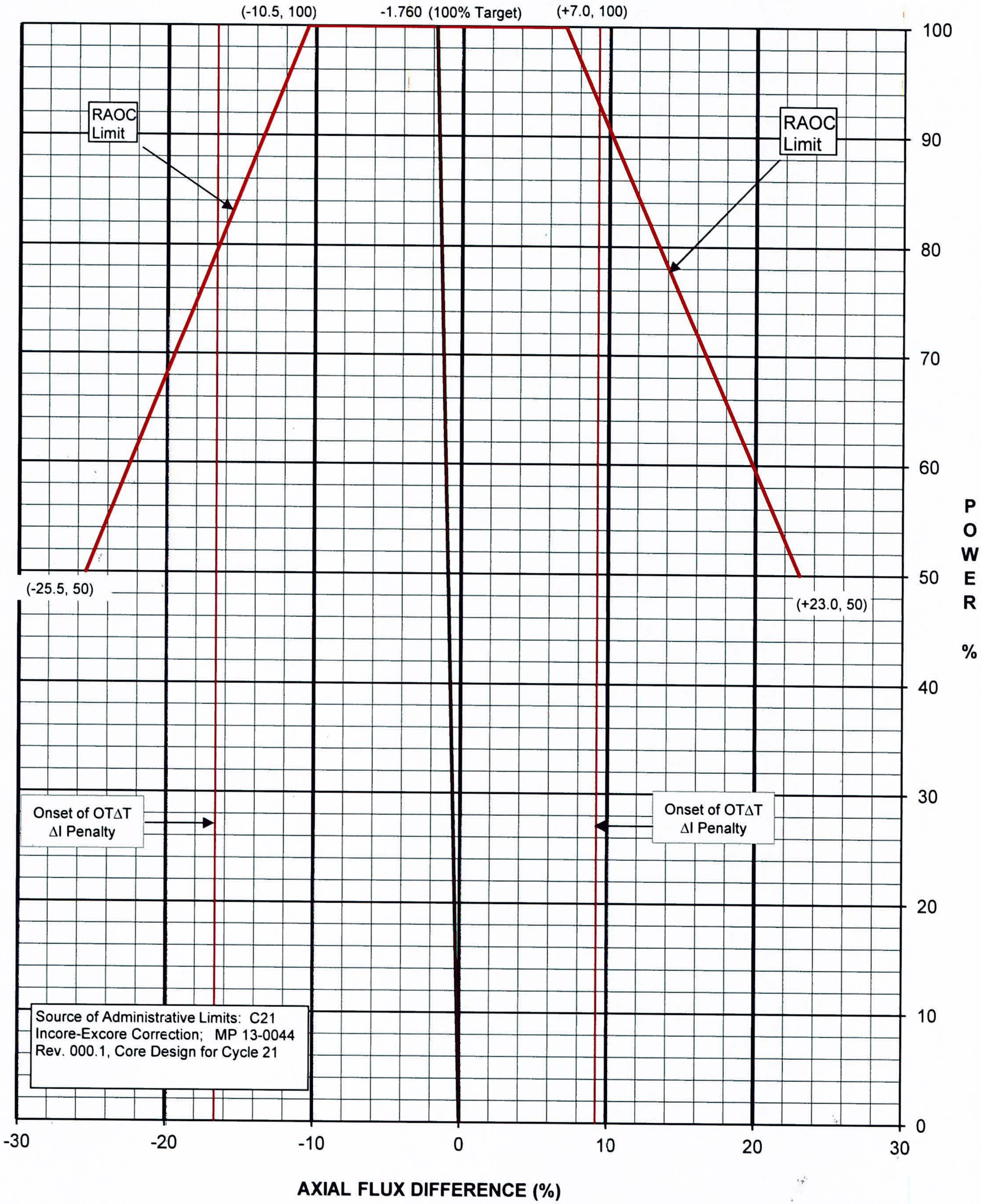
Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?" **NO**

Can question be answered *solely* by knowing the TS Safety Limits? **NO**

Does the question involve one or more of the following for TS, TRM, or ODCM? **YES**

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology **YES**

AXIAL FLUX DIFFERENCE LIMITS
Cycle 21



△ P.P. Roger Wink 6381

△ Annette Grifon 122173 / 3-11-16

Supervising Engineer Date

Ameren MISSOURI

Callaway Plant

OSP-SE-00002

AXIAL FLUX DIFFERENCE MONITOR ALARM INOPERABLE

MINOR Revision 011

AXIAL FLUX DIFFERENCE MONITOR ALARM INOPERABLE

TABLE OF CONTENTS

Section Page Number

AXIAL FLUX DIFFERENCE MONITOR ALARM INOPERABLE

1.0 PURPOSE

Verify the indicated Axial Flux Difference (AFD) is maintained within the limits for Relaxed Axial Offset Control (RAOC) whenever the AFD Monitor Alarm is inoperable.

2.0 SCOPE

Satisfy the requirements of T/S LCO 3.2.3.

3.0 ACCEPTANCE/FUNCTIONAL CRITERIA

> **NOTE**
>
> The RAOC limits of Curve Book, Figure 1-1 may be more restrictive than the corresponding Core Operating Limits Report (COLR). This accounts for additional uncertainty in calibration of AFD that could occur during a transient.

3.1. The AFD, in % flux difference units, shall be maintained within the limits specified in the Curve Book, Figure 1-1. [Ref: 8.1.1]

> **NOTE**
>
> The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging. [Ref: 8.2.1]

3.2. When the AFD Monitor alarm is INOPERABLE, indicated AFD for each OPERABLE excore channel is monitored and logged at least once per hour for the first 24 hours AND at least once per 30 minutes thereafter.

3.3. The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits. [Ref: 8.1.1]

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately $^5/_8$ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among four control banks and five shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. All control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining three shutdown banks (C, D, and E) contain one rod group.

(continued)

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	1		
Steam Generator Over-pressure	**Group #**	2		
	K/A #	W/E 13 EA2.2		
	Importance Rating	3.4		
Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.				

Question # 83

A reactor trip has occurred.

- The crew has transitioned to ES-0.1, Reactor Trip Response.
- Plant parameters are as follows:
  - All SG levels are 85% NR.
  - AFW flow indicates 100,000 LBM/hr.
  - "A", "B", and "C" SG pressures indicate 1050 psig.
  - "D" S/G pressure indicates 1240 psig.

What is the HIGHEST priority procedure related to the Heat Sink CSF that would be implemented based on the above conditions?

A. FR-H.2, Response to Steam Generator Overpressure

B. FR-H.3, Response to Steam Generator High Level

C. FR-H.4, Response to Loss of Normal Steam Release Capability

D. FR-H.5, Response to Steam Generator Low Level

Answer: A

Explanation: CSF-1, Figure 3, Heat Sink status tree

A. Correct, as the "D" SG is greater than 1234 psig and all S/G levels are greater than 7% NR. AFW flow is greater than 285,000lbm/hr
B. Incorrect. NR SG level is higher than the normal control band (45-55%) and a transition to this FR procedure is plausible if it is believed the entry conditions have been satisfied.
C. Incorrect, credible because conditions are met, but this is a subsequent yellow path that has lower priority
D. Incorrect, credible because conditions are met, but this is a subsequent yellow path that has lower priority

Technical Reference(s):

1. CSF-1, CRITICAL SAFETY FUNCTION STATUS TREES, Rev 10

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations: LP #26, FR-H.1/FR-H.2/FR-H.3/FR-H.4/FR-H.5 FRG HEAT SINK (H) SERIES Objective B. DESCRIBE the Symptoms and/or Entry Conditions for:

2. FR-H.2, Response To Steam Generator Overpressure.

Question Source: Bank # __2007 Wolf Creek NRC Exam ML073440493.pdf____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2007 Wolf Creek NRC Exam ML073440493________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

SRO Justification:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

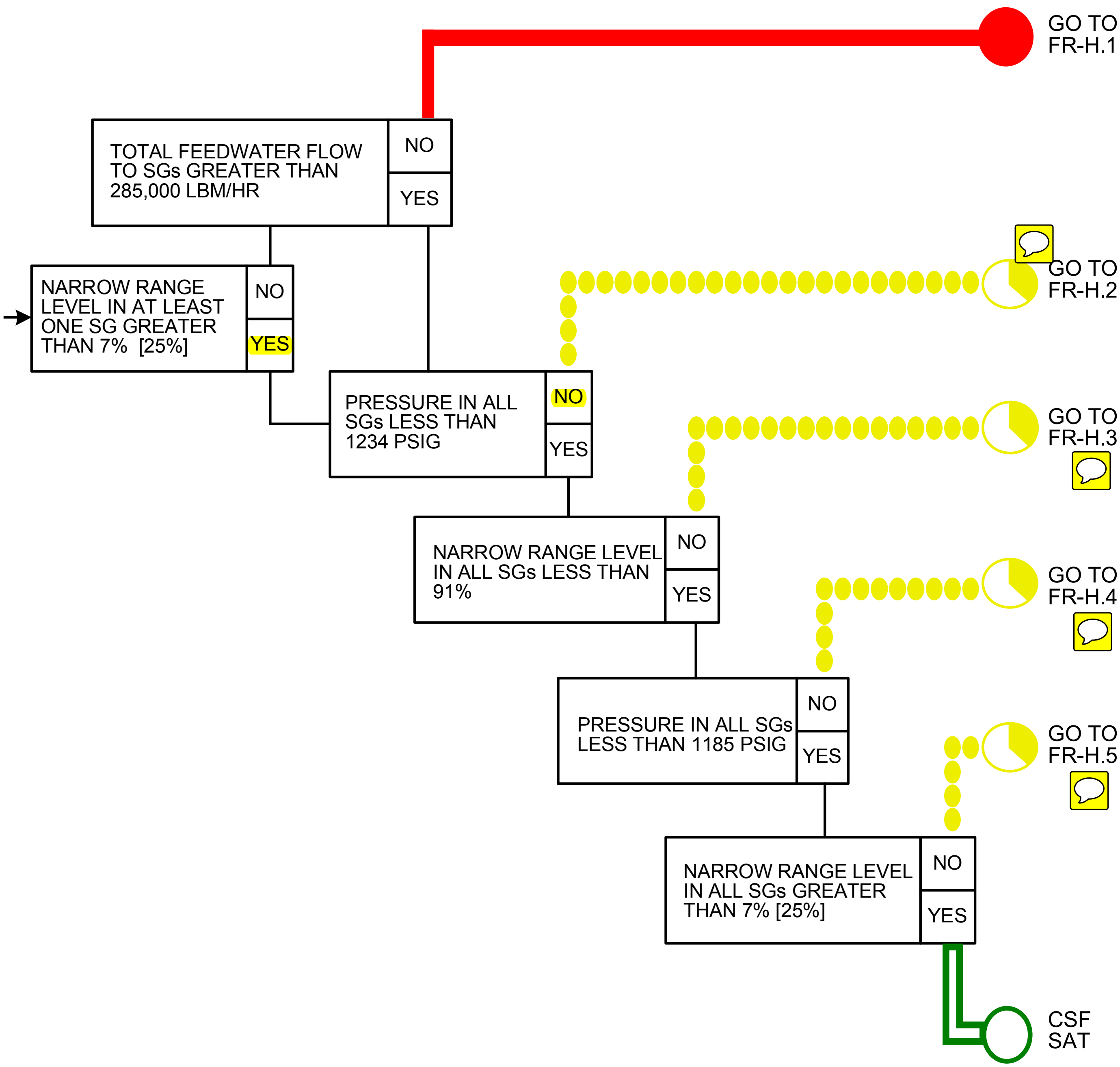
Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **NO**
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures **YES**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures **NO**

Note: all of the potential choices in the question are YELLOW path FR procedures and since they are not Red or Orange Path entry conditions; this question is at the SRO level per ES401 Attachment 2 page 7.

Revised distractor A (included FR-H.3) per NRC comments and reordered the choices numerically.

Figure 3
Heat Sink



Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	1		
High Reactor Coolant Activity / 9	**Group #**	2		
	K/A #	00076 G2.2.40		
	Importance Rating	4.7		
Ability to apply Technical Specifications for a system				

Question # 84

(REFERENCE PROVIDED)

Reactor Power is 50%.

Chemistry reports the following:
- RCS Dose Equivalent I-131 is 2.0 µCi/gm
- Dose Equivalent Xenon-133 Activity is 20 µCi/gm

What is the MOST limiting action or set of actions required by Technical Specifications?

A. Verify Dose Equivalent I-131 ≤60 µCi/gm once per 4 hours
AND
Restore Dose Equivalent I-131 to within limit in 48 hours

B. Restore Dose Equivalent XE-133 to within limit in 48 hours

C. Verify Dose Equivalent I-131 ≤60 µCi/gm once per 4 hours
AND
Restore Dose Equivalent I-131 to within limit in 48 hours
AND
Restore Dose Equivalent XE-133 to within limit in 48 hours

D. Be in MODE 3 in 6 hours AND Be in MODE 5 in 36 hours

Answer: A

Explanation:

A. Correct, The only activity that is exceeding the TS limit is I-131. The I-131 limit is 1.0 µCi/gm. Per TS 3.4.16
B. Incorrect, The XE-133 activity does not exceed its TS limit. Plausible if the applicant does not know the XE limit then this would be the correct action if I-131 was not above its TS limit.
C. Incorrect, The XE-133 activity does not exceed its TS limit. Plausible because if both I-131 and XE-133 were above the TS limit this would be the correct action
D. Incorrect, The time has not been exceeded to correct the high I-131 and I-131 is not above 60

µCi/gm. Plausible because it is the action to take if I-131 was above 60 µCi/gm

Technical Reference(s):
1. TS 3.4.16 RCS Specific Activity

References to be provided to applicants during examination:
1. TS 3.4.16 RCS Specific Activity

Learning Objective: T61.003 B, Off Normal Operations, LP #B-14, OTO-BB-00005, Reactor Coolant System High Activity Obj. D, Given a set of plant conditions or parameters indicating RCS High Activity, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis ___X__

10 CFR Part 55 Content:

10 CFR 55.43(b)(2)

Comments:

All 4 choices are plausible due to the fact that is the sample results given in the stem were changed the SRO would have to reanalyze the conditions and a different choice could be the correct answer.

Can question be answered solely by knowing ≤ 1 hour TS/TRM Action? **NO**

Can question be answered solely by knowing the LCO/TRM information listed “above-the-line?” **NO**

Can question be answered solely by knowing the TS Safety Limits? **NO**

Does the question involve one or more of the following for TS, TRM, or ODCM? **YES**

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1) **YES the SRO must analyze the information given, use the reference provided and determine the correct action to take**
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	------------------ NOTE ------------------ LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	------------------ NOTE ------------------ LCO 3.0.4.c is applicable. ---	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > µCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	------------------- NOTE ------------------- LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 $\leq$ 60 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	------------------- NOTE ------------------- LCO 3.0.4.c is applicable. ---	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	-------------------------------- NOTE --------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 225 μCi/gm.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.2	-------------------------------- NOTE --------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of $\geq$ 15% RTP within a 1 hour period

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
W/E09&E10 Natural Circ	Group #	2		
	K/A #	W/E10 EA2.2		
	Importance Rating	3.9		
Ability to determine and interpret the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.				

Question # 85

A reactor trip occurred due to a loss of offsite power.

- Shortly after the trip, the BOP reports the following annunciators are LIT:
  - 25A, NN01 INST BUS UV
  - 57D, RVLIS B HYD FAIL
  - 57E, RVLIS PWR Failure
- The operating crew is performing ES-0.2 Natural Circulation Cooldown.
- RCS pressure is 1920 psig.
- The RCS Cooldown and depressurization MUST be performed due to secondary systems water inventory concerns.
- It is suspected that a steam void has formed in the RX Vessel.

(1) Which of the following annunciators can be used to verify that a steam void has formed in the RX Vessel?

And

(2) The CRS should direct which of the following procedures?

A. (1) 32A, PZR Level High
(2) Transition to ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)

B. (1) 32A, PZR Level High
(2) Transition to ES-0.4, Natural Circulation Cooldown With Steam Void in Vessel (Without RVLIS)

C. (1) 33C, Pressurizer Pressure Low - Heaters On
(2) Transition to ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)

D. (1) 33C, Pressurizer Pressure Low - Heaters On
(2) Transition to ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel

(Without RVLIS)

Answer: B

Explanation:
'A' RVLIS is powered from NN01 so a loss of NN01 means that A RVLIS is not available. Annunciator 57D indicates that 'B' RVLIS is also not available.

And per the NOTE prior to step 13 in ES-0.2 that states:
"If at any time it is determined that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel, one of the following procedures should be used:
ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS) or
ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)"

Therefore, ES-0.4 is correct based on plant conditions and the fact that no train of RVLIS is operable.

If A Steam void is suspected of forming in the vessel, this void will force water into the pressurizer and annunciate 32A, PZR level high. Pressurizer pressure would be going up not down as the PZR bubble would be squeezed by the incoming surge. This question is basically modeling the TMI accident with the exception of a failed open PZR PORV. With a LOCA in progress, it is plausible that a low PZR Pressure Alarm will be received. While there is no LOCA in this question, 33C is plausible if the student applies the TMI accident concept from memory without understanding the reason. Furthermore, step 13 of ES-0.2, RNO for part C directs using a PZR PORV as letdown would not be in service. Opening the PZR PORV would give the PZR Low alarm as pressure is relieved to the PRT. However, as explained above, the Note prior to step 13 would direct the operator to either ES-0.3 or ES-0.4 and the crew would not be performing step 13. Additionally, the PORV operation leading to a low PZR pressure alarm is plausible as certain steps in ES-0.2 direct use of a PZR PORV which would create a low PZR Pressure.

RCS pressure of 1920 psig indicates that the crew is at step 12 of ES-0.2.

A. Incorrect
B. Correct
C. Incorrect
D. Incorrect

Technical Reference(s):
1. ES-0.3, Natural Circulation Cooldown with Steam Void In Vessel (with RVLIS), Rev 12
2. ES-0.4, Natural Circulation Cooldown with Steam Void In Vessel (without RVLIS), Rev 11
3. EOP Addendum 1, Natural Circulation Verification, Rev 2
4. ES-0.2, Natural circulation Cooldown, Rev 11
5. The following list of Annunciator Response Procedures:
    a. OTA-RK-25A
    b. OTA-RK-32A
    c. OTA-RK-32D
    d. OTA-RK-56B
    e. OTA-RK-57C
    f. OTA-RK-57D
    g. OTA-RK-57E

References to be provided to applicants during examination: None

Learning Objective:

T61.003D, Emergency Operations, LP #7, ES-0.2, ES-0.3, ES-0.4 Natural Circulation Objective:

G. STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from the following procedures to other procedures:
1. ES-0.2
2. ES-0.3
3. ES-0.4

H. OUTLINE procedural flow path including major system and equipment operation in accomplishing the goal of the following procedures:
1. ES-0.2
2. ES-0.3
3. ES-0.4

Question Source: Bank # ______
Modified Bank # __X_L22582___
New _______

Question History: Last NRC Exam ___Modified from 2014 SRO Retake_________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

SRO Justification:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **NO**
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures **YES**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures **NO**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

If at any time it is determined that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel, one of the following procedures should be used:

ES–0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS) or
ES–0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**INITIATE RCS Depressurization:**	
	a. CHECK THREE CRDM Fans – RUNNING • GN HIS–41 • GN HIS–42 • GN HIS–43 • GN HIS–44	a. ENSURE THREE CRDM Fans Running, IF Less Than THREE CRDM Fans RUNNING, PERFORM the following: 1) MAINTAIN RCS subcooling greater than 130°F. 2) Go To Step 13.c.
	b. MAINTAIN RCS subcooling – GREATER THAN 80°F	
	c. CHECK letdown – IN SERVICE	c. PERFORM the following: 1) USE one PZR PORV. 2) Go To Step 14.

(Step 13. continued on next page)

<QQ 22582(1410)><<Given the following plant conditions:

- A Reactor trip occurred due to a loss of offsite power.
- Shortly after the trip, the BOP reports the following annuciators are LIT:
- 25A, NN01 INST BUS UV
- 57E, RVLIS PWR Failure
- The operating crew is performing ES-0.2 Natural Circulation Cooldown.
- RCS pressure is 1920 psig.
- The RCS Cooldown and depressurization MUST be performed due to secondary systems water inventory concerns.
- It is suspected that a steam void has formed in the RX Vessel.

(1) Which of the following annunciators can be used to verify that a steam void has formed in the RX Vessel?

And

(2) The CRS will direct which of the following procedures?>>

A. <QQ 22582(1482)><<(1) 32A, PZR Level High
(2) Transition to ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)>>

B. <QQ 22582(1480:0)><<(1) 32A, PZR Level High
(2) Transition to ES-0.4, Natural Circulation Cooldown With Steam Void in Vessel (Without RVLIS)>>

C. <QQ 22582(1480:1)><<(1) 33C, Pressurizer Pressure Low - Heaters On
(2) Transition to ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)>>

D. <QQ 22582(1480:2)><<(1) 33C, Pressurizer Pressure Low - Heaters On
(2) Transition to ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)>>

Answer: <QQ 22582 (1419) ><<A >>

Question 1 Info	
Question Type:	<QQ 22582(1401)><<Multiple Choice>>
Status:	<QQ 22582(1405)><<Active>>
Always select on test?	<QQ 22582(1406)><<No>>
Authorized for practice?	<QQ 22582(1429)><<No>>
Points:	<QQ 22582(1441)><<1.00>>
Time to Complete:	<QQ 22582(1408)><<0>>
Difficulty:	<QQ 22582(1407)><<0.00>>
System ID:	<QQ 22582(1445)><<22582>>
User-Defined ID:	<QQ 22582(1404)><<L22582>>
Cross Reference Number:	
Topic:	<QQ 22582(1400)><<L22582 Transition to ES-0.3>>
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<QQ 22582(1411)><<2014 SRO Retake. CJE 1/22/16>>

Question 1 History	
Exam Appearances:	<QQ 22582(1449)><<0>>
Student Encounters:	<QQ 22582(1448)><<0>>
Answered Right:	<QQ 22582(1452)><<0>>
Answered Wrong:	<QQ 22582(1453)><<0>>
Partially Correct:	<QQ 22582(1459)><<0>>
Answer Invalid:	<QQ 22582(1455)><<0>>
Unanswered:	<QQ 22582(1454)><<0>>
Ignore Response:	<QQ 22582(1460)><<0>>
Avg Points Awarded:	<QQ 22582(1450)><<0.00>>
... As % of Point Value:	0
Standard Deviation:	<QQ 22582(1456)><<0.00>>

Question 1 Table-Item Links

<TB 5114(1301)><<OPS Procedures>>

<TB 5223(1305)><<ES-0.2, NATURAL CIRCULATION COOLDOWN>>

<TB 5224(1305)><<ES-0.4, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS)>>

<TB 5225(1305)><<ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)>>

<TB 5972(1301)><<OPS Question Catagory>>

<TB 5979(1305)><<SRO NRC Quality>>

<TB 8859(1305)><<SRO Only Question>>

Associated objective(s):

<OB 11341(1101)><<**H. OUTLINE** procedural flowpath including major system and equipment operation in accomplishing the goal of the following procedures:
1. ES-0.2
2. ES-0.3
3. ES-0.4>>

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	2		
Residual Heat Removal	**Group #**	1		
	K/A #	005 G2.4.9		
	Importance Rating	4.2		
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.				

Question # 86

The plant is operating at reduced inventory following a scheduled shutdown for refueling with 'A' RHR train is in service.

Then, a Loss of all AC occurs.

Per OTO-EJ-00003, Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions, what is the FIRST action the CRS should direct?

A. Establish VCT Overpressure Feed per Addendum 4 of OTN-BG-00004, VCT Atmosphere Control

B. Establish SI Accumulator Injection per Addendum 6 of OTN-BB-00002, Draining the RCS to Limited Inventory or Reduced Inventory

C. Initiate RCS Feed and Bleed per Attachment A, of OTO-EJ-00003 Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions

D. Initiate RWST Gravity Feed per Attachment E, of OTO-EJ-00003 Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions

Answer: D

Explanation: *Per OTO-EJ-00003, LOSS OF RHR WHILE OPERATING AT REDUCED INVENTORY OR MID-LOOP CONDITIONS Step 2.a." RHR pumps - ANY RUNNING" the answer is NO based on the information in the stem. RNO step a) is attempt to restore power. Step b) is Initiate RWST gravity Feed per Att E.*

A. Incorrect, Per above, Plausible this is an action the candidate could take to refill RHR in step 12 of the procedure
B. Incorrect, per above. Plausible this is the action the applicant could take in Step 10 of the procedure
C. Incorrect, per above, Plausible as it the action the candidate could take in either step 8 or step 12
D. Correct, per above.

Technical Reference(s):

1. OTO-EJ-00003, LOSS OF RHR WHILE OPERATING AT REDUCED INVENTORY OR MID-LOOP CONDITIONS, Rev 10

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, B-62, LOSS OF RHR, (OTO-EJ-00001), (OTO-EJ-00002), (OTO-EJ-00003) OBJ,H. STATE major action categories and symptoms/entry conditions for OTO-EJ-00003, Loss of RHR while operating at reduced inventory or mid loop.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?
• Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
• Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES, The SRO candidate determine which Attachment to implement.**
• Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
• Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

OTO-EJ-00003

LOSS OF RHR WHILE OPERATING AT REDUCED INVENTORY OR MID-LOOP CONDITIONS

Revision 009

CONTINUOUS USE

A. PURPOSE

This procedure provides the actions necessary for maintaining core cooling and protecting the reactor core in the event that RHR cooling is lost during mid—loop or reduced inventory operations.

Major Action Categories:

- Check If RHR Pumps Should Be Stopped.
- Address Containment Related Concerns.
- Establish Alternate Means of Decay Heat Removal.
- Establish Support Conditions And Restore RHR Cooling.

B. SYMPTOMS OR ENTRY CONDITIONS

1) The following symptoms are indicative of a loss of RHR system while operating at mid—loop or reduced inventory conditions:

   a. No RHR pumps running.

   b. Low RHR flow alarm annunciating.

   c. Air—Binding of the operating RHR pump as indicated by any of the following:

      1) Motor current oscillations.

      2) Erratic flow oscillations.

      3) Excessive pump noise. Pump cavitation.

   d. Other conditions requiring the tripping of RHR pumps.

2) This procedure may also be entered from:

   a. OTO—EJ—00001, Loss Of RHR Flow, when RHR cooling is lost while operating at mid—loop or reduced inventory conditions.

C. REFERENCES

1) Implementing:

   a. ODP—ZZ—00001 Addendum 13, Shift Manager Communications

   b. OSP—GT—00003, Containment Closure

   c. OTO—EJ—00001, Loss OF RHR Flow

   d. OTO—NB—00001, Loss Of Power To NB01

   e. OTO—NB—00002, Loss Of Power To NB02

   f. EOP Addendum 39, Alternate Emergency Power Supply

   g. EOP Addendum 7, Restoring Offsite Power

   h. BD—EJ—00003, Background document for Loss of RHR While Operating at Reduced Inventory or Mid—Loop Operations

2) Developmental:

   a. ARG—1, Loss Of RHR While Operating At Mid—Loop Conditions

   b. WCAP—14089,Rev.1, Supplement 1

   c. WCAP 11916

   d. CAR 200602230

   e. COMN 41768

   f. COMN 41774

   g. COMN 41780

   h. OTN—EG—00001

   i. OTN—BB—00002

   j. Calculation EC—31, Rev.1, Add 1

   k. Calculation EC—31, Rev.1, Add 2

   l. Calculation EC—34, Rev.1, Add 1

   m. Calculation EC—51, Rev.0, Add 1

   n. Calculation EJ—51, Rev 0, Add 2

   o. Calculation EJ—51, Rev 0, Add 3

   p. Calculation ZZ—437

   q. RFR 200603491

   r. Calculation EC—36 Rev. 1 Add 1 for Refuel 19 Hot and Cold Core

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- Changes in RCS pressure could result in inaccuracies in RCS level readings.
- The Standby RHR Pump should not be started unless the cause of the loss of flow is known and corrective action has been taken.

NOTE

IF in MODE 6 AND temperature rises to greater than 140°, the Shift Manager should Refer To EALs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**CHECK RCS Condition — MIDLOOP OR REDUCED INVENTORY**	Go To OTO—EJ—00001, Loss Of RHR Flow.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**CHECK If RHR Pumps Should Be Stopped:**	
	a. RHR pumps — ANY RUNNING	a. PERFORM the following: 1) IF NO AC Emergency Bus is energized, THEN PERFORM the following: a) Refer To the following to restore power while continuing with this procedure: • OTO—NB—00001 • OTO—NB—00002 • EOP Addendum 7 • EOP Addendum 39 b) INITIATE RWST Gravity Feed per Attachment E, RWST Gravity Feed 2) Go To Step 3.
	b. RCS level — GREATER THAN 16 INCHES	b. PERFORM the following: 1) STOP RHR Pumps and PLACE in PULL—TO—LOCK: • EJ HIS—1 • EJ HIS—2 2) Go To Step 3.
	c. RHR flow — LESS THAN 1350 GPM	c. LOWER RHR Flow to 1350 GPM.
	d. MONITOR RHR Pump Suction Conditions — NO INDICATION OF CAVITATION • RHR Pump Discharge Pressure — STABLE AND NORMAL FOR DISCHARGE FLOW • EJ PI—614 • EJ PI—615 • RHR Pump Motor Amps — STABLE AND NORMAL FOR DISCHARGE FLOW	d. PERFORM the following: 1) STOP Cavitiating RHR Pump(s) and PLACE in PULL—TO—LOCK: • EJ HIS—1 • EJ HIS—2 2) Go To Step 3.
	e. Go To appropriate plant procedure as directed by the Shift Manager/Control Room Supervisor	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. ISOLATE Letdown And Known Drain Paths:

a. CVCS letdown:

1) CLOSE Letdown Orifice Isolation valves:

- BG HIS—8149AA
- BG HIS—8149BA
- BG HIS—8149CA

2) CLOSE RCS Letdown To Regen HX valves:

- BG HIS—459
- BG HIS—460

b. RHR letdown:

1) CLOSE RHR Cleanup To Letdown HX Flow Control valve:

- BG HC—128

c. ISOLATE other known drain paths with control switches in Control Room

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Computer generated / daily T—Boil data is preferred.

4. DETERMINE Time To Boiling Based On Existing Conditions:

a. Refer To the following to determine Time To RCS Boiling:

- Time To Boil recorded in RO logs

  OR

- Figure 1, Mid—Loop Time To Boil (Hot Core)

  OR

- Figure 2, Reduced Inventory Time To Boil (Hot Core)

  OR

- If Core is loaded with fresh fuel at mid—loop, Figure 3, Mid—Loop Time To Boil (Cold Core)

  OR

- If Core is loaded with fresh fuel at reduced inventory, Figure 4, Reduced Inventory Time To Boil (Cold Core)

b. Time To Boiling — LESS THAN 30 MINUTES

RESPONSE NOT OBTAINED:

b. PERFORM the following:

1) DIRECT CTMT Coordinator and pre—designated Operations Technicians to COMPLETE CTMT Closure within 30 minutes.

2) Go To Step 8. OBSERVE CAUTIONS prior to Step 8.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. INITIATE Actions To Protect Personnel Working In Containment:

a. EVACUATE NON—essential personnel in containment

1) PUSH the Containment Evacuation Alarm:

- QFHS—1

2) RESET the Containment Evacuation Alarm:

- QFHS—1

3) ANNOUNCE the following on the Gaitronics:

Attention all personnel
Attention all personnel

A Loss Of RHR has occurred

All non—essential personnel evacuate Containment

4) REPEAT the announcement

Response Not Obtained:

a. NOTIFY the Security Shift Supervisor by radio or telephone to Initiate Containment Evacuation

b. Periodically MONITOR containment radiation conditions:

- GTRE0031
- GTRE0032
- GTRE0022
- GTRE0032
- SDRE0039 (Seal Table)
- SDRE0040 (Pers. Hatch)
- SDRE0041 (Man. Crane)
- SDRE0042 (2047' SE Area)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Containment Closure must be completed within 30 minutes of the event or prior to T—Boil, whichever is less.

6. **INITIATE Actions To Establish Containment Closure:**

   a. CLOSE Equipment hatch

   b. CLOSE Personnel hatch

   c. CLOSE Emergency Personnel hatch

   d. CLOSE all open Containment penetrations:

      - Refer To OSP—GT—00003, Containment Closure, as necessary for tracked discrepancies

7. **START Available Containment Cooler Fans:**

   a. PLACE Containment Cooler Fan Speed Selector switch(es) in SLOW:

      - GN HS—9
      - GN HS—17
      - GN HS—5
      - GN HS—13

   b. START Containment Cooler Fan(s):

      - GN HIS—9
      - GN HIS—17
      - GN HIS—5
      - GN HIS—13

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- Personnel working in containment should be warned before refilling the RCS to avoid inadvertent contamination of personnel working near RCS openings.
- Only borated water should be added to the RCS to maintain adequate shutdown margin.

NOTE

If Core Exit TCs are not available RCS wide range temperature may be used.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
# 8.	**CHECK Core Exit TCs — LESS THAN 200°F**	PERFORM the following: a. ESTABLISH SI Pump hot leg injection using at least one SI Pump per Attachment A. b. IF Core Exit TCs continue to rise, THEN ESTABLISH additional ECCS flow as follows: • ESTABLISH additional ECCS Cold Leg injection flow using at least one CCP per Attachment C. <u>OR</u> • ESTABLISH additional ECCS Cold leg injection using at least one SI Pump per Attachment B. c. Go To Step 13.
9.	**CHECK RCS Level:**	
	a. RCS Level — LESS THAN 17 INCHES	a. Go To Step 13.
	b. RCS Level — LESS THAN 9 INCHES	b. Go To Step 12.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. REFILL RCS As Follows:

a. ESTABLISH ECCS flow using at least one CCP aligned for cold leg injection:

1) OPEN CCP Suction From RWST valve(s):
   - BN HIS—112D
   - BN HIS—112E

2) CLOSE VCT Outlet valves:
   - BG HIS—112B
   - BG HIS—112C

3) CLOSE Charging Pumps To Regen HX Containment Isolation valve(s):
   - BG HIS—8105
   - BG HIS—8106

4) OPEN Boron Injection Header Inlet valves:
   - EM HIS—8803A
   - EM HIS—8803B

5) OPEN Boron Injection Header Outlet valves:
   - EM HIS—8801A
   - EM HIS—8801B

6) START at least one CCP:
   - BG HIS—1A
   - BG HIS—2A

RESPONSE NOT OBTAINED:

a. REFILL the RCS using any of the following:

- START at least one SI Pump per Attachment B, SI Pump Injection To Cold Legs

  OR

- GRAVITY FEED per Attachment E, RWST Gravity Feed

  OR

- SI Accumulator Injection

b. REFILL RCS with ECCS UNTIL either of the following conditions satisfied:

- RHR Cooling — RESTORED

  OR

- PZR Level Cold Cal — GREATER THAN 50%
  - BB LI—462

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.	**Go To Step 13.**	
12.	**REFILL RCS To 17 Inches:**	
	a. Charging Pump — IN SERVICE • BG HIS—3 (NCP) • BG HIS—1A (CCP A) • BG HIS—2A (CCP B)	a. START one charging pump. IF NO charging pump can be started, THEN MAKEUP to RCS using any of the following: • RWST gravity feed (Attachment E, RWST Gravity Feed) OR • VCT overpressure feed OR • SI Accumulator injection OR • Other method(s) as determined by Plant Engineering Staff
	b. FEED RCS Until Level — GREATER THAN 17 INCHES	
13.	**IDENTIFY And ISOLATE Any RCS Leakage**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

To maintain RCS pressure low enough for gravity feed, at least two SGs must be refilled and utilized.

14. DETERMINE If A Secondary Heat Sink Should Be Established:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	a. ECCS Feed Path — NOT ESTABLISHED	a. IF ECCS feed path is established, THEN Go To Step 15. OBSERVE NOTES prior to Step 15.
	b. SGs — ANY AVAILABLE	b. PERFORM the following: 1) Try To RESTORE SG(s) to service. 2) WHEN SG(s) become available, THEN PERFORM Steps 14.c through 14.e. 3) CONTINUE with Step 15. OBSERVE NOTES prior to Step 15.
	c. FEED available SG(s) as necessary to establish and maintain SG level	
	d. PLACE associated SG ASD controller(s) in MANUAL	
	e. OPEN associated SG ASD(s)	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- The time to boiling in the RCS should be taken into consideration when determining how much time should be spent venting the RHR system prior to taking additional actions for alternate cooling sources.
- If adequate time to completely vent the RHR system is NOT available, air can be swept out of the RHR lines by filling the RCS to 27 Inches and running an RHR pump at a flowrate greater than 3850 GPM.

15. VENT RHR System As Necessary:

a. MAINTAIN RCS level while venting RHR system

b. VENT the desired RHR train per Attachment F, Venting The RHR System

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.	**ESTABLISH Conditions To Start RHR Pump:**	
	a. RCS Level — GREATER THAN 17 INCHES	a. PERFORM the following: 1) TREND Core Exit TCs per Attachment D, Core Exit Thermocouple Temperature Monitoring, If Core Exit Thermocouples are not available use RCS Wide Range Temperature 2) CONSULT Plant Engineering Staff for recommended alternate cooling methods: • RCS Feed and Bleed (Att A,B,or C) • RWST Gravity Feed (Att E) • Steam Generator Reflux Cooling • SI Accumulator Injection 3) Return To Step 5.

(Step 16. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 16. (continued from previous page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. RHR Pump — AVAILABLE	b. PERFORM the following: 1) TREND Core Exit TCs per Attachment D, Core Exit Thermocouple Temperature Monitoring, If Core Exit Thermocouples are not available use RCS Wide Range Temperature 2) CONSULT Plant Engineering Staff for recommended alternate cooling methods: • RCS Feed and Bleed (Att A,B,or C) • RWST Gravity Feed (Att E) • Steam Generator Reflux Cooling • SI Accumulator Injection 3) Return To Step 5.

(Step 16. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 16. (continued from previous page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. CHECK RHR system valves – IN PROPER ALIGNMENT	c. ALIGN valve(s) as necessary.

1) RWST To RHR Pump Suction valves – CLOSED
   - BN HIS–8812A
   - BN HIS–8812B

2) Containment Recirc Sump To RHR Pump Suction valves – CLOSED
   - EJ HIS–8811A
   - EJ HIS–8811B

3) Loop Hot Leg To RHR Pump Suction valves – OPEN
   - RHR Pump A:
     - BB HIS–8702A
     - EJ HIS–8701A
   - RHR Pump B:
     - BB HIS–8702B
     - EJ HIS–8701B

4) RHR To Accumulator Injection Loop valves – OPEN:
   - EJ HIS–8809A
   - EJ HIS–8809B

(Step 16. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 16. (continued from previous page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
d. CHECK CCW cooling to RHR System — IN SERVICE	
1) CCW To RHR HX valves — OPEN • EG HIS—101 • EG HIS—102	1) ALIGN valve(s) as necessary.

2) CCW Pumps — ONE RUNNING IN EACH TRAIN
• Red Train:
  • EG HIS—21 or EG HIS—23
• Yellow Train:
  • EG HIS—22 or EG HIS—24
	2) START CCW Pump(s) as necessary.

e. CHECK Core Exit TCs — LESS THAN 200°F	e. PERFORM the following: 1) ESTABLISH SI Pump hot leg injection using at least one SI Pump per Attachment A 2) IF Core Exit TCs continue to rise, THEN ESTABLISH additional ECCS flow as follows: • ESTABLISH additional ECCS Cold Leg injection flow using at least one CCP per Attachment C <u>OR</u> • ESTABLISH additional ECCS Cold leg injection using at least one SI Pump per Attachment B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Starting an RHR pump may result in RCS level lowering due to shrink or void collapse.

NOTE

The RCS level necessary to operate RHR pumps is a function of RHR flow. Figure 5, Minimum Hot Leg Level Vs RHR Flow, should be referred to in order to determine the required level necessary.

17. RESTORE RHR Flow:

a. CLOSE RHR HX Flow Control valves:

- EJ HIC—606
- EJ FK—618 (Bypass)
- EJ HIC—607
- EJ FK—619 (Bypass)

b. START one RHR Pump:

- EJ HIS—1
- EJ HIS—2

c. MAINTAIN RCS level within the acceptable region of Figure 5, Minimum Hot Leg Level Vs RHR Flow

d. Slowly RAISE RHR HX Bypass Flow to desired flowrate:

- EJ FK—618
- EJ FK—619

(Step 17. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Step 17.	(continued from previous page)	
	e. RHR Flow — RESTORED	e. PERFORM the following: 1) TREND Core Exit TCs per Attachment D, Core Exit Thermocouple Temperature Monitoring 2) CONSULT Plant Engineering Staff for recommended alternate cooling methods: • RCS Feed and Bleed (Att A,B,or C) • RWST Gravity Feed (Att E) • Steam Generator Reflux Cooling • SI Accumulator Injection 3) Return To Step 5.
	f. ESTABLISH desired RCS cooldown rate: • EJ HIC—606 • EJ HIC—607	
18.	**CHECK If RCS Makeup Should Be Reduced:**	
	a. RCS Temperature — LESS THAN 200°F	a. CONTINUE cooling with RHR. Return To Step 17.f.
	b. RCS Level — STABLE OR RISING	b. Return To Step 17.c.
	c. STOP any running SI Pump: • EM HIS—4 • EM HIS—5	
	d. CONTROL charging flow to maintain RCS level within the acceptable region of Figure 5, Minimum Hot Leg Level Vs RHR Flow	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	**CHECK RCS Temperature — LESS THAN 140°F**	CONTINUE cooling with RHR. Return To Step 17.f.
20.	**Go To Appropriate Plant Procedure As Directed By The Shift Manager/Control Room Supervisor**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 1 of 5)
SI Pump Injection To Hot Legs

A1. DETERMINE Appropriate SI Pump To Be Started:

a. At least One SI Pump is available

RNO: a. RESTORE One SI Pump incapable of injection per OSP—EM—00002.

b. ENSURE appropriate support conditions are met

A2. OPEN RWST To SI Pump Suction valve(s):

- BN HIS—8806A (SI Pump A)
- BN HIS—8806B (SI Pump B)

A3. OPEN SI Pump Suction valve(s):

- EM HIS—8923A (SI Pump A)
- EM HIS—8923B (SI Pump B)

A4. OPEN SI Pump Recirc To RWST valve(s):

a. OPEN the following valve(s):

- EM HIS—8814A (SI Pump A)
- EM HIS—8814B (SI Pump B)

b. CHECK BN HIS—8813 OPEN

RNO: b. PERFORM the following:

1) PLACE Power Lockout for BN HIS—8813 in NON ISO position:

- BN HIS—8813A

2) OPEN BN HIS—8813

3) PLACE Power Lockout for BN HIS—8813 in ISO position:

- BN HIS—8813A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 2 of 5)
SI Pump Injection To Hot Legs

A5. CLOSE SI Pump Discharge To Cold Leg Injection valve(s):

- EM HIS—8821A (SI Pump A)
- EM HIS—8821B (SI Pump B)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 3 of 5)
SI Pump Injection To Hot Legs

A6. OPEN SI Pump Discharge To Hot Leg Injection valve(s):

- SI Pump A:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. CHECK SI Pump A Discharge To Hot Leg Injection valve — OPEN • EM HIS—8802A	a. PERFORM the following: 1) Locally close Feeder Breaker to EMHV8802A A SI Pump Discharge To Hot Leg Injection ISO • NG01BGR3 2) PLACE Power Lockout For EM HIS—8802A in NON ISO position: • EM HIS—8802AA 3) OPEN SI Pump A Discharge To Hot Leg Injection valve: • EM HIS—8802A 4) PLACE Power Lockout For EM HIS—8802A in ISO position: • EM HIS—8802AA

(Step A6. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 4 of 5)
SI Pump Injection To Hot Legs

Step A6. (continued from previous page)

- SI Pump B:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. CHECK SI Pump B Discharge To Hot Leg Injection valve - OPEN • EM HIS-8802B	a. PERFORM the following: 1) Locally close Feeder Breaker to EMHV8802B B SI Pump Discharge To Hot Leg Injection ISO • NG02BCR2 2) PLACE Power Lockout For EM HIS-8802B in NON ISO position: • EM HIS-8802BA 3) OPEN SI Pump B Discharge To Hot Leg Injection valve: • EM HIS-8802B 4) PLACE Power Lockout For EM HIS-8802B in ISO position: • EM HIS-8802BA

A7. START At Least One SI Pump:

- EM HIS-4 (SI Pump A)
- EM HIS-5 (SI Pump B)

NOTES

- IF refilling the RCS to support RHR Pump start, the goal is to refill at max rate.
- If SI flow is providing long term core cooling, the goal is to match the flow curve per the appropriate figure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A8. CHECK SI Pump Flow Being Used For Core Cooling	Go To Step A10.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 5 of 5)
SI Pump Injection To Hot Legs

A9. DETERMINE Adequate Flow Requirements:

a. Refer To Figure 6, Fill And Spill Decay Heat Removal Curve, to determine adequate flow requirements.

b. MONITOR SI Pump Disch Flow:

- EM FI—918
- EM FI—922

c. DISPATCH operator to locally throttle makeup flow:

- EMHV8802A

  OR

- EMHV8802B

d. TREND RWST Level changes to determine flow rate as necessary.

e. Locally THROTTLE flow as necessary to match Figure 6, Fill And Spill Decay Heat Removal Curve.

- EMHV8802A

  OR

- EMHV8802B

A10. Return To Applicable Step In The Body Of The Procedure

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 1 of 5)
SI Pump Injection To Cold Legs

B1. DETERMINE Appropriate SI Pump To Be Started:

a. At least One SI Pump is available

RNO: a. RESTORE One SI Pump incapable of injection per OSP-EM-00002.

b. ENSURE appropriate support conditions are met

B2. OPEN RWST To SI Pump Suction valve(s):

- BN HIS-8806A (SI Pump A)
- BN HIS-8806B (SI Pump B)

B3. OPEN SI Pump Suction valve(s):

- EM HIS-8923A (SI Pump A)
- EM HIS-8923B (SI Pump B)

B4. OPEN SI Pump Recirc To RWST valve(s):

a. OPEN the following valve(s):

- EM HIS-8814A (SI Pump A)
- EM HIS-8814B (SI Pump B)

b. CHECK BN HIS-8813 OPEN

RNO: b. PERFORM the following:

1) PLACE Power Lockout for BN HIS-8813 in NON ISO position:

- BN HIS-8813A

2) OPEN BN HIS-8813

3) PLACE Power Lockout for BN HIS-8813 in ISO position:

- BN HIS-8813A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 2 of 5)
SI Pump Injection To Cold Legs

B5. OPEN SI Pump Discharge To Cold Leg Injection valve(s):

- EM HIS—8821A (SI Pump A)
- EM HIS—8821B (SI Pump B)

B6. OPEN SI Pump Discharge To Cold Leg Injection valve:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. CHECK SI Pumps To Cold Leg Injection valve — OPEN • EM HIS—8835	a. PERFORM the following: 1) PLACE Power Lockout For EM HIS—8835 in NON ISO position: • EM HIS—8835A 2) OPEN SI Pumps To Cold Leg Injection valve: • EM HIS—8835 3) PLACE Power Lockout For EM HIS—8835 in ISO position: • EM HIS—8835A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 3 of 5)
SI Pump Injection To Cold Legs

B7. CLOSE SI Pump Discharge To Hot Leg Injection valve(s):

- SI Pump A:

  a. CHECK SI Pump A Discharge To Hot Leg Injection valve — CLOSED

  - EM HIS—8802A

  RESPONSE NOT OBTAINED:

  a. PERFORM the following:

  1) PLACE Power Lockout For EM HIS—8802A in NON ISO position:
     - EM HIS—8802AA
  2) CLOSE SI Pump A Discharge To Hot Leg Injection valve:
     - EM HIS—8802A
  3) PLACE Power Lockout For EM HIS—8802A in ISO position:
     - EM HIS—8802AA

- SI Pump B:

  a. CHECK SI Pump B Discharge To Hot Leg Injection valve — CLOSED

  - EM HIS—8802B

  RESPONSE NOT OBTAINED:

  a. PERFORM the following:

  1) PLACE Power Lockout For EM HIS—8802B in NON ISO position:
     - EM HIS—8802BA
  2) CLOSE SI Pump B Discharge To Hot Leg Injection valve:
     - EM HIS—8802B
  3) PLACE Power Lockout For EM HIS—8802B in ISO position:
     - EM HIS—8802BA

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 4 of 5)
SI Pump Injection To Cold Legs

B8. START At Least One SI Pump:

- EM HIS—4 (SI Pump A)
- EM HIS—5 (SI Pump B)

NOTES

- IF refilling the RCS to support RHR Pump start, the goal is to refill at max rate.
- If SI flow is providing long term core cooling, the goal is to match the flow curve per the appropriate figure.

B9. CHECK SI Pump Flow Being Used For Core Cooling

RESPONSE NOT OBTAINED: Go To Step B11.

B10. DETERMINE Adequate Flow Requirements:

a. Refer To Figure 6, Fill And Spill Decay Heat Removal Curve, to determine adequate flow requirements.

b. MONITOR SI Pump Disch Flow:

- EM FI—918
- EM FI—922

c. DISPATCH operator to locally throttle makeup flow:

- EMHV8835

d. TREND RWST Level changes to determine flow rate as necessary.

e. THROTTLE flow as necessary to match Figure 6, Fill And Spill Decay Heat Removal Curve.

- EMHV8835

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 5 of 5)
SI Pump Injection To Cold Legs

B11. Return To Applicable Step In The Body Of The Procedure

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C
(Page 1 of 2)
Charging Pump Injection To Cold Legs

NOTE

Compy with T/S 3.4.12 for COMS when restoring a CCP.

C1. DETERMINE Appropriate CCP To Be Started:

a. At least one Charging Pump is available

Response Not Obtained: a. RESTORE ONE CCP incapable of injection per OSP-BG-00002.

b. ENSURE appropriate support conditions are met

C2. OPEN CCP Suction From RWST valve(s):

- BN HIS-112D
- BN HIS-112E

C3. CLOSE Both VCT Outlet valves:

- BG HIS-112B
- BG HIS-112C

C4. CLOSE Charging Pumps To Regen HX Containment Isolation valve(s):

- BG HIS-8105
- BG HIS-8106

C5. OPEN Boron Injection Header Inlet valve(s):

- EM HIS-8803A
- EM HIS-8803B

C6. OPEN Boron Injection Header Outlet valve(s):

- EM HIS-8801A
- EM HIS-8801B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT C
(Page 2 of 2)
Charging Pump Injection To Cold Legs

C7. START At Least One CCP:

- BG HIS—1A (CCP A)
- BG HIS—2A (CCP B)

NOTES

- IF refilling the RCS to support RHR Pump start, the goal is to refill at max rate.
- If Charging flow is providing long term core cooling, the goal is to match the flow curve per the appropriate figure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C8.	**CHECK Charging Pump Flow Being Used For Core Cooling**	Go To Step C10.
C9.	**DETERMINE Adequate Flow Requirements:**	
	a. Refer To Figure 6, Fill And Spill Decay Heat Removal Curve, to determine adequate flow requirements.	
	b. MONITOR CCP To Boron Inj Flow: • EM FI—917A • EM FI—917B	
	c. THROTTLE Boron Injection Header Inlet valves as necessary to match Figure 6, Fill And Spill Decay Heat Removal Curve: • EM HIS—8803A • EM HIS—8803B	
C10.	**Return To Applicable Step In The Body Of The Procedure**	

-END-

ATTACHMENT D
(Page 1 of 1)
Core Exit Thermocouple Temperature Monitoring

D1. RECORD At Least One Operable Train A AND Train B Core Exit Thermocouple And The Core Location Every 15 Minutes:

TIME	TRAIN A CET (GD TC1)		TRAIN B CET (GD TC2)	
----	Temp °F	Location	Temp °F	Location

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 1 of 8)
RWST Gravity Feed

CAUTION

RWST Gravity Feed to the RCS must be established prior to the onset of core boiling to ensure RWST head pressure is sufficient to provide flow to the RCS.

NOTES

- Fill And Spill Decay Heat Removal Curve provides flow required to maintain subcooling and prevent boiling.
- Feed And Bleed Decay Heat Removal Curve provides the minimum flow required to match RCS boil—off and prevent core uncovery.

E1. DETERMINE Step Transition:

- *(Preferred)* To Align RWST Gravity Feed Via The RHR A Suction Piping, THEN Go To Step E2
- *(Preferred)* To Align RWST Gravity Feed Via The RHR B Suction Piping, THEN Go To Step E3
- To Align RWST Feed to At Least One RHR Discharge via BN8717, THEN Go To Step E4
- Other Gravity Feed Flowpaths as determined Plant Engineering Staff

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 2 of 8)
RWST Gravity Feed

NOTE

RWST gravity feed through the RHR suction piping is calculated to be approximately 3000 gpm assuming RWST level at 93%.

E2. ALIGN RWST Gravity Feed Via The RHR A Suction Piping:

a. OPEN RHR A Pump Suction valves:

- BB HV—8702A
- EJ HV—8701A

b. Refer To Figure 6, Fill And Spill Decay Heat Removal Curve, to determine flow requirements.

c. TREND RWST Level changes to determine flow rate.

d. Locally THROTTLE RWST To RHR Pump Suction valve, as follows:

- THROTTLE BNHV8812A (A RHR Pmp Rm) to match Figure 6, Fill And Spill Decay Heat Removal Curve.

  AND

- MAINTAIN Core Exit TCs less than 200°

RESPONSE NOT OBTAINED:

d. If Core Exit TCs can NOT be maintained less than 200°,
THEN PERFORM the following:

- Refer To Figure 7, Feed And Bleed Decay Heat Removal Curve, to determine adequate flow requirements.

  AND

- Locally THROTTLE BNHV8812A to match feed and bleed heat removal rate.

e. Return To applicable step in body of procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 3 of 8)
RWST Gravity Feed

NOTE

RWST gravity feed through the RHR suction piping is calculated to be approximately 3000 gpm assuming RWST level at 93%.

E3. ALIGN RWST Gravity Feed Via The RHR B Suction Piping:

a. OPEN RHR B Pump Suction valves

- BB HV—8702B
- EJ HV—8701B

b. Refer To Figure 6, Fill And Spill Decay Heat Removal Curve to determine flow requirements.

c. TREND RWST Level changes to determine flow rate.

d. Locally THROTTLE RWST To RHR Pump Suction valve, as follows:

- THROTTLE BNHV8812B (B RHR Pmp Rm) to match Figure 6, Fill And Spill Decay Heat Removal Curve.

  AND

- MAINTAIN Core Exit TCs less than 200°

RESPONSE NOT OBTAINED:

d. If Core Exit TCs can NOT be maintained less than 200°, THEN PERFORM the following:

- Refer To Figure 7, Feed And Bleed Decay Heat Removal Curve, to determine adequate flow requirements

  AND

- Locally THROTTLE BNHV8812B to match feed and bleed heat removal rate.

e. Return To applicable step in body of procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 4 of 8)
RWST Gravity Feed

NOTE

RWST gravity feed through BN8717 to the RHR C/L Injection line is calculated to be approximately 500 gpm assuming RWST level at 93%.

E4. ALIGN RWST Feed To At Least One RHR Cold Leg Injection Line Via BN8717

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. OPEN at least one RHR To Accumulator Injection Loop valve: • EJ HIS—8809A OR • EJ HIS—8809B	a. Locally OPEN at least one RHR To Accumulator Injection Loop valve: • EJHV8809A (N Pipe Pen Rm, Pen 82) OR • EJHV8809B (S Pipe Pen Rm, Pen 27)
b. OPEN applicable RHR Train Hot Leg Recirc valve: • EJ HIS—8716A OR • EJ HIS—8716B	b. Locally OPEN applicable RHR Train Hot Leg Recirc valve: • EJHV8716A(A RHR HX Rm) OR • EJHV8716B(B RHR HX Rm)

NOTE

BN8717, RHR SPLY TO RWST ISO is locked with a non—frangible lock. Key PA—300 can be obtained from key issue (Tag 101) with SM permission.

c. UNLOCK And OPEN BN8717, RHR SPLY TO RWST ISO (A RHR Hx Rm)

(Step E4. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 5 of 8)
RWST Gravity Feed

Step E4. (continued from previous page)

d. Refer To Figure 6, Fill And Spill Decay Heat Removal Curve, to determine flow requirements.

e. TREND RWST Level changes to determine Flow Rate.

f. MONITOR RHR Cold Leg Injection Flow:

- EJ FI—618

  OR

- EJ FI—619

g. Locally THROTTLE BN8717, RHR SPLY TO RWST ISO, as follows:

- THROTTLE flow to match Figure 6, Fill And Spill Decay Heat Removal Curve

  AND

- MAINTAIN Core Exit TCs less than 200°

RESPONSE NOT OBTAINED:

g. If Core Exit TCs can NOT be maintained less than 200°,
THEN PERFORM the following:

- Refer To Figure 7, Feed And Bleed Decay Heat Removal Curve, to determine adequate flow requirements

  AND

- Locally THROTTLE BN8717 to match feed and bleed heat removal rate.

h. Return To applicable step in body of procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 6 of 8)
RWST Gravity Feed

NOTE

RWST gravity feed through BN8717 to the the RHR H/L Injection line is calculated to be approximately 500 gpm assuming RWST level at 93%.

E5. ALIGN RWST Feed To RHR Hot Leg Injection Line Via BN8717

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. OPEN EJ HIS—8840, RHR Hot Leg Recirc valve as follows: 1) PLACE Power Lockout For EJ HIS—8840 in NON ISO position: • EJ HIS—8840A 2) OPEN RHR Hot Leg Recirc valve: • EJ HIS—8840 3) PLACE Power Lockout For EJ HIS—8840 in ISO position: • EJ HIS—8840A	a. Locally OPEN RHR TRAIN A & B SI SYS HOT LEG RECIRC ISO • EJHV8840 (S Pipe Pen Rm, Pen 21)
b. CLOSE RHR Train Hot Leg Recirc valves: • EJ HIS—8716A • EJ HIS—8716B	b. Locally CLOSE RHR Train Hot Leg Recirc valves: • EJHV8716A(A RHR HX Rm) • EJHV8716B(B RHR HX Rm)

(Step E5. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 7 of 8)
RWST Gravity Feed

Step E5. (continued from previous page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. CLOSE applicable RHR To Accumulator Injection Loop valve: • EJ HIS—8809A OR • EJ HIS—8809B	c. Locally CLOSE applicable RHR To Accumulator Injection Loop valve: • EJHV8809A (N Pipe Pen Rm, Pen 82) OR • EJHV8809B (S Pipe Pen Rm, Pen 27)

NOTE

BN8717, RHR SPLY TO RWST ISO is locked with a non—frangible lock. Key PA—300 can be obtained from key issue (Tag 101) with SM permission.

d. UNLOCK And OPEN BN8717, RHR SPLY TO RWST ISO (A RHR Hx Rm)

e. Refer To Figure 6, Fill And Spill Decay Heat Removal Curve, to determine flow requirements.

f. TREND RWST Level changes to determine Flow Rate.

g. MONITOR RHR Hot Leg Recirc Flow:

- EJ FI—988

(Step E5. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT E
(Page 8 of 8)
RWST Gravity Feed

Step E5. (continued from previous page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
h. Locally THROTTLE BN8717, RHR SPLY TO RWST ISO, as follows: • THROTTLE flow to match Figure 6, Fill And Spill Decay Heat Removal Curve AND • MAINTAIN Core Exit TCs less than 200°	h. If Core Exit TCs can NOT be maintained less than 200°, THEN PERFORM the following: • Refer to Figure 7, Feed And Bleed Decay Heat Removal Curve, to determine adequate flow requirements AND • Locally THROTTLE BN8717 to match feed and bleed heat removal rate.
i. Return To applicable step in body of procedure.	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT F
(Page 1 of 2)
Venting The RHR System

NOTES

- If adequate time to completely vent the RHR system is NOT available, air can be swept out of the RHR lines by filling the RCS to 27 Inches and running an RHR pump at a flowrate greater than 3850 GPM.
- "Affected" refers to the RHR train that has experienced air entrainment.

F1. VENT Affected RHR Suction Headers At The Following Locations:

- Train A:
  - AB 2000 RM 1323, North piping penetration room:
    - EJV0047 (suction)
    - EJV0048 (suction)
- Train B:
  - AB 2000 RM 1322, South piping penetration room:
    - EJV0051 (suction)
    - EJV0052 (suction)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
F2.	**CHECK Air Present When Suction Header Vent Valves Were Opened**	Go To Procedure Steps, Step 16.

ATTACHMENT F
(Page 2 of 2)
Venting The RHR System

F3. PERFORM Static Vent Of Affected RHR Train As Necessary:

a. Train A:

- AB 2000 RM 1323, North piping penetration room:
  - EJV0047 (suction)
  - EJV0048 (suction)
  - EJV0053 (discharge)
  - EJV0026 (discharge)
  - EJV0196 (discharge)
- AB 2000 RM 1310, RHR HX room A:
  - EJV0146 and EJV0204
  - EJV0158 and EJV0208
  - EJV0140 and EJV0205
  - EJV0193 and EJV0201

b. Train B:

- AB 2000 RM 1322, South piping penetration room:
  - EJV0051 (suction)
  - EJV0052 (suction)
- AB 2000 RM 1309, RHR HX room B:
  - EJV0104 and EJV0206
  - EJV0098 and EJV0207
  - EJV0094 and EJV0198

F4. Go To Procedure Steps, Step 16.

-END-

Figure 1
Mid-Loop Time To Boil (Hot Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

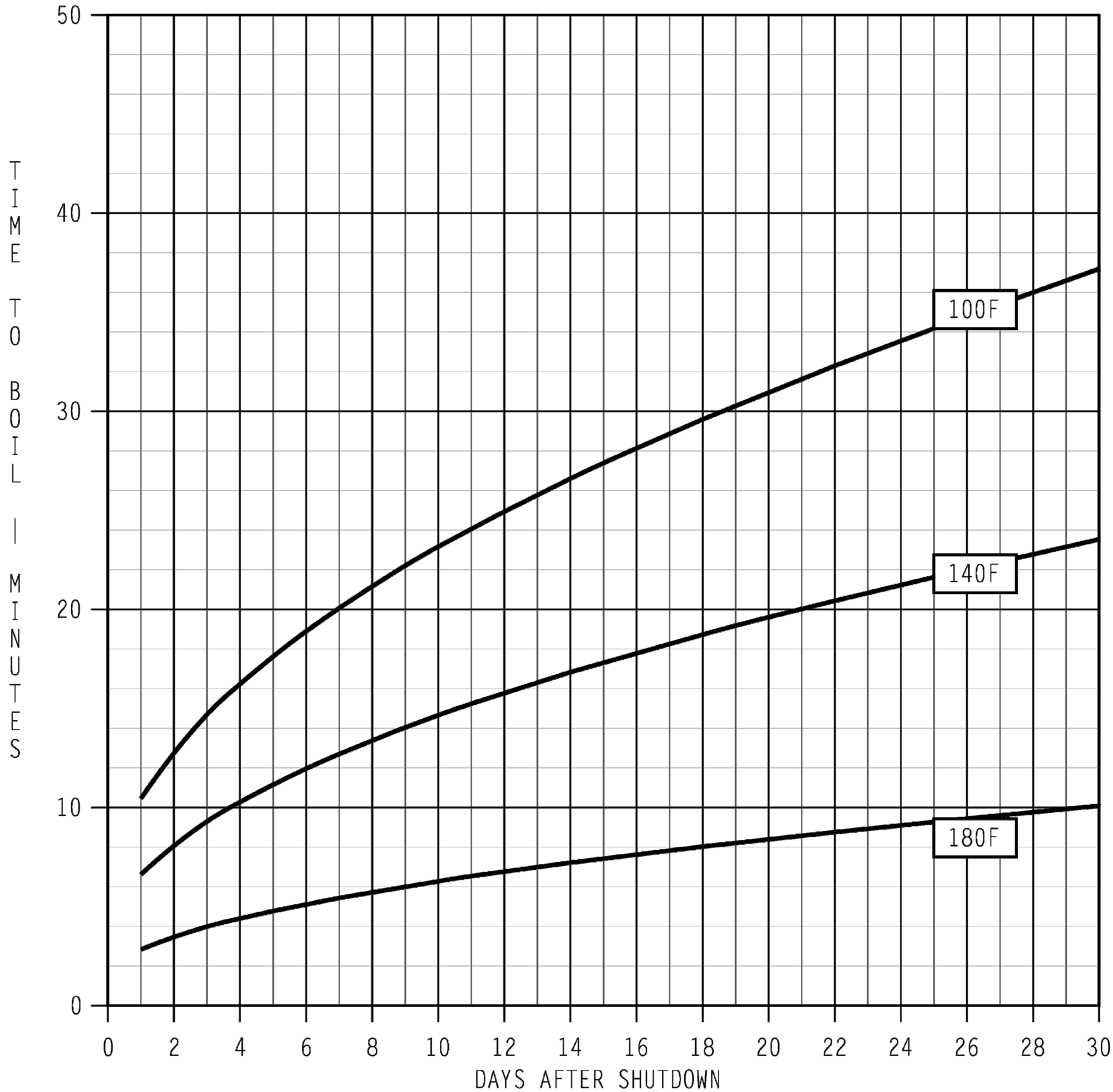


Figure 2
Reduced Inventory Time To Boil (Hot Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

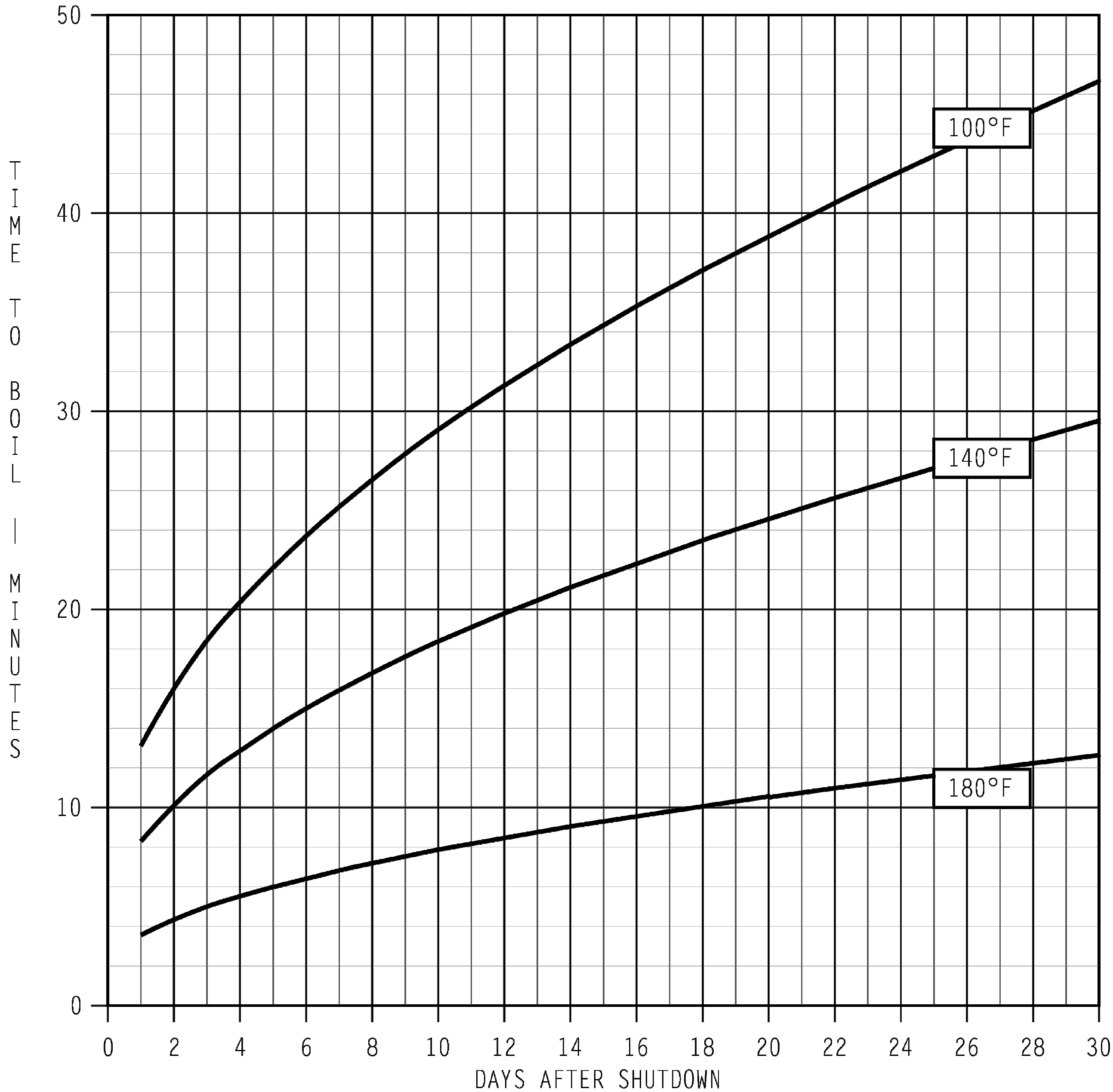


Figure 3
Mid-Loop Time To Boil (Cold Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

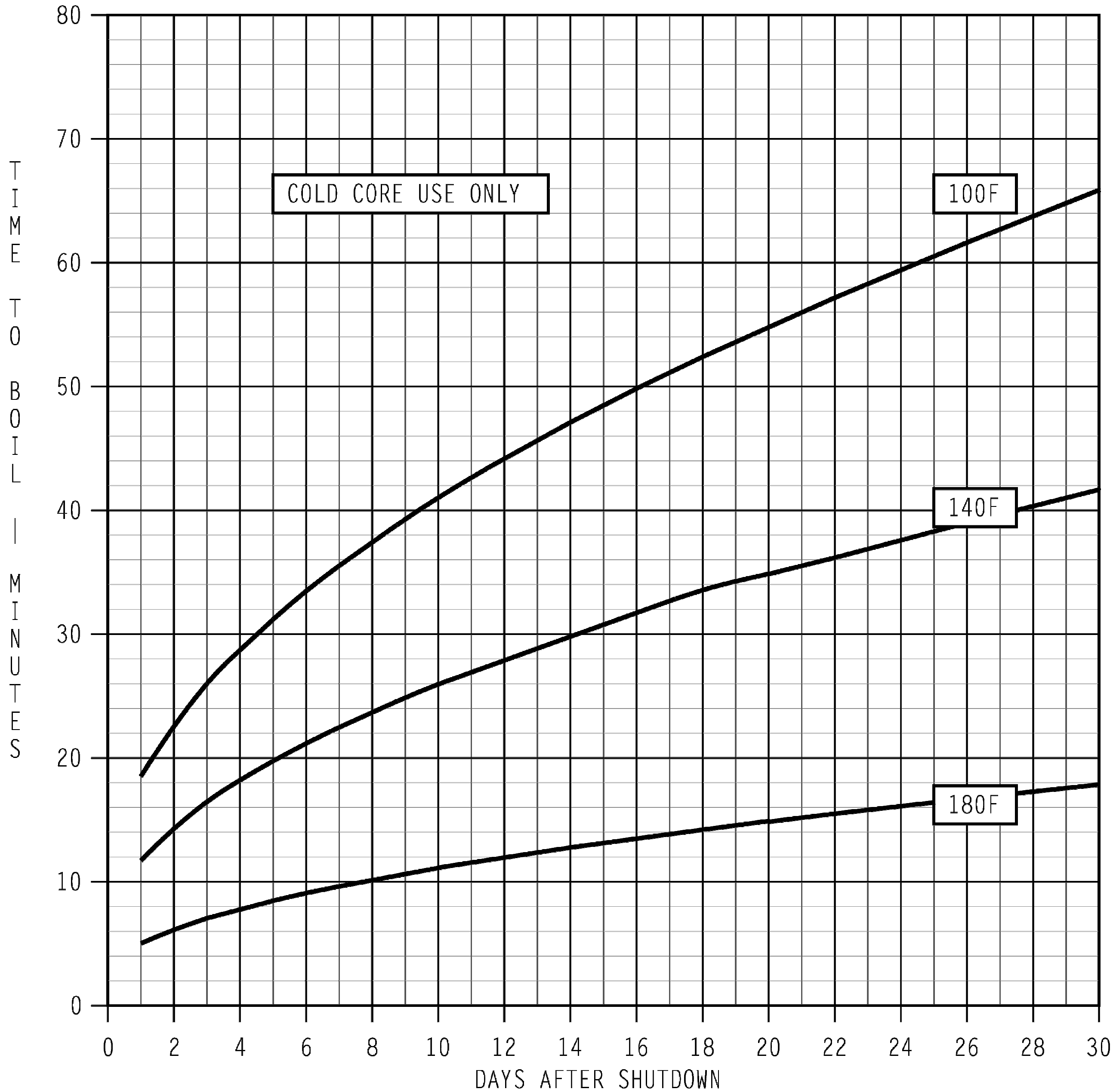


Figure 4
Reduced Inventory Time To Boil (Cold Core)

CAUTION: The best estimate of Time To Boil is provided by real time T-Boil calculations. The graph below provides conservative time to boil values and should be used if no other information is available.

NOTE: The temperatures (e.g.100°F,140°,180°F) for different curves refer to the initial RCS temperature at the time loss of RHR occurs.

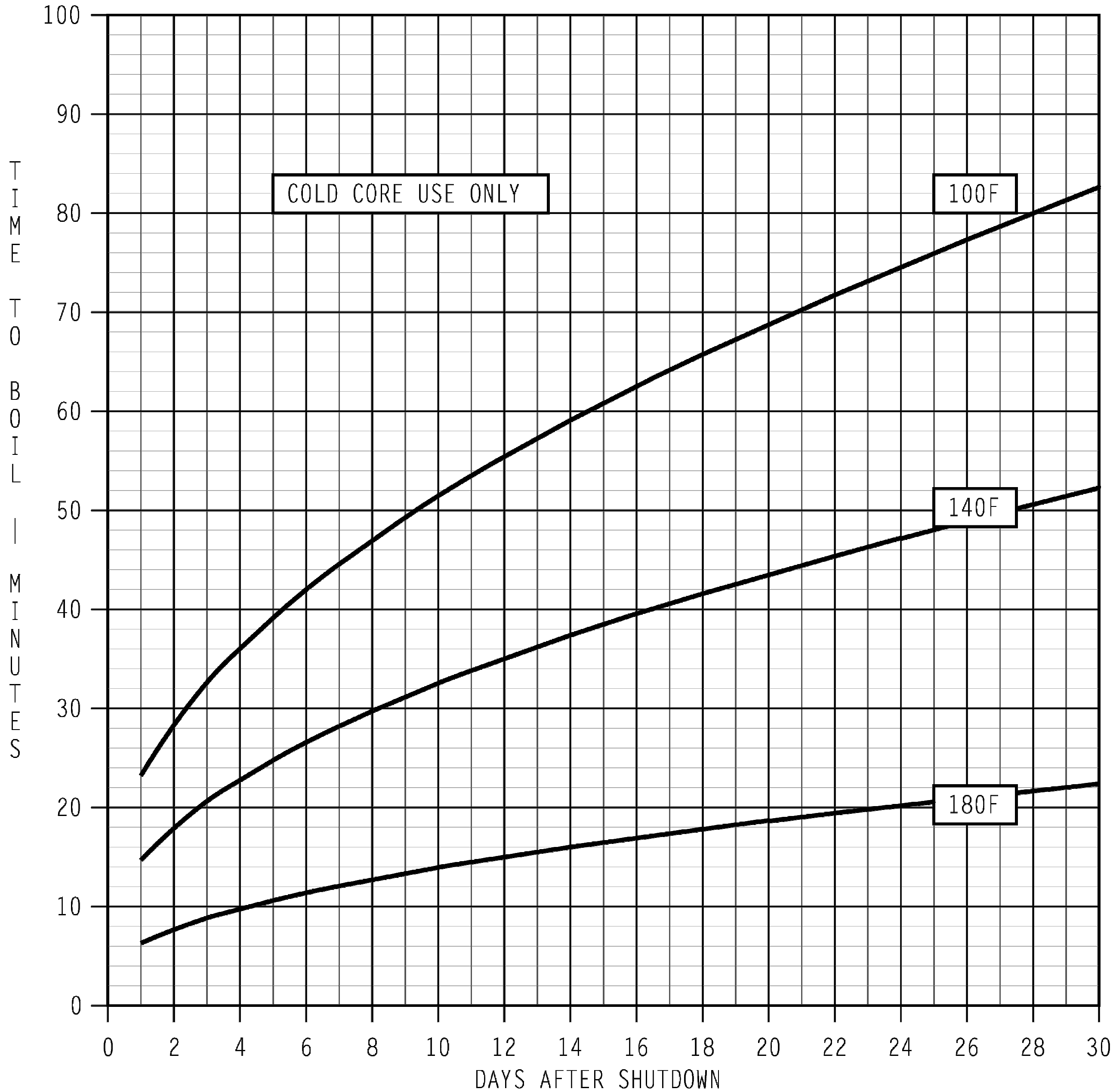


Figure 5
Minimum Hot Leg Level Vs RHR Flow

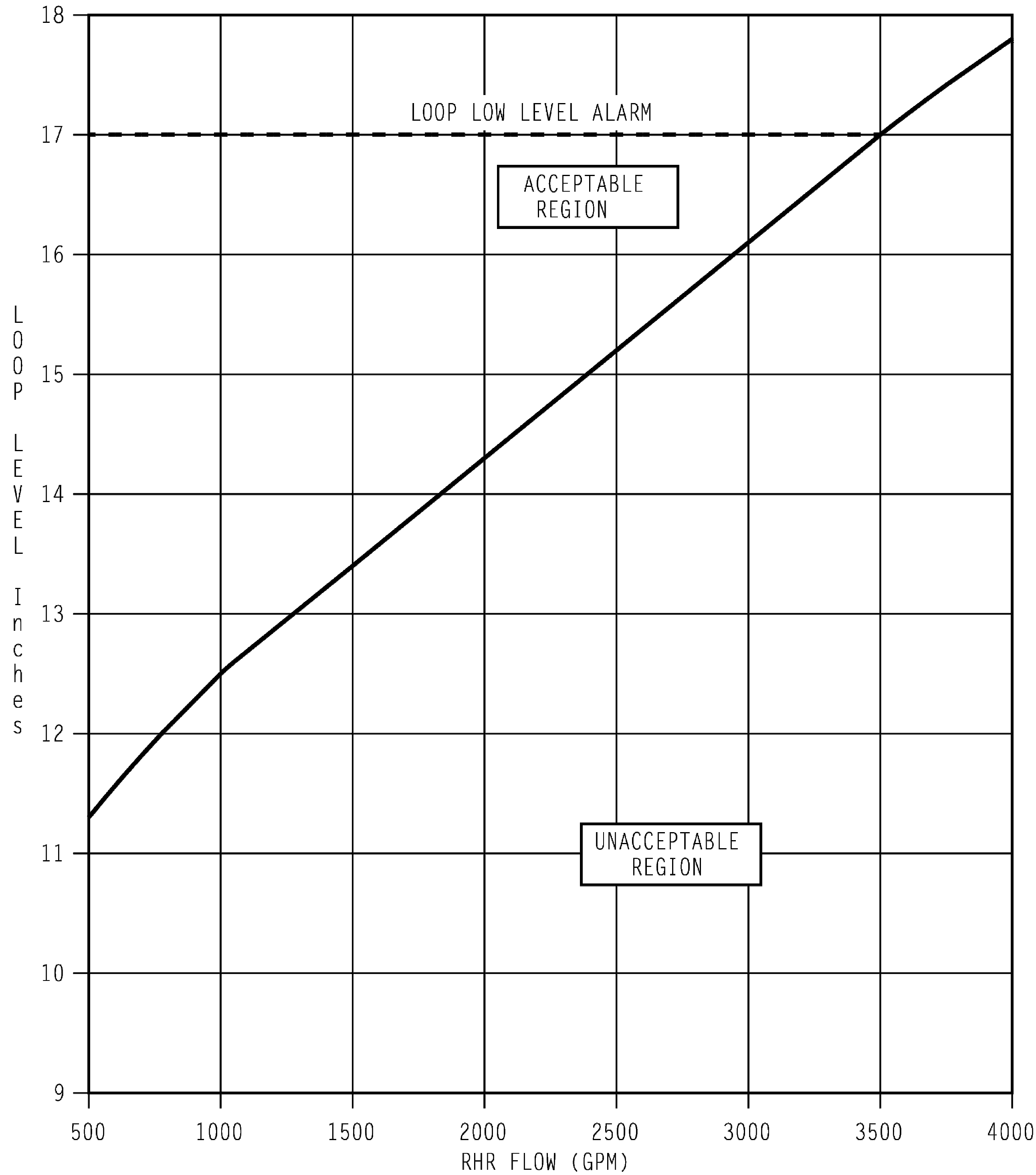


Figure 6
Fill And Spill Decay Heat Removal Curve

Note: The Fill and Spill Curve is developed to provide the flowrate needed to maintain the RCS Subcooled. Curve assumes RWST temperature of 100°F and Core Exit temperature of 200°F.

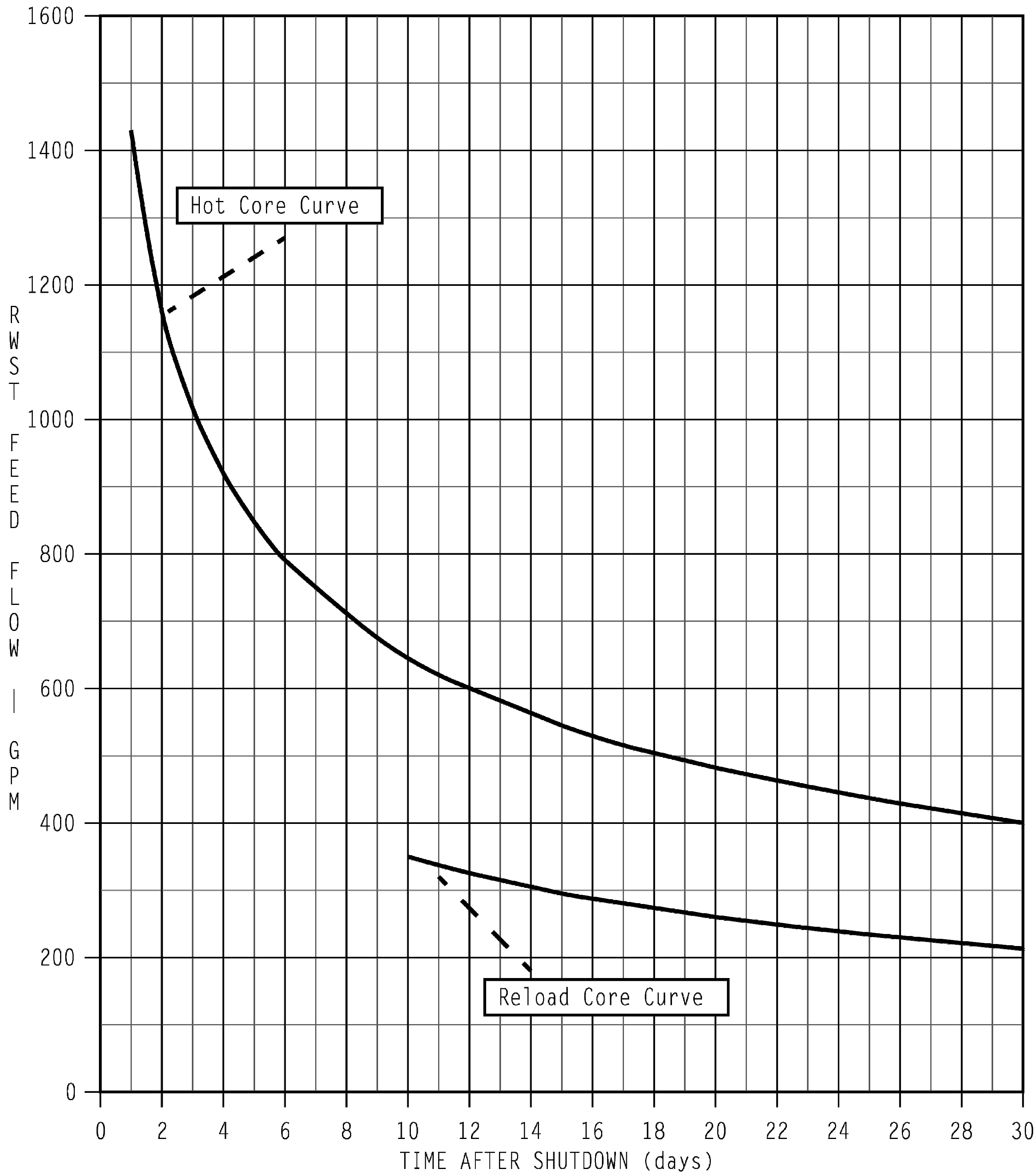
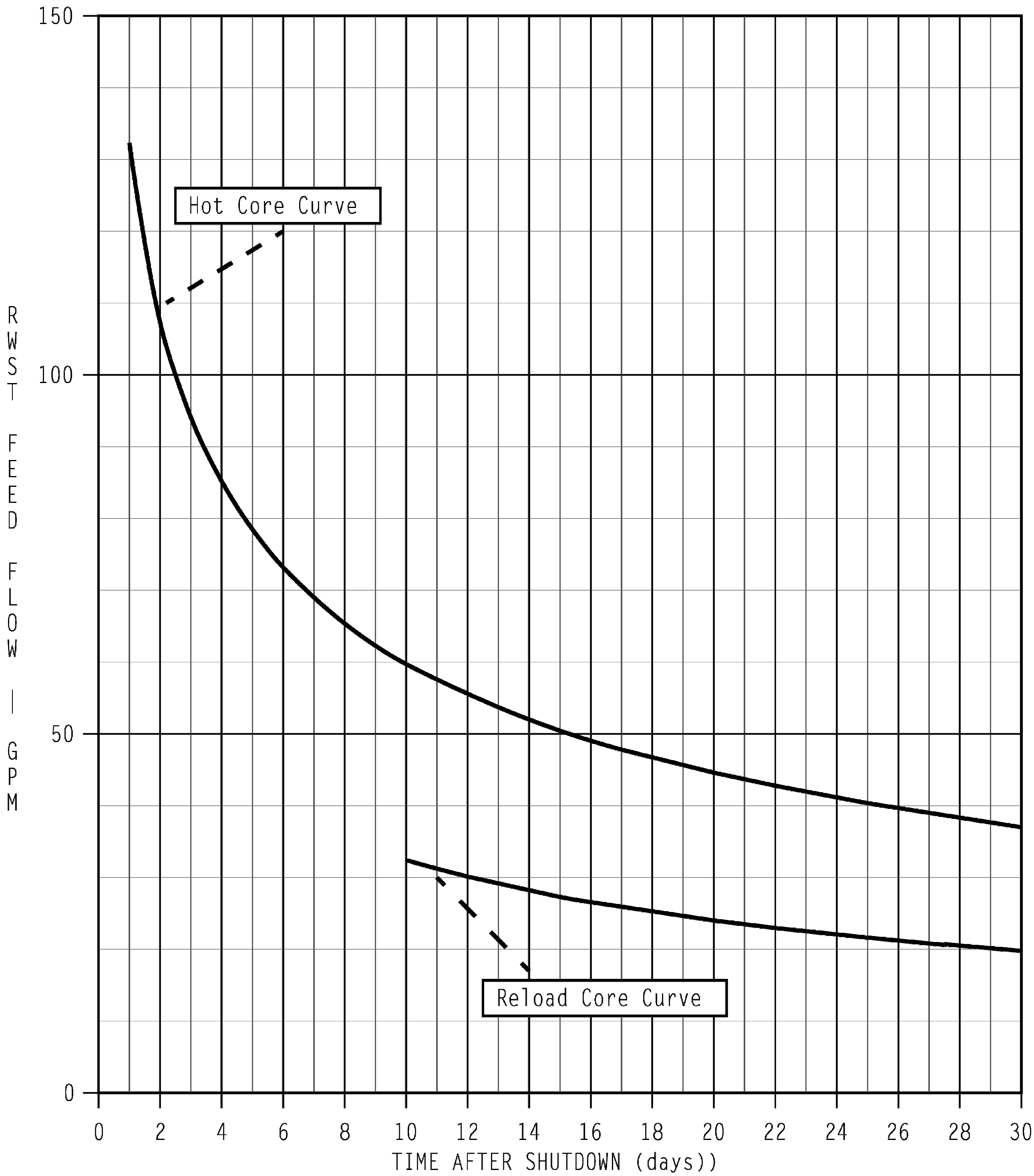


Figure 7
Feed And Bleed Decay Heat Removal Curve

Note: The Feed and Bleed Curve is developed to provide the flowrate needed to maintain the core covered (saturated). Curve assumes RWST temperature of 100°F and Core Exit temperature of 200°F.



Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
Pressurizer Relief/Quench Tank	Group #	1		
	K/A #	007 A2.04		
	Importance Rating	2.9		
Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overpressurization of the waste gas vent header				

Question # 87

The Plant is in MODE 4.

- While in the process of putting a nitrogen blanket on the PRT, a regulator fails causing the waste gas header pressure and PRT pressure to rise.
- Annunciator 34E, PRT Pressure Hi, is LIT.
- PRT Pressure is 10 psig and rising.

The CRS should direct which section of OTN-BB-00004, Pressure Relief Tank, to mitigate the event?

A. Section 5.3, Venting PRT Pressure

B. Section 5.4, PRT Hydrogen / Fission Gas Removal

C. Section 5.5, PRT Venting to Auxiliary Building

D. Section 5.8, PRT Cooling by Spraying

Answer: A

Explanation:

Section 5.3 is the appropriate section to vent the PRT in these conditions.

Section 5.4 is used normally after the PZR PORVS are opened during a plant cooldown. This is a plausible method of lower PRT if the candidate does not know the purpose of this procedure section.

Section 5.5 is used when "necessary to open PRT to atmosphere for maintenance" per a note in the beginning of the section but is plausible as it is a method to vent the PRT.

Section 5.8 is plausible if the candidate believes that cooling the PRT by spraying it will lower the pressure enough to mitigate the event. Nitrogen is a noncondensible and still be added to the PRT, therefore spray cooling would not mitigate the event.

Per M-22BB02 and M-22KH01, steps 5.3.5 and 5.3.6 isolates the incoming nitrogen supply by closing one valve on each print. The vent path through radwaste is still available as shown on print M-22BB02 and M-22HA03 by opening HAV0133 and flowing to Gas Decay Tank #7 or Tank #8 (print M-22HA03)

Note: a nitrogen failure was the only physical way to cause both the waste gas header pressure to rise and PRT pressure to rise. The event is not service gas event it is a pressurization of the waste gas header

A. Correct - see above explanation
B. Incorrect – see above explanation
C. Incorrect – see above explanation
D. Incorrect – see above explanation

Technical Reference(s):

1. OTN-BB-00004, Pressure Relief Tank, Rev 37
2. M-22BB02 P&ID RCS sheet 2, Rev 33
3. M-22HA03, P&ID Gaseous Radwaste System, Rev 8
4. M-22KH01, P&ID Service Gas System, Rev 28
5. OTA-RK-00018, ADD 34E. PRT Pressure High, Rev 0

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #9, RCS, Objective B and U:

B DESCRIBE the purpose and operation of the following RCS components to include interlocks, controller operations and power supply:

9. Pressurizer Relief Tank (PRT)
10. All interfacing system connections

U. EXPLAIN the precautions, limitations and bases for the following processes/conditions associated with OTN-BB-00004, "Pressurizer Relief Tank":

1. PRT TEMP HI alarm
2. Normal tank level
3. Nitrogen blanket
4. PRT venting for refueling

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

Revised question per NRC Comments. Per NRC, it is acceptable to write question to part b of the k/a.

SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

OTN-BB-00004

PRESSURIZER RELIEF TANK

ADMINISTRATIVE CORRECTION Revision 037

PRESSURIZER RELIEF TANK

TABLE OF CONTENTS

PRESSURIZER RELIEF TANK

1.0 PURPOSE

Provide instructions to fill, drain, vent and cool Pressurizer Relief Tank (PRT).

2.0 SCOPE

This procedure is used for:

- Normal PRT operation
- Hydrogen /fission gas atmosphere removal
- Oxygen atmosphere removal
- Cooling PRT

3.0 PRECAUTIONS AND LIMITATIONS

3.1. PRT alarms:

Alarm	Setpoint
High Pressure	6 psig
High Level	81%
Low Level	64%
High Temperature	115°F

3.2. PRT rupture disc is designed to relieve in a range between 86 to 100 psig.

3.3. PRT high temperature alarm could indicate RCS leakage or lifting of a PORV. High temperature alarm source should be determined prior to taking corrective actions.

3.4. PRT level should be maintained in 64% to 81%, except when draining or purging during a refueling outage.

3.5. PRT level must be maintained > 19% until RCS is degasified to prevent PRT atmosphere going to RCS.

3.6. A nitrogen blanket should be maintained in PRT to prevent formation of an explosive hydrogen oxygen mixture.

3.7. Unless hydrogen concentration is verified less than 4%, draining PRT to normal containment sump should be restricted to emergency conditions. [Ref: 6.2.4]

3.8. PRT oxygen concentration must be reduced to less than 3% prior to placing hydrogen on the Volume Control Tank (VCT).

3.9. Hydrogen concentration must be reduced to less than 4% prior to venting PRT to atmosphere.

3.10. During plant STARTUP venting the PRT to the Containment Building is the preferred method because the oxygen content of the PRT cannot be easily dealt within the Waste Gas System. Venting oxygen to the Gas System requires either the manual addition of hydrogen from a portable hydrogen bottle, a dangerous evolution, or dilution with nitrogen.

3.11. RCDT Pump damage, due to cavitation, could occur if suction pressure drops below 3 psig, and / or discharge pressure below 80 psig.

3.12. Performance of this procedure removes locking devices from valves. If evolution is terminated or temporarily postponed prior to these valves being returned to their normally locked positions, they must be logged in Locked Component Deviation List.

3.13. Checklist 1, PRT Inside Containment Equipment Lineup, provides system normal operating lineup inside containment.

3.14. Checklist 2, PRT Outside Containment Equipment Lineup, provides system normal operating lineup in Auxiliary Building.

3.15. Checklist 3, PRT Main Control Board Equipment Lineup, provides system normal operating lineup in Main Control Room

3.16. Performance of all procedure instructions sections may not be required.

4.0 PREREQUISITES

None

-END OF SECTION-

5.0 PROCEDURE INSTRUCTIONS

5.1. Lowering PRT Level with RCDT Pumps

NOTE

The PRT level should be maintained above 19% until RCS has been degasified to prevent PRT atmosphere going to RCS

5.1.1. *Radwaste* - ENSURE RCDT Pump switches are in PULL TO LOCK: (HB115)

- HBHS/1003A, RCDT PMP A HAND SW
- HBHS/1003B, RCDT PMP B HAND SW

5.1.2. *Radwaste* - CLOSE HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.1.3. *Radwaste* - CLOSE HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.1.4. Using BB HIS-8031, PRT TO RCDT, OPEN BBHV8031. (RL021)

5.1.5. Using HB HIS-7176, RCDT HX OUTLET INNER CTMT ISO VLV, ENSURE OPEN HBHV7176. (RL021)

5.1.6. Using HB HIS-7136, RCDT HX OUTLET OUTER CTMT ISO VLV, ENSURE OPEN HBHV7136. (RL021)

NOTE

Steps 5.1.7 through 5.1.9 are NOT required to be performed when the purpose of lowering level is to lower PRT pressure.

5.1.7. IF required, using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.1.8. IF required, using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.1.9. IF required, ENSURE BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, is set to control at approximately 3 and 4 psig. BBPCV8034 may be adjusted as necessary while in use to maintain PRT pressure between 3 and 6 psig. (1322)

5.1.10. ESTABLISH communications between Radwaste Control Room, and Main Control Room.

5.1.11. *Radwaste* - PLACE Level Controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and 50% – 100%OPEN.

NOTE

Unless level and flow jumpers are installed per PM0824600:

- For the RCDT pump to run in AUTO, RCDT level must be greater than 20%.
- The RCDT pump switch must be held in the RUN position until flow has been established.

Steps 5.1.12 and 5.1.13 may be performed simultaneously.

5.1.12. *Radwaste* - START one RCDT Pump: (HB115)

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A

  - OR -

- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.1.13. *Radwaste* - CONTROL HBLC1003, RCDT LEVEL CONTROLLER, to achieve:

- Level 64% - 81%
- Pressure 2 - 6 psig

5.1.14. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.1.15. *Radwaste* - WHEN desired level / pressure is reached, OR Low Level Alarm is received, STOP RCDT Pumps:

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.1.16. *Radwaste* - PLACE level controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and CLOSE. (HB115)

5.1.17. IF desired PRT level / pressure has been reached, Go To Step 5.1.26.

5.1.18. IF PRT level is to be reduced below Low Level Alarm:

- STATION an operator at RCDT Pumps to monitor pump performance.
- ESTABLISH communications between RCDT Pumps, Radwaste Control Room, and Main Control Room.

NOTE

Pump cavitation are indicated by wildly fluctuating suction or discharge pressure, suction pressure below 3 psig, and / or discharge pressure below 80 psig.

Pump discharge valves are locked throttled to maintain RCDT recirculation flow less than 140 GPM.

As PRT level is decreased, associated RCDT Pump discharge valve may need to be throttled to prevent pump cavitation.

5.1.19. *Radwaste* - START one RCDT Pump: (HB115)

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A

  - OR -

- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.1.20. MONITOR RCDT Pump suction and discharge pressure and listen for cavitation.

5.1.21. IF cavitation occurs as PRT level decreases, THROTTLE (do NOT close) associated discharge valve until it is eliminated:

- HB7134A, RCDT PMP A DISCH THROT VLV
- HB7134B, RCDT PMP B DISCH THROT VLV

CAUTION

Lowering PRT level below 19% prior to RCS degasification may result in PRT atmosphere going to RCS

5.1.22. *Radwaste* - CONTROL HBLC1003, RCDT LEVEL CONTROLLER, to achieve:

- Level 5% - 81%
- Pressure 2 - 6 psig

5.1.23. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.1.24. *Radwaste* - WHEN desired level is reached, STOP RCDT Pumps:

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.1.25. *Radwaste* - PLACE level controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and CLOSE. (HB115)

5.1.26. Using BB HIS-8031, PRT TO RCDT, CLOSE BBHV8031. (RL021)

5.1.27. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, ENSURE CLOSED BBHV8026. (RL021)

5.1.28. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, ENSURE CLOSED BBHV8027. (RL021)

5.1.29. IF used, ENSURE BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, is set to control at approximately 3 to 4 psig. (1322)

5.1.30. *Radwaste* - OPEN HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.1.31. *Radwaste* - OPEN HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.1.32. IF RCDT discharge throttle valves were repositioned, PERFORM Checklist 4, PRT Drain Via RCDT Pumps Restoration.

-END OF SECTION-

5.2. Raising PRT Level

NOTE

Raising PRT level in accordance with this Section and venting PRT pressure in accordance with Section 5.3 may be performed concurrently.

5.2.1. IF RCS will be less than PRT pressure anytime during level change: [Ref: 6.2.1 and 6.2.2]

a. UNLOCK and CLOSE BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I) [Ref: 6.2.1 and 6.2.2]

NOTE

Using the Locked Component Deviation List is the method used in this procedure to perform independent verification for restoration of locked components.

b. PLACE BBV0065 in Locked Component Deviation List.

c. Using BG HC-123, EXCESS LETDOWN FLOW CONTROL VALVE, ENSURE CLOSED BGHCV0123. (RL021) [Ref: 6.2.1 and 6.2.2]

5.2.2. Using BB HIS-8045, REACTOR M/U WTR TO PRT, OPEN BBHV8045. (RL021)

5.2.3. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.2.4. WHEN desired PRT level is obtained OR pressure indicates 40 psig, CLOSE BBHV8045.

5.2.5. OPEN and LOCK BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I)

5.2.6. REMOVE BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO, from Locked Component Deviation List.

-END OF SECTION-

5.3. Venting PRT Pressure

5.3.1. If PRT pressure increase is due to a level increase, CONSIDER performing Section 5.1 to lower PRT level which will lower pressure.

5.3.2. *Radwaste* - ENSURE a Shutdown Gas Decay Tank (GDT) is available with pressure lower than PRT pressure.

5.3.3. *Radwaste* - ENSURE Gaseous Radwaste Release is NOT in process.

5.3.4. *Radwaste* - ENSURE neither Shutdown GDT is in service.

5.3.5. CLOSE BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, by adjusting controller setpoint to 0 psig. (1322)

5.3.6. CLOSE KHV0146, NITROGEN TO PRT ISO. (1322)

5.3.7. *Radwaste* - OPEN HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117)

5.3.8. *Radwaste* - OPEN outlet valve to desired Shutdown GDT:

- HAV0140, S/D GAS DECAY TK D OUT ISO (7117)
- HAV0150, S/D GAS DECAY TK H OUT ISO (7117)

5.3.9. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.3.10. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.3.11. MONITOR pressure using control room instrumentation or computer point:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)

5.3.12. WHEN pressure is reduced to desired value:

- Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)
- Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

5.3.13. *Radwaste* - CLOSE HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117)

5.3.14. *Radwaste* - CLOSE outlet valve to Shutdown GDT:

- HAV140, SD GAS DECAY TK D OUT ISO (7117)
- HAV150, S/D GAS DECAY TK H OUT ISO (7117)

5.3.15. OPEN KHV0146, NITROGEN TO PRT ISO. (1322)

5.3.16. ADJUST BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, to control between 3 and 4 psig at local controller. (1322)

-END OF SECTION-

5.4. PRT Hydrogen / Fission Gas Removal

NOTE

This section is normally performed after Pressurizer PORVs were opened to remove hydrogen from Pressurizer steam space during shutdown and cooldown.

Chemistry should be notified at least 15 minutes prior to required PRT atmosphere sample to allow for setup.

5.4.1. *Radwaste* - ENSURE both Shutdown Gas Decay Tanks are < 2 psig.

5.4.2. *Radwaste* - ENSURE Gaseous Radwaste Release is NOT in process.

5.4.3. *Radwaste* - ENSURE neither Shutdown GDT is in service.

5.4.4. IF RCS will be less than PRT pressure anytime during venting:

a. UNLOCK and CLOSE BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I) [Ref: 6.2.1 and 6.2.2]

NOTE

Using the Locked Component Deviation List is the method used in this procedure to perform independent verification for restoration of locked components.

b. PLACE BBV0065 in Locked Component Deviation List.

c. Using BG HC-123, EXCESS LETDOWN FLOW CONTROL VALVE, ENSURE CLOSED BGHCV0123. (RL021) [Ref: 6.2.1 and 6.2.2]

5.4.5. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.4.6. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.4.7. CLOSE BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, by adjusting local controller setpoint to 0 psig. (1322)

5.4.8. CLOSE KHV0146, NITROGEN TO PRT ISO. (1322)

5.4.9. *Radwaste* - OPEN HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117)

5.4.10. *Radwaste* - OPEN HAV150, S/D GAS DECAY TK H OUT ISO. (7117)

5.4.11. *Radwaste* - INITIATE Gas Decay Tank H evacuation per RTN-HA-00200, Gas Decay Tank Evacuation.

NOTE

Annunciator 38E and 34F are expected to alarm during this evolution.

5.4.12. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.4.13. Using BB HIS-8045, REACTOR M/U WTR TO PRT, OPEN BBHV8045. (RL021)

5.4.14. WHEN PRT Level reaches 90% - 95% OR pressure indicates 40 psig, CLOSE BBHV8045.

5.4.15. REPEAT Steps 5.4.13 and 5.4.14 as necessary to attain 90% - 95% level.

5.4.16. *Radwaste* - MONITOR Shutdown GDT H pressure and PRT level and pressure:

- HAPIS1054, SHUTDOWN GAS DECAY TK H PRESS IND SW (HA116)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.4.17. *Radwaste* - WHEN PRT level is ≥ 90% AND GDT H and PRT pressure are equalized:

a. CLOSE HAV150, S/D GAS DECAY TK H OUT ISO. (7117)

b. OPEN HAV140, SD GAS DECAY TK D OUT ISO. (7117)

c. SECURE evacuating GDT H per RTN-HA-00200, Gas Decay Tank Evacuation

d. INITIATE Gas Decay Tank D evacuation per RTN-HA-00200, Gas Decay Tank Evacuation.

5.4.18. *Radwaste* - WHEN GDT D is 2 – 4 psig and PRT is 90 - 95% level, CLOSE HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117)

5.4.19. *Radwaste* - SECURE evacuation of GDT D per RTN-HA-00200, Gas Decay Tank Evacuation.

5.4.20. *Radwaste* - INITIATE evacuation of GDT H per RTN-HA-00200, Gas Decay Tank Evacuation.

5.4.21. OPEN KHV0146, NITROGEN TO PRT ISO. (1322)

5.4.22. ADJUST BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, setpoint to 30 psig. (1322)

5.4.23. *Radwaste* - CLOSE HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.4.24. *Radwaste* - CLOSE HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.4.25. Using BB HIS-8031, PRT TO RCDT, OPEN BBHV8031. (RL021)

5.4.26. Using HB HIS-7176, RCDT HX OUTLET INNER CTMT ISO VLV, ENSURE OPEN HBHV7176. (RL021)

5.4.27. Using HB HIS-7136, RCDT HX OUTLET OUTER CTMT ISO VLV, ENSURE OPEN HBHV7136. (RL021)

5.4.28. ESTABLISH communications between Radwaste Control Room, and Main Control Room.

5.4.29. *Radwaste* - PLACE Level Controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and 50% – 100%OPEN.

NOTE

Unless level and flow jumpers are installed per PM0824600:

- For the RCDT pump to run in AUTO, RCDT level must be greater than 20%.
- The RCDT pump switch must be held in the RUN position until flow has been established.

Steps 5.4.30 and 5.4.31 may be performed simultaneously.

5.4.30. *Radwaste* - START one RCDT Pump: (HB115)

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A

  - OR -

- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

NOTE

When lowering level pressure will drop below desired value. Pump down rate must be sufficiently slow to maintain a positive pressure in the PRT.

5.4.31. *Radwaste* - CONTROL HBLC1003, RCDT LEVEL CONTROLLER, to achieve:

- Level 64% - 66%
- Pressure 18 - 20 psig

5.4.32. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.4.33. *Radwaste* - WHEN desired STOP RCDT Pumps:

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.4.34. *Radwaste* - PLACE level controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and CLOSE. (HB115)

5.4.35. Using BB HIS-8031, PRT TO RCDT, CLOSE BBHV8031. (RL021)

5.4.36. When pressure in PRT is 18 psig to 20 psig:

a. CLOSE BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, by adjusting local controller setpoint to 0 psig. (1322)

b. CLOSE KHV0146, NITROGEN TO PRT ISO. (1322)

5.4.37. ESTABLISH communications between Chemistry Technician in room 1322 and Radwaste Technician in room 7117.

5.4.38. NOTIFY Chemistry to valve in sample bomb to prepare to sample PRT gas space for oxygen and hydrogen concentrations during PRT purge using CTP-ZZ-01114, Sampling of Primary and Radwaste Bomb Sample Points.

5.4.39. *Radwaste* - SECURE evacuation of GDT H per RTN-HA-00200, Gas Decay Tank Evacuation.

5.4.40. *Radwaste* - INITIATE evacuation of GDT D per RTN-HA-00200, Gas Decay Tank Evacuation.

NOTE

Sample flow should start as soon as HAV0133 is open.

5.4.41. *Radwaste* - WHEN Chemistry informs Radwaste sample bomb is aligned for sample, OPEN HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117)

5.4.42. *Radwaste* - WHEN pressure in PRT is between 3 and 5 psig:

a. ENSURE Chemistry sample is complete with sample bomb removed.

b. CLOSE HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117).

5.4.43. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)

5.4.44. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

5.4.45. IF PRT hydrogen / fission gas concentration is acceptable, Go To Step 5.4.70.

5.4.46. IF additional hydrogen / fission gas removal is necessary:

- IF RCS is depressurized, DECREASE level below 64% in PRT to dilute hydrogen / fission gas, Go To Step 5.4.47.

  \- OR -

- IF RCS is NOT depressurized, Return To Step 5.4.7.

5.4.47. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.4.48. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.4.49. OPEN KHV0146, NITROGEN TO PRT ISO. (1322)

5.4.50. ADJUST BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, setpoint to 30 psig. (1322)

5.4.51. Using BB HIS-8031, PRT TO RCDT, OPEN BBHV8031. (RL021)

5.4.52. Using HB HIS-7176, RCDT HX OUTLET INNER CTMT ISO VLV, ENSURE OPEN HBHV7176. (RL021)

5.4.53. Using HB HIS-7136, RCDT HX OUTLET OUTER CTMT ISO VLV, ENSURE OPEN HBHV7136. (RL021)

5.4.54. *Radwaste* - PLACE Level Controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and 50% – 100%OPEN.

NOTE

Unless level and flow jumpers are installed per PM0824600:

- For the RCDT pump to run in AUTO, RCDT level must be greater than 20%.
- The RCDT pump switch must be held in the RUN position until flow has been established.

Steps 5.4.55 and 5.4.56 may be performed simultaneously.

5.4.55. *Radwaste* - START one RCDT Pump: (HB115)

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A

  - OR -

- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

NOTE

When lowering level pressure will drop below desired value. Pump down rate must be sufficiently slow to maintain a positive pressure in the PRT.

5.4.56. *Radwaste* - CONTROL HBLC1003, RCDT LEVEL CONTROLLER, to achieve:

- Level 19% - 64%
- Pressure 18 - 20 psig

5.4.57. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.4.58. *Radwaste* - WHEN desired STOP RCDT Pumps:

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.4.59. *Radwaste* - PLACE level controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and CLOSE. (HB115)

5.4.60. Using BB HIS-8031, PRT TO RCDT, CLOSE BBHV8031. (RL021)

5.4.61. When pressure in PRT is 18 psig to 20 psig:

- CLOSE BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, by adjusting controller setpoint to 0 psig. (1322)
- CLOSE KHV0146, NITROGEN TO PRT ISO. (1322)

5.4.62. ESTABLISH communications between Chemistry Technician in room 1322 and Radwaste Technician in room 7117.

5.4.63. NOTIFY Chemistry to valve in sample bomb to prepare to sample PRT gas space for oxygen and hydrogen concentrations during PRT purge using CTP-ZZ-01114, Sampling of Primary and Radwaste Bomb Sample Points.

NOTE

Sample is being taken as soon as HAV0133 is open.

5.4.64. *Radwaste* - WHEN Chemistry informs Radwaste sample bomb is aligned for sample, OPEN HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117)

5.4.65. *Radwaste* - WHEN pressure in PRT is between 3 and 5 psig:

a. ENSURE Chemistry sample is complete with sample bomb removed.

b. CLOSE HAV0133, PRT TO S/D GAS DECAY TKS HDR ISO. (7117).

5.4.66. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)

5.4.67. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

5.4.68. IF PRT hydrogen / fission gas concentration is acceptable, Go To Step 5.4.70.

5.4.69. IF additional hydrogen / fission gas removal is necessary, Return To Step 5.4.7.

5.4.70. OPEN KHV0146, NITROGEN TO PRT ISO. (1322)

5.4.71. ADJUST BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, to control between 3 and 4 psig. (1322)

5.4.72. IF RCDT discharge throttle valves were repositioned, PERFORM Checklist 4, PRT Drain Via RCDT Pumps Restoration.

5.4.73. IF BBV0065 was closed in Step 5.4.4:

a. OPEN and LOCK BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I)

b. REMOVE BBV0065 from the Locked Component Deviation List.

5.4.74. Using BB HIS-8031, PRT TO RCDT, ENSURE CLOSED BBHV8031. (RL021)

5.4.75. *Radwaste* - OPEN HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.4.76. *Radwaste* - OPEN HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.4.77. *Radwaste* - CLOSE HAV0140, SD GAS DECAY TK D OUT ISO. (7117)

5.4.78. *Radwaste* - IF desired, SECURE evacuation of GDT D per RTN-HA-00200, Gas Decay Tank Evacuation.

5.4.79. *Radwaste* - IF desired, INITIATE evacuation of GDT H per RTN-HA-00200, Gas Decay Tank Evacuation.

5.4.80. IF necessary, to raise level to normal band PERFORM Section 5.2.

5.4.81. IF desired to vent PRT to atmosphere PERFORM Section 5.5.

-END OF SECTION-

5.5. PRT Venting to Auxiliary Building

NOTE

Use this Section when necessary to open PRT to atmosphere for maintenance.

5.5.1. ENSURE PRT hydrogen and fission gasses have been removed in accordance with Section 5.4.

CAUTION

When BL water pressure is greater than RCS pressure, the potential exists for water to enter the RCS through the Excess Letdown line. [Ref: 6.2.2]

5.5.2. IF RCS will be less than PRT pressure anytime during venting:

a. UNLOCK and CLOSE BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I) [Ref: 6.2.1 and 6.2.2]

NOTE

Using the Locked Component Deviation List is the method used in this procedure to perform independent verification for restoration of locked components.

b. PLACE BBV0065 in Locked Component Deviation List per.

c. Using BG HC-123, EXCESS LETDOWN FLOW CONTROL VALVE, ENSURE CLOSED BGHCV0123. (RL021) [Ref: 6.2.1 and 6.2.2]

5.5.3. NOTIFY RP to monitor airborne activity during venting to atmosphere. [Ref: 6.2.1]

5.5.4. INSTALL a Tygon hose at BBV0243, RCS PRT N2 SERV GAS UPSTRM TEST CONN, and direct the hose to the north end of the Auxiliary Building exhaust duct. (1322)

5.5.5. CLOSE BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, by adjusting the local controller setpoint to 0 psig. (1322)

5.5.6. CLOSE BBV0091, RCS PRT N2 SERV SPLY SUPSTRM ISO. (1322)

5.5.7. CLOSE KHV0146, LP N2 SPLY TO PRT ISO. (1322)

5.5.8. CHECK Auxiliary Building Exhaust System in service per OTN-GL-00001, Auxiliary Building HVAC System.

5.5.9. CHECK Ventilation System radiation monitor GTRE0021B in service.

5.5.10. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.5.11. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.5.12. OPEN BBV0243, RCS PRT N2/SERV GAS SPLY UPSTRM TEST CONN. (1322)

5.5.13. WHEN venting is complete, CLOSE BBV0243, RCS PRT N2/SERV GAS SPLY UPSTRM TEST CONN. (1322)

5.5.14. OPEN KHV0146, LP N2 SPLY TO PRT ISO. (1322)

5.5.15. ENSURE BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, is set to control between 3 and 4 psig. (1322)

5.5.16. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)

5.5.17. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

5.5.18. REMOVE the Tygon hose from BBV0243, RCS PRT N2/SERV GAS SPLY UPSTRM TEST CONN, and INSTALL the pipe cap. (1322)

5.5.19. OPEN BBV0091, RCS PRT N2 SERV SPLY UPSTRM ISO. (1322)

5.5.20. IF desired to remove oxygen from PRT, Go To Section 5.7.

5.5.21. IF BBV0065 was closed in Step 5.5.2:

a. OPEN and LOCK BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I)

b. REMOVE BBV0065 from the Locked Component Deviation List per.

-END OF SECTION-

5.6. Draining PRT via Containment Normal Sumps

5.6.1. ENSURE PRT hydrogen concentration is less than 4%.

5.6.2. ENSURE PRT level is maintained above 19% until RCS has been degasified.

5.6.3. ENSURE RCS is depressurized prior to lowering PRT level below 64%.

5.6.4. PRIOR to venting to atmosphere, ENSURE PRT hydrogen concentration is < 4%.

5.6.5. Using BB HIS-8031, PRT TO RCDT, ENSURE CLOSED BBHV8031. (RL021)

5.6.6. ENSURE BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, is set to control between 3 and 4 psig. (AB 2000 RM 1322)

5.6.7. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.6.8. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

CAUTION

Containment normal sump pumps do NOT have capacity to keep up with PRT outlet valves. Containment sump lineup should be checked and level monitored to prevent overflowing sump.

5.6.9. OPEN one PRT outlet valve to containment normal sump:

- BBHV8037A, by operating from RL021using BB HIS-8037A, PRT DRN TO CTMT NORM SUMP, **OR**
  by OPENING breaker NG01BDF1, FDR BKR TO BBHV8037A PRT OUT TO CTMT NORM SMP HV, declutching and operating manually.
- BBHV8037B, by operating from RL021using BB HIS-8037B, PRT DRN TO CTMT NORM SUMP, **OR**
  by OPENING breaker NG02BHR3, FDR BKR TO BBHV8037B PRT OUT TO CTMT NORM SMP HV, declutching and operating manually.

5.6.10. MONITOR PRT level using control room instrumentation or computer point:

- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.6.11. MONITOR Containment Normal Sump level using control room instrumentation or computer points:

- LF LI-9, CTMT NORM SUMP LEV (RL018)
- LF LI-89, CTMT NORM SUMP LEV (RL023)
- LFL0009, CTMT NORMAL SUMP A/B LEV (computer point)
- LFL0089, CTMT NORMAL SUMP A/B LEV (computer point)

5.6.12. WHEN containment normal sump level reaches 30 inches, or PRT reaches desired level, CLOSE:

- BBHV8037A, by operating from RL021using BB HIS-8037A, PRT DRN TO CTMT NORM SUMP, **OR**
  by declutching and closing manually.
- BBHV8037B, by operating from RL021using BB HIS-8037B, PRT DRN TO CTMT NORM SUMP, **OR**
  by declutching and closing manually.

5.6.13. IF additional PRT draining is desired, wait until sump level reaches a low level, and Return To Step 5.6.9.

5.6.14. IF BBHV8037A or BBHV8037B were operated manually:

- ENSURE breaker NG01BDF1, FDR BKR TO BBHV8037A PRT OUT TO CTMT NORM SMP HV, is ON.
- ENSURE breaker NG02BHR3, FDR BKR TO BBHV8037B PRT OUT TO CTMT NORM SMP HV, is ON.
- ENSURE BBHV8037A, indicates CLOSE from RL021using BB HIS-8037A, PRT DRN TO CTMT NORM SUMP.
- ENSURE BBHV8037B, indicates CLOSE from RL021using BB HIS-8037B, PRT DRN TO CTMT NORM SUMP

5.6.15. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)

5.6.16. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

-END OF SECTION-

5.7. PRT Oxygen Removal

NOTE

Use this section to remove excess oxygen from PRT when system has been opened to atmosphere during outages.

If pressurizer is aligned to PRT through an open vent path a slight pressure changes could affect mid-loop level indication. (BBLI-53A and BBLI-53B)

The RCDT can NOT be pumped down during this evolution.

CAUTION

Reactor Makeup water pressure greater than RCS pressure has resulted in inadvertent backflow through excess letdown line. [Ref: 6.2.2]

5.7.1. IF RCS will be less than PRT pressure anytime during venting:

a. UNLOCK and CLOSE BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I) [Ref: 6.2.1 and 6.2.2]

NOTE

Using the Locked Component Deviation List is the method used in this procedure to perform independent verification for restoration of locked components.

b. PLACE BBV0065 in Locked Component Deviation List per.

c. Using BG HC-123, EXCESS LETDOWN FLOW CONTROL VALVE, ENSURE CLOSED BGHCV0123. (RL021) [Ref: 6.2.1 and 6.2.2]

5.7.2. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)

5.7.3. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

5.7.4. CLOSE BBPCV8034, RCS PRT N2 SERV GAS SPLY PCV, by adjusting local controller setpoint to 0 psig. (1322)

5.7.5. CLOSE KHV0146, NITROGEN TO PRT ISO. (1322)

5.7.6. REMOVE cap and OPEN BBV0096, RCS PRT VENT. (RB 2000 C17P)

NOTE

Annunciator 38E and 34F are expected to alarm during this evolution.

5.7.7. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.7.8. Using BB HIS-8045, REACTOR M/U WTR TO PRT, OPEN BBHV8045. (RL021)

5.7.9. WHEN PRT Level reaches 90% - 95% OR pressure indicates 40 psig, CLOSE BBHV8045.

5.7.10. REPEAT Steps 5.7.8 and 5.7.9 as necessary to attain 90% - 95% level.

5.7.11. WHEN PRT is depressurized (no gas escaping from vent), OPEN KHV0146, NITROGEN TO PRT ISO. (1322)

5.7.12. ADJUST BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, to control between 18 and 20 psig. (1322)

5.7.13. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.7.14. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.7.15. INFORM Chemistry PRT vapor space is venting and REQUEST Chemistry monitor oxygen concentration at BBV0096, RCS PRT VENT. (RB 2000 C17P)

5.7.16. WHEN oxygen concentration in PRT vapor space has been reduced to < 2%, CLOSE and cap BBV0096, RCS PRT VENT. (RB 2000 C17P)

5.7.17. ADJUST BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, to control between 3 and 4 psig. (1322)

5.7.18. *Radwaste* - CLOSE HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.7.19. *Radwaste* - CLOSE HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.7.20. Using BB HIS-8031, PRT TO RCDT, OPEN BBHV8031. (RL021)

5.7.21. Using HB HIS-7176, RCDT HX OUTLET INNER CTMT ISO VLV, ENSURE OPEN HBHV7176. (RL021)

5.7.22. Using HB HIS-7136, RCDT HX OUTLET OUTER CTMT ISO VLV, ENSURE OPEN HBHV7136. (RL021)

5.7.23. ESTABLISH communications between Radwaste Control Room, and Main Control Room.

5.7.24. *Radwaste* - PLACE Level Controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and 50% – 100%OPEN.

NOTE

Unless level and flow jumpers are installed per PM0824600:

- For the RCDT pump to run in AUTO, RCDT level must be greater than 20%.
- The RCDT pump switch must be held in the RUN position until flow has been established.

Steps 5.7.25 and 5.7.26 may be performed simultaneously.

5.7.25. *Radwaste* - START one RCDT Pump: (HB115)

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A

  - OR -

- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.7.26. *Radwaste* - CONTROL HBLC1003, RCDT LEVEL CONTROLLER, to achieve:

- Level 64% - 81%
- Pressure 2 - 6 psig

5.7.27. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.7.28. *Radwaste* - WHEN desired STOP RCDT Pumps:

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.7.29. *Radwaste* - PLACE level controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and CLOSE. (HB115)

5.7.30. Using BB HIS-8031, PRT TO RCDT, CLOSE BBHV8031. (RL021)

5.7.31. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)

5.7.32. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

5.7.33. *Radwaste* - OPEN HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.7.34. *Radwaste* - OPEN HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.7.35. IF BBV0065 was closed in Step 5.7.1:

a. OPEN and LOCK BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO. (RB 2002 D07F-I)

b. Remove BBV0065, RCS LOOP 4 XOVER LEG TO CVCS EX LTDN ISO, from Locked Component Deviation List.

-END OF SECTION-

5.8. PRT Cooling by Spraying

> **NOTE**
>
> Cooling time required following a design maximum discharge is approximately one (1) hour by spraying or eight (8) hours by RCDT heat exchanger.

5.8.1. IF PRT is going to be cooled by recirculation through RCDT heat exchanger use Section 5.9.

5.8.2. *Radwaste* - ENSURE RCDT Pump switches are in PULL TO LOCK: (HB115)

- HBHS/1003A, RCDT PMP A HAND SW
- HBHS/1003B, RCDT PMP B HAND SW

5.8.3. Using BB HIS-8045, REACTOR M/U WTR TO PRT, OPEN BBHV8045. (RL021)

5.8.4. WHEN PRT Level increases to 81%, CLOSE BBHV8045.

5.8.5. *Radwaste* - CLOSE HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.8.6. *Radwaste* - CLOSE HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.8.7. Using BB HIS-8031, PRT TO RCDT, OPEN BBHV8031. (RL021)

5.8.8. Using HB HIS-7176, RCDT HX OUTLET INNER CTMT ISO VLV, ENSURE OPEN HBHV7176. (RL021)

5.8.9. Using HB HIS-7136, RCDT HX OUTLET OUTER CTMT ISO VLV, ENSURE OPEN HBHV7136. (RL021)

5.8.10. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, OPEN BBHV8026. (RL021)

5.8.11. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, OPEN BBHV8027. (RL021)

5.8.12. ENSURE BBPCV8034, RCS PRT N2/SERV GAS SPLY PCV, is set to control between 3 and 4 psig. (1322)

5.8.13. ESTABLISH communications between Radwaste Control Room, and Main Control Room.

5.8.14. *Radwaste* - PLACE Level Controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and 50% – 100%OPEN.

NOTE

Unless level and flow jumpers are installed per PM0824600:

- For the RCDT pump to run in AUTO, RCDT level must be greater than 20%.
- The RCDT pump switch must be held in the RUN position until flow has been established.

Steps 5.8.15 and 5.8.15 may be performed simultaneously.

5.8.15. *Radwaste* - START one RCDT Pump: (HB115)

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A

  - OR -
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.8.16. *Radwaste* - CONTROL HBLC1003, RCDT LEVEL CONTROLLER, to achieve:

- Level 64% - 66%
- Pressure 2 - 6 psig

5.8.17. MONITOR level and pressure using control room instrumentation or computer points:

- BBPI0469, RCS PRESSURIZER RELIEF TANK PRESSURE IND (RL021)
- REP0485A, PZR RELIEF TANK PRESS (computer point)
- BBLI0470, RCS PRESSURIZER RELIEF TANK LEVEL IND (RL021)
- REL0485A, PZR RELIEF TANK LEVEL (computer point)

5.8.18. *Radwaste* - WHEN desired level is reached, OR Low Level Alarm is received, STOP RCDT Pumps:

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.8.19. *Radwaste* - PLACE level controller HBLC1003, RCDT LEVEL CONTROLLER, in MANUAL and CLOSE. (HB115)

5.8.20. Monitor PRT temperature:

- BB TI-468, PRESSURIZER RELIEF TANK TEMP IND (RL021)
- RET0485A, PZR RELIEF TANK TEMP (computer point)

5.8.21. IF PRT temperature is ≥ 100°F, Return To Step 5.8.2.

5.8.22. Using BB HIS-8031, PRT TO RCDT, CLOSE BBHV8031. (RL021)

5.8.23. Using BB HIS-8026, PRT N2 SPLY INNER CTMT ISO VLV, CLOSE BBHV8026. (RL021)

5.8.24. Using BB HIS-8027, PRT N2 SPLY OUTER CTMT TSO VLV, CLOSE BBHV8027. (RL021)

5.8.25. *Radwaste* - OPEN HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.8.26. *Radwaste* - OPEN HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

-END OF SECTION-

5.9. PRT Cooling by RCDT Heat Exchanger

5.9.1. IF PRT is going to be cooled by spraying use Section 5.8.

NOTE

The RCDT level should be monitored when RCDT heat exchanger is used to cool PRT.

5.9.2. *Radwaste* - ENSURE RCDT Pump switches are in PULL TO LOCK: (HB115)

- HBHS/1003A, RCDT PMP A HAND SW
- HBHS/1003B, RCDT PMP B HAND SW

5.9.3. *Radwaste* - CLOSE HB-HV-7144/HS-1003F RCDT PUMPS DISCH HX RECIRC HAND CTRL VLV. (HB115)

5.9.4. *Radwaste* - CLOSE HB-HV-7127/HS-1003C, RCDT OUTLET TO RCDT PMPS HAND CTRL VLV. (HB115)

5.9.5. *Radwaste* - OPEN BBHV7141/HS-1003D, RCS RCDT HX TO PRT BBHV7141 HAND SW. (HB115)

5.9.6. Using BB HIS-8031, PRT TO RCDT, OPEN BBHV8031. (RL021)

NOTE

Unless level and flow jumpers are installed per PM0824600:

- For the RCDT pump to run in AUTO, RCDT level must be greater than 20%.
- The RCDT pump switch must be held in the RUN position until flow has been established.

5.9.7. *Radwaste* - START one RCDT Pump: (HB115)

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A

  - OR -

- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.9.8. Monitor PRT temperature:

- BB TI-468, PRESSURIZER RELIEF TANK TEMP IND (RL021)
- RET0485A, PZR RELIEF TANK TEMP (computer point)

5.9.9. *Radwaste* - WHEN PRT temperature is < 100°F, STOP RCDT Pumps:

- Using HBHS/1003A, RCDT PMP A HAND SW, for PHB02A, RCDT PMP A
- Using HBHS/1003B, RCDT PMP B HAND SW, for PHB02B, RCDT PMP B

5.9.10. Using BB HIS-8031, PRT TO RCDT, CLOSE BBHV8031. (RL021)

5.9.11. CLOSE BBHV7141/BBHS-1003D, RCS RCDT HX TO PRT BBHV7141 HAND SW,. (HB115)

5.9.12. Using HBHS/1003F, RCDT PUMPS DISCH HX HBHV7144 HAND SW, OPEN HBHV7144. (HB115)

5.9.13. Using HBHS/1003C, RCDT OUTLET HDR HBHV7127 HAND SW, OPEN HBHV7127. (HB115)

6.0 REFERENCES

6.1. Implementing

6.1.1. CTP-ZZ-01114, Sampling of Primary and Radwaste Bomb Sample Points

6.1.2. OTN-GL-00001, Auxiliary Building HVAC System

6.1.3. RTN-HA-00200, Gas Decay Tank Evacuation

6.1.4. RTN-HB-00100, Reactor Coolant Drain Tank Operation

6.2. Developmental

6.2.1. CARS 200101743, *Sig 2 – Hydrogen intrusion into RCS when venting PRT*

6.2.2. CARS 200102514, *Sig 2 – During PRT fill potential backflow into excess letdown*

6.2.3. FSAR 5.4.11, Pressurizer Relief Discharge System

6.2.4. FSAR 5.4.11.2

6.2.5. RFR 014288A

7.0 RECORDS

7.1. Records generated by this procedure are filed with appropriate work authorizing document.

7.1.1. Checklist 1, PRT Inside Containment Equipment Lineup

7.1.2. Checklist 2, PRT Outside Containment Equipment Lineup

7.1.3. Checklist 3, PRT Main Control Board Equipment Lineup

7.1.4. Checklist 4, PRT Drain Via RCDT Pumps Restoration

8.0 SUMMARY OF CHANGES

Page(s)	Section or Step Number	Description
6	5.1.15	Corrected cut and paste error to ensure both pumps are listed to secure, not just A pump. CAR 201403723

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
Emergency Core Cooling System	Group #	1		
	K/A #	006 G2.1.32		
	Importance Rating	4.0		
Ability to explain and apply system limits and precautions.				

Question # 88

Reactor Power is 100%.

0700 SI Accumulator boron concentrations and pressures are as listed:

SI Accumulator	Boron Concentration	Pressure
A	2295 ppm	614 psig
B	2292 ppm	610 psig
C	2306 ppm	601 psig
D	2320 ppm	608 psig

- 0800 ‘A’ SI Accumulator is drained, refilled and pressurized, boron concentration is 2308 ppm and pressure is 611 psig.
- 0815 ‘B’ SI Accumulator is drained, refilled and pressurized, boron concentration is 2314 ppm and pressure is 610 psig.
- 0830 ‘C’ SI Accumulator is drained, refilled and pressurized, boron concentration is 2310 ppm and pressure is 608 psig.
- 0845 ‘D’ SI Accumulator is pressurized to 610 psig.

What is the EARLIEST time T.S. 3.0.3 can be exited?

A. 0800

B. 0815

C. 0830

D. 0845

Answer: B

Explanation:
Parameters in the stem have SI Accumulators ‘B’ and ‘C’ inoperable. Based on this TS 3.0.3 is entered. ‘D’ SI Accumulator's initial parameters are within technical specifications and no action

is required for accumulator 'D'.

To exit TS 3.0.3 using the statement “Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.”

At 0800 2 SI accumulators are still inoperable, at 0815 only one SI Accumulator is inoperable and TS 3.0.3 may be exited.

A. Incorrect, per above, Plausible if the candidate believes that either only one SI accumulator remains inoperable
B. Correct, per above.
C. Incorrect, per above, Plausible if the candidate believes that all SI accumulator must be operable for TS 3.0.3 to be exited.
D. Incorrect, Plausible if the candidate believes SI Accumulator 'D' is initially out of specifications and must be addressed such that TS 3.0.3 can be exited.

Technical Reference(s):
1. Technical Specification 3.5.1
2. Technical Specification 3.0.3
3. OTN-EP-00001, ACCUMULATOR SAFETY INJECTION SYSTEM, Rev 26 Section 3.0 PRECAUTIONS AND LIMITATIONS P&L step 3.1. Four SI Accumulators shall be OPERABLE in accordance with T/S LCO 3.5.1 in MODES 1 and 2, and also in MODE 3 with RCS pressure greater than 1000 psig.

References to be provided to applicants during examination: None

Learning Objective:

T61.0110, Systems, LP #19, SAFETY INJECTION ACCUMULATORS – EP, Objective F. STATE the LCO for the SI Accumulator Technical Specifications and EXPLAIN bases for Technical Specification (T/S) 3.5.1.

T61.0110, Systems, LP-77 INTRODUCTION TO TECHNICAL SPECIFICATIONS Objective F. DEMONSTRATE the proper use of Technical Specifications, Bases and FSAR Chapter 16.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(2)

Comments:

Revised distractor D per NRC Comments

The KA is met because the SRO candidate has to understand the system limits and precautions and apply this information along with the knowledge of when TS 3.0.3 no longer applies and can be exited.

SRO ONLY due to ES401 Figure 1 of NUREG 1021 as follows:

Can question be answered solely by knowing ≤ 1 hour TS/TRM Action? **NO**

Can question be answered solely by knowing the LCO/TRM information listed "above-the-line?" **NO**

Can question be answered solely by knowing the TS Safety Limits? **NO**

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- **Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4) YES**
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	2		
Reactor Coolant Pump	**Group #**	1		
	K/A #	003 A2.02		
	Importance Rating	3.9		
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: Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP.				

Question # 89

The Plant is in MODE 3.

- OTG-ZZ-00006, Plant Cooldown Hot Standby to Cold Shutdown is in progress.
- All control and shutdown rods are fully inserted.
- "B" and "D" RCPs are in service.
- RCS Pressure is 800 psig.
- The Reactor Operator reports the following "B" RCP indications:
    - RCP #1 Seal ΔP is 300 psid
    - RCP #1 Seal Leakoff flow is 1.0 gpm
    - RCP Frame Vibration is 6 mils
    - RCP Shaft Vibration is 13 mils

Based on the above plant conditions, what is the NEXT procedure the CRS should direct?

A. OTO-BB-00002, Attachment E, "RCP Trip".

B. OTG-ZZ-00006, Attachment 2, "RCP Operations".

C. OTG-ZZ-00006, Addendum 8, "PZR Auxiliary Spray Operation".

D. OTN-BB-00003, Section 5.5, "Stopping a Reactor Coolant Pump".

Answer: A

Explanation:

Per OTO-BB-00002, with the parameters given, the vibration data is outside the limits of the OTO Attachment A step #A1 the frame vibration is greater than the 5 mil requirement and the RNO applies which directs the RCP to be secured per Attachment E of the OTO.

OTG-ZZ-00006, Attachment 2, RCP Operations, step #1.b provides a list of RCP trip criteria that may exist during a plant cooldown. Specifically, Step1.b states" RCP trip criteria include the following (Also on MCB placards):"

- *RCS Pressure less than 250 psig (MODE 5)*
- *RCP #1 Seal ΔP less than 200 psid*
- *RCP #1 Seal Leakoff less than 0.2 gpm*

As these specific criteria have not been met this is not the appropriate procedural flowpath for the CRS to direct and hence incorrect but plausible as this Attachment does include RCP trip criteria.

OTN-BB-00003, Section 5.5 is plausible as the candidate may recognize that RCP "C" #1 Seal leakoff flows are degraded but with the range of normal parameters (less than 6 gpm but greater than 0.8 gpm). Furthermore, #1 Seal ΔP is lower than normal but greater than the 200 psid that would require a manual RCP trip. Therefore, since these seal parameters don't meet the trip criteria in OTO Attachment B, "RCP Seal Parameters Abnormal", the candidate may determine that normal shutdown per OTN-BB-00003 is the correct action to direct. Additionally, the candidate may believe that the normal procedure is appropriate as the plant is in MODE 3 and the OTO is incorrect due to OTO Mode applicability.

OTG-ZZ-00006 Addendum 8 is plausible as the candidate must decide which is more important: securing the RCP or establishing Aux Spray. In the lineup in the stem, B and D RCP are providing the different pressure for PRZ Spray. If the B RCP must be secured, it may be falsely believed that the first priority is AUX Spray for pressure control making this a plausible distractor.

Note: Procedurally when the conditions given in the stem, the operators are at Step #5.2.8

A. Correct - See above explanation
B. Incorrect - See above explanation
C. Incorrect - See above explanation
D. Incorrect - See above explanation

Technical Reference(s):

1. OTN-BB-00003, Reactor Coolant Pumps, Rev 26
2. OTO-BB-00001, RCP Off-Normal, Rev 32
3. OTG-ZZ-00006, Plant Cooldown Hot Standby to Cold Shutdown, Rev 74
4. OTG-ZZ-00006, ADDENDUM 8, PZR Auxiliary Spray Operation, Rev 9

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#9, RCS Objective B, F, &Y:

B. DESCRIBE the purpose and operation of the following RCS components to include interlocks, controller operations and power supply:

1. Reactor Vessel
2. Steam Generators (Primary Side)
3. Reactor Coolant Pumps (RCPs)

F. DESCRIBE the purpose and operation of the following RCP components:

1. Thermal Barrier
2. Thermal Barrier Heat Exchanger
3. Pump Radial Bearing
4. Shaft Seal Assembly, and the No-Leak Seal

5. Lower Guide Bearing
6. Stator
7. Thrust Bearing
8. Upper Guide Bearing
9. Anti-reverse rotation device
10. Flywheel
11. Oil Lift Pump
12. Air Cooler

Y. EXPLAIN the precautions, limitations and bases for the following components/conditions associated with OTN-BB-00003, "Reactor Coolant Pumps":

1 CCW flow to Thermal Barriers with RCS temperature > 160□F
2 Seal Leakoff Valves with RCS pressure < 100 psig
3 Seal Injection flow
4 RCP starting conditions
5 #1 seal cooling
6 Mode 4 RCP operability
7 RCP starting limitations
8 Vibration
9 Seal Injection Angle Throttle Valves differential pressure (□P)
10 RCP Motor Space Heaters
11 Stopping an RCP below 48% power

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ______N/A______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis ___X__

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

Revised Question per NRC Comment – verbally agreed to writing the question to the b part of the k/a only.

CCW parameters to a RCP were not used to avoid overlap with RO question #6

SRO ONLY has the question does not ask if a TRIP of the RCP is required / not required (which is RO knowledge). The question and its 4 possible choices all concede that a "B" RCP will be required. This question makes the candidate assess the situation and determine the priorities and correct procedural flowpath to implement to stabilize the plant.

SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

OTO-BB-00002

RCP OFF-NORMAL

Revision 032

CONTINUOUS USE

A. PURPOSE

This procedure provides instructions to respond to any of the following RCP Off Normal events:

- Abnormal RCP vibration
- Abnormal RCP seal flow:
  - No. 1 Seal Leak Off Flow Low
  - No. 2 Seal Leak Off Flow High
- Abnormal flow CCW to the RCP(s)

B. SYMPTOMS OR ENTRY CONDITIONS

1) Abnormal RCP Vibration:

   a. Abnormal or unusual rise in RCP vibrations.

   b. Any of the following Control Room annunciators in alarm:

   - Annunciator 70A, RCP Vib Danger
   - Annunciator 70B, RCP Vib/Sys Alert

2) Abnormal CCW flow to the RCP(s):

   a. RCP motor bearing high temperature alarm from the computer.

   b. RCP motor stator high temperature alarm from the computer.

   c. Any of the following Control Room annunciators in alarm:

   - Annunciator 70C (71C, 72C, 73C), RCP A (B, C, D) Thrm CCW Bar Flow
   - Annunciator 74A, RCP Mtr CCW Flow Hi Lo
   - Annunciator 74C, RCP Thrm Bar CCW Flow

3) Abnormal RCP Seal Flow:

   a. Loss of Seal Injection:

      1) Rise in #1 Seal Leak Off Flow.

      2) Rising Thermal Barrier CCW outlet temperature.

      3) Any of the following Control Room annunciators in alarm:

         - Annunciator 41A, Seal Inj To RCP Flow Lo
         - Annunciator 41B, Seal Inj/RC Filter ΔP Hi
         - Annunciator 72A, RCP #1 Seal Flow Hi

   b. No. 1 Seal Leak Off Flow High:

      1) Rising Thermal Barrier CCW outlet temperature(s).

      2) Any of the following Control Room annunciators in alarm:

         - Annunciator 72A, RCP #1 Seal Flow Hi

   c. No. 1 Seal Leak Off Flow Low:

      1) Excessive leakage of the #2 Seal.

      2) RCDT Level rising at a higher than normal rate.

      3) Any of the following Control Room annunciators in alarm:

         - Annunciator 72B, RCP #1 Seal Flow Lo

   d. No. 2 Seal Leak Off Flow High:

      1) RCDT Level rising at a higher than normal rate.

      2) Any of the following Control Room annunciators in alarm:

         - Annunciator 72B, RCP #1 Seal Flow Lo
         - Annunciator 73A, RCP #2 Seal Flow Hi

C. REFERENCES

1) Implementing:

   a. ODP-ZZ-00001 Addendum 13, Shift Manager Communications

   b. OSP-BL-00001, Rx M/U Wtr Iso Vlvs W/O RCS Loops In Operation/Mode 6 Alignment

2) Developmental:

   a. COMN 3148, RCP Auxiliary Water Services

   b. COMN 5582, Reactor Coolant Pump Seal Leakage

   c. COMN 5583, Bearing Integrity

   d. COMN 5584, Check Pump Vibration Readings

   e. M-712-00068, RCP Technical Manual

   f. RFR 016805E, Change To Administrative Controls For EGHV0132

   g. ESBU-TB-93-01-R1, Westinghouse Technical Bulletin

   h. CAR 201007678, Technical Specification (T/S) Bases 3.6.3 are not being met

   i. MP 10-0009 New RCP Seals

   j. Westinghouse DW 09-010 RCP Seals

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

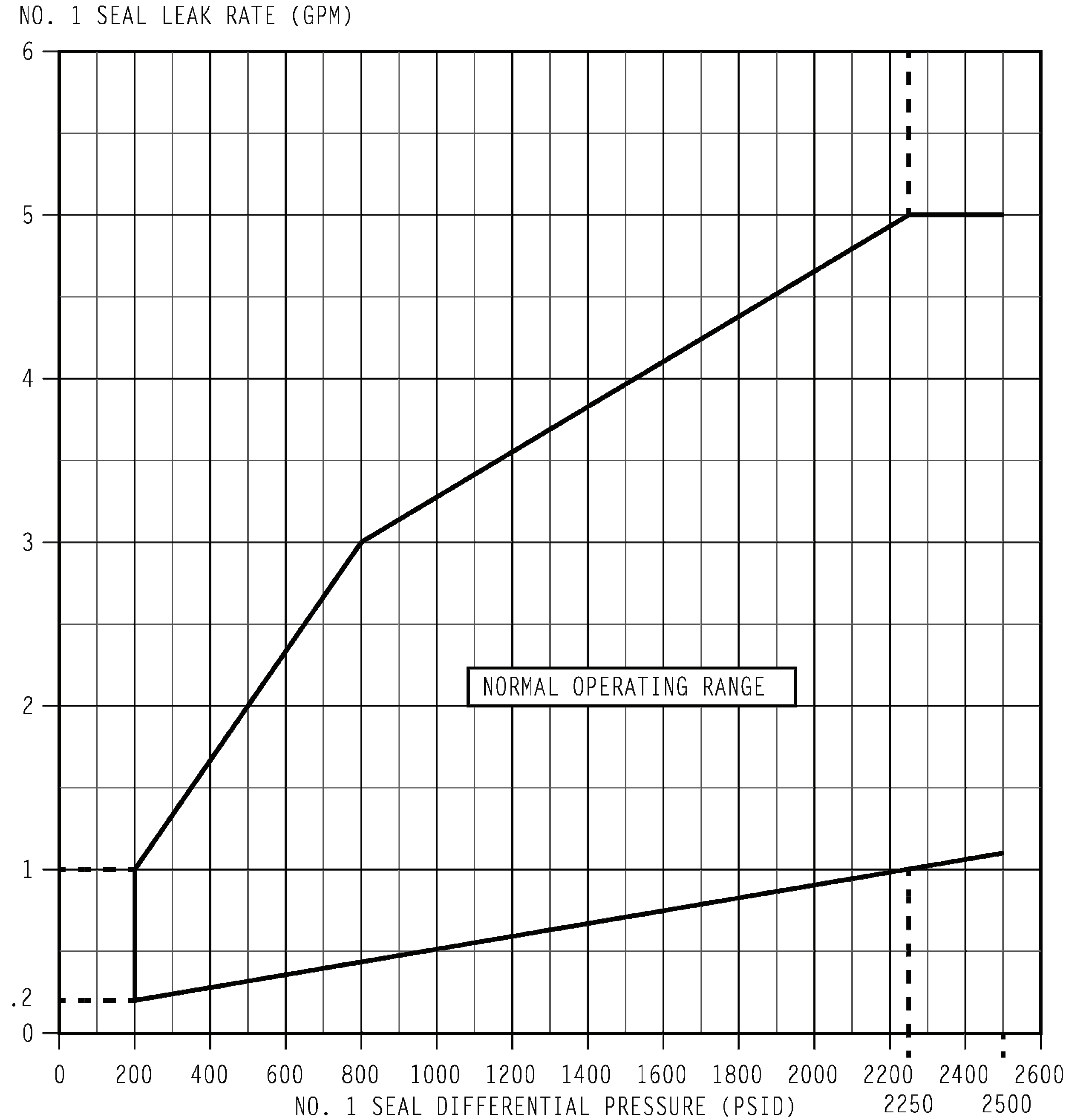
NOTE

RCPs that lose Seal Injection AND CCW To Thermal Barrier Heat Exchanger must have at least one restored within 6 minutes or the RCP MUST be secured.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**CHECK All RCPs - RUNNING**	IF Reactor power is greater than or equal to 48% (P-8 lit), THEN PERFORM the following: a. Manually TRIP the Reactor. b. IF A or B RCP is Tripped, THEN PLACE associated pressurizer spray valve controller in Manual at zero output: • BB PK-455B for A RCP <u>OR</u> • BB PK-455C for B RCP c. Go To E-0, Reactor Trip Or Safety Injection. IF Reactor power is less than 48% (P-8 extinguished), THEN Go To Attachment E, RCP Trip
2.	**Go To One Of The Following Attachments, As Applicable:** • Attachment A, RCP High Vibration • Attachment B, RCP Seal Parameters Abnormal • Attachment C, CCW To RCP Abnormal	

-END-

Figure 1
No 1 Seal Normal Operating Range



Use the following indicators if less than 400 PSID

RCP	Seal DP Indicator
A	BB PI-153A
B	BB PI-152A
C	BB PI-151A
D	BB PI-150A

Otherwise, use BG PI-120A - BG-PI-115

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 1 of 4)
RCP High Vibration

NOTE

RCP vibration can be monitored using Group Display RCPVIB or monitoring RP312, BB YI-471.

# A1.	**CHECK RCP Vibration Level:** • ALL RCPs vibration on the frame - LESS THAN 5 MILS • ALL RCPs vibration on the shaft - LESS THAN 20 MILS	PERFORM ONE of the following: • IF Reactor power is greater than or equal to 48% (P-8 lit), THEN Go To Attachment D, RCP AND Reactor Trip. OR • IF Reactor power is less than 48% (P-8 extinguished), THEN Go To Attachment E, RCP Trip.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 2 of 4)
RCP High Vibration

A2. CHECK RCP Vibration Level:

- ALL RCPs vibration on the frame - LESS THAN 3 MILS
- ALL RCPs vibration on the shaft - LESS THAN 15 MILS

PERFORM the following:

a. IF vibration on the frame rises at a rate of greater than 1 Mil/hr OR vibration on the shaft rises at a rate greater than 2 Mils/hr,
THEN PERFORM ONE of the following:

- IF Reactor power is greater than or equal to 48% (P-8 lit),
THEN Go To Attachment D, RCP AND Reactor Trip.

<u>OR</u>

- IF Reactor power is less than 48% (P-8 extinguished),
THEN Go To Attachment E, RCP Trip.

b. IF vibration on the frame rise at a rate of less than 1 Mil/hr AND vibration on the shaft rises at a rate less than 2 Mils/hr,
THEN PERFORM the following:

1) CONTINUE monitoring RCP vibration.

2) CONTACT the RCP and Vibration System Engineers to determine if securing the RCP is required.

(Step A2. continued on next page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 3 of 4)
RCP High Vibration

Step A2. (continued from previous page)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		c. IF the affected RCP is to be secured, THEN PERFORM the following: 1) REDUCE Reactor power to less than 48% (P-8 extinguished) using any of the following: • OTO-MA-00008, Rapid Load Reduction • OTG-ZZ-00004, Power Operation 2) TRIP the affected RCP per Attachment E, RCP Trip. 3) MONITOR RCP parameters for all RCPs. 4) CONTINUE plant shutdown.
A3.	**CHECK Both Of The Following:** • RCP Seal Parameters - NORMAL • CCW to RCPs - NORMAL	PERFORM the following: a. IF RCP seal parameters are NOT normal, THEN Go To Attachment B, RCP Seal Parameters Abnormal. b. IF CCW to the RCPs are NOT normal, THEN Go To Attachment C, CCW To RCP Abnormal.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 4 of 4)
RCP High Vibration

NOTE

In Modes 1 and 2 when a RCP is stopped, the idle loop RTD channel is inoperable.

A4.	**CHECK Reactor Power - GREATER THAN 48% (P-8 lit)**	IF the plant is in Mode 1 or 2 AND any RCP is secured, THEN PERFORM OTO-BB-00004, RTD Channel Failures.
A5.	**CHECK Any RCPs - RUNNING**	PERFORM OSP-BL-00001, Rx M/U Wtr Iso Vlvs W/O RCS Loops In Operation/Mode 6 Alignment.
A6.	**REVIEW Applicable Technical Specifications:** • Refer To Attachment G, Technical Specifications	
A7.	**PERFORM Notifications Per ODP-ZZ-00001 Addendum 13, Shift Manager Communications**	
A8.	**Go To Appropriate Plant Procedure As Directed By The Shift Manager/Control Room Supervisor**	

-END-

NRC Site-Specific Written Examination
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Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
Instrument Air	**Group #**	1		
	K/A #	078 A2.01		
	Importance Rating	2.9		
Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions				

Question # 90

Reactor Power is 70%.

- Instrument Air Dryer Train A is in service.
  - o During the regeneration cycle KAFV0345, DRYER FKA03B INLET FLO VLV, fails closed.
  - o Instrument Air Header pressure lowers to 103 psig and then stabilizes.

(1) What is the impact of this failure?

And

(2) What action will the CRS direct to address this failure?

A. (1) A power reduction due to a loss of Instrument Air is required.
(2) Reduce Reactor Power using Attachment A, Load Reduction, of OTO-KA-00001, Partial or Total Loss of Instrument Air.

B. (1) The Instrument Air Dryer Train B will automatically go into service with the flow path failed open.
(2) Select B Train and then ensure proper operation of Dryer Train ‘B’ per OTN-KA-00001, ADDENDUM 2, INSTRUMENT AIR DRYER OPERATION.

C. (1) A power reduction due to a loss of Instrument Air is required.
(2) Reduce Reactor Power using OTO-MA-00008, Rapid Power Reduction.

D. (1) The Instrument Air Dryer Train B will automatically go into service with the flow path failed open.
(2) Verify KAFV0344, ‘A’ Tower Inlet isolation valve on the ‘A' Instrument Air Dryer Train failed open per OTO-KA-00001 ADDENDUM 01, TURBINE BUILDING LOSS OF INSTRUMENT AIR.

Answer: B

Explanation: *When Instrument Air Pressure drops below 105 psig the Instrument Air Dryer that is NOT in service will be automatically go into service with the flow path failed open placed in service. While at 103 psig instrument air pressure is low, a power reduction is not needed because the initial power stated in the stem was less than 80% (action from OTO-KA-00001 Step #3).*

The proper action per OTA-KA-146 Addendum 4A, Low Pressure, is to Select B Train and then ensure proper operation of Dryer Train 'B' per OTN-KA-00001, ADDENDUM 2, INSTRUMENT AIR DRYER OPERATION

A. Incorrect per above. Plausible because instrument air pressure is reduced and if Reactor Power was greater than 80% and condition were not stable this would be the correct action to take.
B. Correct, see above
C. Incorrect per above. Plausible because OTO-MA-00008 is entered if the SM or CRS determine plant conditions require a timely load reduction
D. Incorrect per above. The impact is correct however the action is not. Plausible because this is an action that is taken at the instrument air dryer if a TOTAL loss of instrument air has occurred.

Technical Reference(s):
1. OTA-KA-00146 Addendum 4A, Low Pressure, Rev 1
2. OTN-KA-00001, ADDENDUM 2, INSTRUMENT AIR DRYER OPERATION, Rev 5
3. OTO-KA-00001, Partial or Total Loss of Instrument Air, Rev 23

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, B-20, OTO-KA-00001, PARTIAL OR TOTAL LOSS OF INSTRUMENT AIR Objective: D. Given a set of plant conditions or parameters indicating a Partial or Total Loss of Instrument Air, IDENTIFY the correct procedure(s) to be utilized and OUTLINE the high level actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

NRC Site-Specific Written Examination
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Comments:

SRO Only due to:
Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES**
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Initiating Device:	Setpoint:	Reset:
1. KAPSL0068A	1. ≤105 psig	1. ≥115 psig

LOW PRESSURE

1.0 AUTOMATIC ACTIONS:

1.1. IF KAHS0358, LOAD TRN SEL SW, is in DRYER A, the following valves OPEN, placing Dryer Train B in service:

- KAFV0356, PREFILTER FKA01B INLET FLO VLV
- KAFV0357, AFTERFILTER FKA02B DISCH FLO VLV

2.0 IMMEDIATE ACTIONS:

2.1. None

3.0 OPERATOR ACTIONS:

3.1. IF KAHS0358, LOAD TRN SEL SW, is in DRYER B, Go To Addendum 4B.

> **NOTE:**
> High d/p across <u>both</u> FKA01A, COMPRESS AIR SYS INST AIR DRYERS PREFILTER A, and FKA02A, COMPRESSED AIR SYSTEM INSTR AIR DRYERS AFTERFILTER A, indicates excessive Instrument Air usage downstream of the Air Dryer.

3.2. CHECK the following for high d/p:

- KAPDI0012A, PREFILTER FKA01A DIFF PRESS IND
- KAPDI0025A, AFTERFILTER FKA02A DIFF PRESS IND

3.3. IF excessive Instrument Air usage downstream of the Air Dryers is indicated, ISOLATE the Instrument Air supply to this excessive usage.

3.4. IF cause of low pressure is due to a malfunction or leak from Dryer Train A, PERFORM the following:

3.4.1. PLACE KAHS0358, LOAD TRN SEL SW, in DRYER B.

3.4.2. ENSURE the following are OPEN:

- KAFV0356, PREFILTER FKA01B IN FLO VLV
- KAFV0357, AFTERFILTER FKA02B DISCH FLO VLV

OPERATOR ACTIONS (Cont'd):

> **NOTE:**
> Valve position indication may be erroneous in the case of valve stem failure. Valve position should be verified by checking the stem indicator on the valve side that is opposite the valve operator.
>
> A failed open valve would be indicated by excessive flow noise through the tower muffler.

3.4.3. IF Dryer Train A is leaking:

a) CLOSE the following:

- KAV0755, PREFILTER FKA01A IN ISO
- KAV0759, PREFILTER FKA01A & B DISCH CROSSTIE
- KAV0766, AFTERFILTER FKA02A DISCH VLV
- KAV0768, DRYERS FKA03A & B AND FKA04A & B CROSSTIE

b) CHECK the following applicable valve of the Dryer Tower in service CLOSED:

- KAFV0348, DRYER FKA03A REGEN OUTLET VLV
- KAFV0349, DRYER FKA03B REGEN OUTLET VLV

3.4.4. ENSURE proper operation of Dryer Train B per OTN-KA-00001, Addendum 2, Instrument Air Dryer Operation, Attachment 1 - Sequence and Indications of the Drying and Regeneration Cycle.

3.4.5. IF Dryer Train A does NOT appear to be leaking, ENSURE the following Dryer Train A valves are OPEN:

- KAV0755, PREFILTER FKA01A IN ISO
- KAV0757, PREFILTER FKA01A DISCH VLV
- KAV0764, AFTERFILTER FKA02A IN ISO
- KAV0766, AFTERFILTER FKA02A DISCH VLV

> **CAUTION:**
> The following step will fail open the Dryer Train A inlet and outlet valves with Dryer Tower A in service. There will be no regeneration capability for either Dryer Tower. Operation in this mode permits about 4 hours of system operation before the Dryer desiccant will be saturated.

3.4.6. IF Dryer Train B is NOT available AND Dryer Train A does NOT have a leak, OBTAIN emergency Instrument Air Flow through Dryer Train A as follows:

a) PLACE KAHS0359A, DRYERS FKA03A & B PWR ON SW, in OFF.

OPERATOR ACTIONS (Cont'd):

b) ENSURE the following valves fail OPEN:

- KAFV0354, PREFILTER FKA01A INLET FLO VLV
- KAFV0344, DRYER FKA03A INLET FLO VLV
- KAFV0355, AFTERFILTER FKA02A DISCH FLO VLV

c) ENSURE KAFV0345, DRYER FKA03B INLET FLO VLV, fails CLOSED.

d) ENSURE KAPI0360, DRYER FKA03A PRESS IND, is approximately the same pressure as any of the following:

- KAPI0327A, AIR CMPSR A TKA01A PRESS IND
- KAPI0327B, AIR CMPSR B TKA01B PRESS IND
- KAPI0327C, AIR CMPSR C TKA01C PRESS IND

e) IF required to reduce moisture carryover, THROTTLE OPEN the following:

- KAV0775, PREFILTER FKA01A CNDS TRAP BYP
- KAV1114, AFTERFILTER FKA02A DRN VLV
- KAV0444, COMPRESS AIR SYS AFTERFILTERS OUT HDR DRAIN VLV

f) WHEN the system is restored, PERFORM the following:

1) CLOSE the following valves:
    - KAV0775, PREFILTER FKA01A CNDS TRAP BYP
    - KAV1114, AFTERFILTER FKA02A DRN VLV
    - KAV0444, COMPRESS AIR SYS AFTERFILTERS OUT HDR DRAIN VLV
2) PERFORM the section Returning Train A Air Dryers to Service, in OTN-KA-00001, Addendum 2, Instrument Air Dryer Operation.

3.5. IF excessive air usage OR malfunction of the Air Dryer Train are NOT indicated, CHECK for malfunction of the Air Compressors (i.e. poppet valve failure).

3.6. IF operation of Dryer Train B is no longer needed, using KAHS0370A, DRYER TRAIN A ALARM RESET SW, RESET alarm.

4.0 SUPPLEMENTAL INFORMATION:

4.1. Drawings:

- M-1046-00004, Duplex Air Dryers Wiring Schematic
- M-22KA06, P&ID Instrument Air Filter/Dryer
- M-1046-00002, Duplex Air Dryers and Filters Schematic

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
Non-Nuclear Instrumentation	**Group #**	2		
	K/A #	016 G2.1.23		
	Importance Rating	4.4		
Ability to perform specific system and integrated plant procedures during all modes of plant operation.				

Question # 91

Reactor Power is 100%.

- BB PT-458, Pressurizer Pressure Instrument - Channel 4, has failed.
- ALL bistables associated with Pressurizer pressure channel BB PT-458 have been TRIPPED per OTO-BB-00006 Attachment E, Tripping Pressurizer Pressure Protective Bistable.

What is the procedural flow path the CRS should direct WHEN BB TI-412, Loop 1 Reactor Coolant Tavg, fails HIGH?

A. E-0, Reactor Trip or Safety Injection, then to ES-1.1, SI Termination.

B. E-0, Reactor Trip or Safety Injection, then to ES-0.1, Reactor Trip Response.

C. OTO-BB-00004, RCS RTD Channel Failures, Attachment B, Tripping Loop 1 RTD Channel Bistable

D. OTO-MA-00008, Rapid Load Reduction, then to OTG-ZZ-00005, Plant Shutdown – 20% Power to Hot Standby and shutdown the unit within 7 hours.

Answer: B

Explanation:

The failed BB PT-458 channel, with the associated tripped bistables places the unit in a half tripped condition for LO PZR Press SI, LO PZR Press Rx trip, HIGH Pzr Press Rx Trip and OTDT Rx trip on 4 channel. When the Tavg instrument failed High on loop 1, a 2 out of 4 coincidence is made up on OTDT Rx trip. This results in a reactor trip. When the reactor trips, the crew will enter 1 E-0 and then transition to ES-0.1 to stabilize/recover the plant.

A. Incorrect. Transitioning to ES1.1 is plausible if the candidate believes an SI signal will be generated with the Reactor Trip. The candidate may believe that an SI signal is generated on low pressurizer pressure due to Tavg channel failing high which affect steam dumps post trip. Auctioneered Hi TAVG is used as the delta Temperature input into the steam dump control signal and with a reactor trip (arming signal for the steam dumps), the candidate may believe they

remain open causing a cooldown and PZR pressure to lower to the SI setpoint. This is a plausible scenario if the candidate misunderstands integrated plant response and the fact that the permissive, P-12, will disarm the steam dumps when Tavg reach 550F to prevent this from occurring.
B. Correct, See above
C. Incorrect. Entering OTO-BB-00004, RCS RTD Channel Failures, Attachment B, Tripping Loop 1 RTD Channel Bistable, is plausible if the examinee feels a reactor trip will not occur
D. Incorrect, Entering OTO-MA-00008 to perform a shutdown is plausible if the candidate recognizes we have entered TS 3.0.3 but fails to recognize the reactor trip

Technical Reference(s):
1. OTO-BB-00006, PZR Pressure Control Malfunction, Rev 20
2. OTO-BB-00004, RCS RTD Channel Failures, Rev 18

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, REACTOR PROTECTION – SB LP 27, Objective C: LIST all the Reactor Trip Signals supplied to RPS, including setpoint, coincidence, interlocks and protection afforded.

Question Source: Bank # ___X_(only procedure names were changed from Byron 2013 SRO exam)__
Modified Bank # ______
New _______

Question History: Last NRC Exam ______2013 Byron SRO ILT Written Exam Question #90 _ http://pbadupws.nrc.gov/docs/ML1314/ML13148A356.pdf_____

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

Revised question per NRC comments.

Question meets the KA by testing the examinee on their ability to perform system and integrated plant procedure to address failures with Non Nuclear Instrumentation. The question is High Cog and at the SRO level, as the SRO will direct the crew procedure transitions.

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

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Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE Steps 1 through 4 are immediate action steps.	
1.	**CHECK Reactor Trip:** • Rod Bottom Lights — ALL LIT • Reactor Trip and Bypass Breakers — OPEN • Neutron Flux — LOWERING	Manually TRIP Reactor. IF Reactor Power is greater than or equal to 5% OR Intermediate Range SUR is positive, THEN Go To FR-S.1, Response To Nuclear Power Generation/ATWS, Step 1.
2.	**CHECK Turbine Trip:** a. All Turbine Stop valves — CLOSED	a. Manually TRIP Turbine. IF Turbine will NOT trip, THEN FAST CLOSE all MSIVs and Bypass valves: • AB HS-79 • AB HS-80

FOLDOUT PAGE FOR E—0

1. **RCP TRIP CRITERIA**
   IF BOTH conditions listed below occur,
   THEN TRIP all RCPs:
   - CCPs or SI Pumps — AT LEAST ONE RUNNING

     AND
   - RCS pressure — LESS THAN 1425 PSIG

2. **FAULTED SG ISOLATION CRITERIA**
   IF any SG pressure is lowering in an uncontrolled manner OR is completely depressurized,
   THEN PERFORM the following as desired:
   - FAST CLOSE MSIVs.
   - Manually CLOSE or locally ISOLATE any failed open ASD(s).
   - ISOLATE feed flow to faulted SG(s).
   - MAINTAIN total feed flow greater than 285,000 lbm/Hr until narrow range level is greater than 7% [25%] in at least one SG.

3. **RUPTURED SG ISOLATION CRITERIA**
   IF BOTH conditions listed below occur,
   THEN ISOLATE feed flow to affected SG(s) as desired:
   - Level in any SG rises in an uncontrolled manner OR any SG has abnormal radiation.

     AND
   - Narrow range level in affected SG(s) — GREATER THAN 7% [25%].

4. **COLD LEG RECIRCULATION CRITERIA**
   IF RWST level lowers to less than 36%,
   THEN Go To ES—1.3, Transfer To Cold Leg Recirculation, Step 1.

5. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

6. **SPENT FUEL POOL COOLING**
   IF SFP Cooling pumps have tripped,
   THEN monitor SFP level and temperature and implement the following as resources permit:
   - OTO—EC—00001, Loss of SFP/Refuel Pool Level
   - OTO—EC—00002, Spent Fuel Pool High Temperature

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**CHECK Power To AC Emergency Buses:**	
	a. AC emergency buses — AT LEAST ONE ENERGIZED • NB01 OR • NB02	a. Perform the following: 1) Depress START/RESET pushbutton for any stopped Diesel Generator: • KJ HS-8A • KJ HS-108A 2) IF DG started AND output breaker did NOT close, THEN CLOSE DG output breaker: • NE HS-25 • NE HS-26 3) IF neither AC emergency bus is energized, THEN go to ECA-0.0, Loss Of All AC Power, Step 1.
	b. AC emergency buses — BOTH ENERGIZED	b. TRY to restore power to deenergized AC emergency bus as time permits: 1) Depress START/RESET pushbutton for any stopped Diesel Generator: • KJ HS-8A • KJ HS-108A 2) If DG started AND output breaker did NOT close, THEN close DG output breaker: • NE HS-25 • NE HS-26

FOLDOUT PAGE FOR E—0

1. **RCP TRIP CRITERIA**
   IF BOTH conditions listed below occur,
   THEN TRIP all RCPs:
   - CCPs or SI Pumps — AT LEAST ONE RUNNING
     AND
   - RCS pressure — LESS THAN 1425 PSIG

2. **FAULTED SG ISOLATION CRITERIA**
   IF any SG pressure is lowering in an uncontrolled manner OR is completely depressurized,
   THEN PERFORM the following as desired:
   - FAST CLOSE MSIVs.
   - Manually CLOSE or locally ISOLATE any failed open ASD(s).
   - ISOLATE feed flow to faulted SG(s).
   - MAINTAIN total feed flow greater than 285,000 lbm/Hr until narrow range level is greater than 7% [25%] in at least one SG.

3. **RUPTURED SG ISOLATION CRITERIA**
   IF BOTH conditions listed below occur,
   THEN ISOLATE feed flow to affected SG(s) as desired:
   - Level in any SG rises in an uncontrolled manner OR any SG has abnormal radiation.
     AND
   - Narrow range level in affected SG(s) — GREATER THAN 7% [25%].

4. **COLD LEG RECIRCULATION CRITERIA**
   IF RWST level lowers to less than 36%,
   THEN Go To ES—1.3, Transfer To Cold Leg Recirculation, Step 1.

5. **AFW SUPPLY SWITCHOVER CRITERIA**
   IF CST to AFP suction header pressure lowers to less than 2.75 PSIG,
   THEN PERFORM EOP Addendum 19, Aligning ESW To AFW Suction.

6. **SPENT FUEL POOL COOLING**
   IF SFP Cooling pumps have tripped,
   THEN monitor SFP level and temperature and implement the following as resources permit:
   - OTO—EC—00001, Loss of SFP/Refuel Pool Level
   - OTO—EC—00002, Spent Fuel Pool High Temperature

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**CHECK SI Status:**	
	a. CHECK if SI is actuated: • Any SI annunciator 88A through 88D — LIT OR • SB069 SI Actuate RED light — LIT OR • LOCA Sequencer annunciators 30A or 31A — LIT	a. CHECK if SI is required: • PZR pressure less than or equal to 1849 PSIG OR • Any SG pressure less than or equal to 615 PSIG OR • Containment pressure greater than or equal to 3.5 PSIG IF SI is required, THEN manually ACTUATE SI: • SB HS-27 • SB HS-28 IF SI is NOT required, THEN Go To ES-0.1, Reactor Trip Response, Step 1.
	b. CHECK both Trains of SI — ACTUATED • LOCA Sequencer annunciator 30A — LIT • LOCA Sequencer annunciator 31A — LIT • SB069 SI Actuate RED light — LIT SOLID (NOT blinking)	b. Manually ACTUATE SI: • SB HS-27 • SB HS-28
5.	**PERFORM Attachment A, Automatic Action Verification, While Continuing With This Procedure**	

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Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	2		
In-Core Temperature Monitor	**Group #**	2		
	K/A #	017 G2.4.30		
	Importance Rating	4.1		
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.				

Question # 92

Reactor Power is 100%.

What event would REQUIRE notification of the NRC Resident Inspector in accordance with ODP-ZZ-00001 ADDENDUM 13, SHIFT MANAGER COMMUNICATIONS?

A. Loss of potable water to the EOF.

B. ‘A’ SG Chlorides are at Action Level 1.

C. No In-Core thermocouples are operable in Quadrant 1 for 7 days.

D. Scheduled maintenance on emergency diesel generator, NE01, exceeds 50% of the allowed out of service time.

Answer: C

Explanation: *Per Attachment 1 of ODP-ZZ-00001 Add 13 only answer C requires notification of the NRC RI.*

A. Incorrect. Plausible per Note 6 of attachment 1 of ODP-ZZ-00001 Add 13 “Any significant event that reduces the services provided to an Emergency response Facility (EOF or TSC). This includes electrical power, ventilation, and potable water.” the NRC RI only needs to be notified on a loss of power to the EOF or TSC
B. Incorrect, Plausible due to the NRC RI must be notified for entry into a Chemistry Action Level 2 or 3 per APA-ZZ-01020 or 01021.
C. Correct. This condition is an unplanned entry into a Tech Spec 3.3.3 (PAM Instrumentation) action statement C and then D since no thermocouple in restored within 7 days. Condition D applies with per the PAM T.S. table requires entry into Condition E. Per Note 1 of ODP-ZZ-00001 ADD 13 "Note 1: Entry into Tech. Spec. action statement with < 24 hours completion time, contact immediately. If >24 hours but < 72 hours completion time, contact as soon as possible between 0800 – 2200, regardless of day of the week." Therefore the completion time had to be less than 72 hours for this to apply and hence why condition D and E were utilized.

D. Incorrect, Plausible per attachment 1 of ODP-ZZ-00001 Add 13 this is a condition that the Duty Manager, Director, Nuclear Operations, and the Senior Director, Nuclear Operations must be notified.

Technical Reference(s):
1. TS and TS Bases 3.3.3 PAM Instrumentation
2. ODP-ZZ-00001 ADDENDUM 13, SHIFT MANAGER COMMUNICATIONS. Rev 20

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP-77 INTRODUCTION TO TECHNICAL SPECIFICATIONS G. EXPLAIN and APPLY the LCO/SR applicability section of Technical Specifications.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ______N/A______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(1)

Comments:

Revised answer and distractor wording per NRC Comments. Reordered choices based on length.

SRO per criteria 1 due to the reporting requirements associated with the facility license.

Attachment 1

Matrix

Sheet 1 of 6

	DM	NRC RI	SDNO/ DNO	RM & VPN or SVPC	RP MNGR	SUPT CHEM	SSS	DUTY TEAM
Entry into an Action Statement requiring "immediate" corrective action.	X	X	X					X
Unplanned entry into a Tech. Spec. action statement.	X	X Note 1	X					X
High potential exists to have to shut down the unit due to Tech. Spec. action statement, equipment problem (non-Tech. Spec.), Chemistry limits, etc.	X Note 2	X	X	X			X Only required for a complete loss of the CST OR all Fire Pumps.	X
Tornado WARNING issued for Callaway, Gasconade, Osage, or Montgomery County.	X	X	X					
Equipment inoperability that results in the unplanned entry of into a Red Condition per EDP-ZZ-01129.	X	X	X				X	X
Equipment issue that requires an Immediate Operability Determination and support is needed to investigate and define the degraded condition								X

Attachment 1 (Cont'd.)

Sheet 2 of 6

	DM	NRC RI	SDNO/ DNO	RM & VPN or SVPC	RP MNGR	SUPT CHEM	SSS	DUTY TEAM
A plant trip or an Unplanned load reduction of > 2% reactor power.	X Note 2 DM Contact Reg Affairs (If SD >3 days is expected)	X Note 9	X	X Note 3 RM Contact Corporate Communications	X >15% Rx Power	X >15% Rx Power		X
Unplanned entry into an OTO procedure.	X							
Unscheduled maintenance activity that requires calling out an engineer, planner, or QC personnel to support.	X							
Event that results in a potentially reportable NPDES concern.	X	X				X		
Release of a contaminate to the Missouri River, such as oil.	X	X Note 5	X	X Note 3 RM Contact Corporate Communications		X		
Event involving a significant potential threat to personnel or nuclear safety.	X	X	X	X			X	

Attachment 1 (Cont'd.)

Sheet 3 of 6

	DM	NRC RI	SDNO/ DNO	RM & VPN or SVPC	RP MNGR	SUPT CHEM	SSS	DUTY TEAM
Events involving an unplanned radioactive release or unplanned personnel exposure.	X	X	X	X Note 3 RM Contact Corporate Communications	X			
Significant spread of contamination beyond expected boundaries requiring control, clean-up measures, or precautionary evacuation of personnel.	X	X	X	X Note 3 RM Contact Corporate Communications	X			
Condition when the RERP EALs will or potentially could require a declaration.	X	X	X	X Note 3 RM Contact Corporate Communications RM Contact Emergency Preparedness				
Scheduled maintenance in an LCO that exceeds 50% of the allowed out of service time.	X Note 2		X					
Callout of the MERT that requires transfer of personnel for off-site medical assistance.	X DM Contact Safety Supervisor	X Note 8	X	X Note 3 RM Contact Corporate Communications				
Ambulance called to site	X DM Contact Safety Supervisor	X Note 8	X					

Attachment 1 (Cont'd.)

Sheet 4 of 6

	DM	NRC RI	SDNO/ DNO	RM & VPN or SVPC	RP MNGR	SUPT CHEM	SSS	DUTY TEAM
Injury which may potentially be recordable or lost time away.	X							
Contaminated, injured person.	X DM Contact Safety Supervisor	X	X	X Note 3 RM Contact Corporate Communications	X			
A significant medical emergency that involves a fatality, serious injury (loss of finger, limb, etc.), life threatening Injury (burn over large portion of body), or one that draws media attention.	X DM Contact Safety Supervisor	X	X	X Note 3 RM Contact Corporate Communications RM Perform Policy 41 and Notify Emerg. Contact.				
Activation of the Fire Brigade.	X	X	X	X		X Only if chemical spill		
Entry into a Chemistry Action Level per APA-ZZ-01020 or 01021.	X	X Action Levels 2 & 3 only	X		X Note 7	X		
Events that are determined to be reportable or potentially reportable in accordance with APA-ZZ-00520.	X RM Contact Reg Affairs	X	X					

 INFORMATION USE

Attachment 1 (Cont'd.)

Sheet 5 of 6

	DM	NRC RI	SDNO/ DNO	RM & VPN or SVPC	RP MNGR	SUPT CHEM	SSS	DUTY TEAM
Unannounced inspection by any state or federal agency during off-normal hours.	X DM Contact Director NOS	X	X	X				
Unplanned engineering safety features or component actuation.	X	X	X					X
Event Review Teams required by procedure.	X	X	X					
Loss of Control Room annunciators (ten or more).	X	X	X					X
Total loss of decay heat removal (for any reason and for any duration, even those that are allowed by T/Ss).	X	X	X	X				
Evidence of steam generator tube leaks or fuel failures while at power.	X	X	X		X	X		
Evidence of visible fuel damage while refueling.	X	X	X					
Control rod mispositioning events.	X	X	X					
Any event where media interest is shown or anticipated, such as pickets, protesters, fire, police etc.	X	X	X	X Note 3 RM Contact Corporate Communications RM Contact Emergency Preparedness				

INFORMATION USE

Attachment 1 (Cont'd.)

Sheet 6 of 6

	DM	NRC RI	SDNO/ DNO	RM & VPN or SVPC	RP MNGR	SUPT CHEM	SSS	DUTY TEAM
Any significant event that reduces the services provided to an Emergency response Facility (EOF or TSC). This includes electrical power, ventilation, and potable water.	X DM Contact Emerg. Prep.	X Note 6						
Any Transient Event as defined in EDP-ZZ-01007.	X	X	X					X
EAP Crisis Counseling assistance request for personal tragedy situations. (800-888-2273)	X		X					
Any Security Event of a suspicious nature, such as, unexpected flyovers by non-commercial aircraft, suspicious vehicles near the plant, or suspicious activity by individuals near the plant.	X	X Note 4	X					

Note 1: Entry into Tech. Spec. action statement with $\leq$ 24 hours completion time, contact immediately. If >24 hours but $\leq$ 72 hours completion time, contact as soon as possible between 0800 – 2200, regardless of day of the week.
Note 2: The SM should contact the PSS (Power Supply Supervisor) via the "Dedicated Line" from the Control Room (Backup: x43988) for any imminent load reduction.
Note 3: Corporate Communications should be contacted and informed of the circumstances surrounding the occurrence in the event that they are contacted for additional information by the media. If a news release is issued, review applicability of APA-ZZ-00520, Reporting Requirements And Responsibilities. Emergency Preparedness should be contacted and directed to contact the counties, SEMA and the NEI Duty Officer and brief them on the occurrence. (Corporate Media Hotline 314-554-2182)
Note 4: The SM should notify the NRC Resident and also notify the NRC Operations Center using NRC Form 361 as an information only report.
Note 5: The SM should notify the NRC Resident if ANY report is required by APA-ZZ-00520, Reporting Requirements And Responsibilities
Note 6: The SM should notify the NRC Resident only on a loss of electric power to the EOF or TSC.
Note 7: The SM should notify the RPM if polisher operation support is anticipated.
Note 8: The SM should also notify the NRC Resident if an ambulance is requested, even if not transported.
Note 9: If downpower of >2% occurs immediately due to a failed component.

 INFORMATION USE

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
Spent Fuel Pool Cooling	Group #	2		
	K/A #	033 A2.01		
	Importance Rating	3.5		
Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadequate SDM				

Question # 93

Fuel is being moved in the Spent Fuel Pool.

Spent fuel pool boron concentration is 1950 ppm.

(1) Spent fuel pool Shutdown Margin (k_{eff}) is limited to a MAXIMUM of _____(1)_____? (Assume only one fuel assembly is mispositioned)

And

(2) Action must be initiated immediately to restore the MINIMUM boron concentration to ….?

A. (1) 0.95
(2) 2000 ppm

B. (1) 0.99
(2) 2165 ppm

C. (1) 0.95
(2) 2165 ppm

D. (1) 0.99
(2) 2000 ppm

Answer: C

Explanation: *Per FSAR Appendix 9.1A.2.1.2 (Abnormal and Accident condition “The inadvertent misplacement of a fresh fuel assembly has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. Assuring the presence of soluble poison will preclude the simultaneous occurrence of the two independent accident conditions during fuel handling operations. The largest reactivity increase would occur if a fresh fuel assembly of 5.0 wt% 235U enrichment were to be inadvertently loaded into an empty cell in the checkerboard configuration* *with the remainder of the rack fully loaded with fuel of the highest permissible reactivity. For the MZTR configuration, when a fresh fuel assembly of 5.0 wt% 235U enrichment is inadvertently loaded into a Region 2 location (with the remainder of the rack fully loaded with fuel of the highest permissible reactivity), the overall reactivity is slightly less reactive. However, it still exceeds the limiting value without the presence of soluble boron. Under these accident conditions, credit for the presence of soluble poison is permitted by the NRC guidelines. Calculations indicate that 500 ppm soluble boron would be adequate to reduce the keff to below 0.95"*

The action directed by TS 3.7.16 Action A is to immediately initiate action to restore fuel storage pool boron concentration within the limit. The limit is ≥2165 ppm boron in the spent fuel pool.

The basis of T.S. 3.7.16 states "The water in the fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting keff of 0.95 be evaluated in the absence of soluble boron."

A. Incorrect, The k_{eff} value is correct per the reference stated above, The boron concentration is incorrect. It is plausible if the applicant misapplies TS 3.9.1 (boron concentration during refueling). The limit for TS 3.9.1 is 2000 ppm during refueling in MODE 6.
B. Incorrect, The k_{eff} valve is incorrect per the reference. It is a plausible if the operator misapplies the k_{eff} values associated with MODE changes. The boron concentration is correct,
C. Correct, see above
D. Incorrect, both part are incorrect per the references, plausible as described above

Technical Reference(s):
1. *FSAR Appendix 9.1A.2.1.2 and*
2. *Technical Specification and its Bases3.7.16 – Fuel Storage Pool Concentration*

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#24, FUEL POOL COOLING AND CLEANUP SYSTEM – EC, Objective H: STATE the LCOs associated with the Fuel Pool Cooling and Cleanup System (FPCCS) Technical Specifications (T/S) and Final Safety Analysis Report (FSAR).
2. T/S 3.7.16

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ______NA______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(2)

Comments:

The question meets the KA due to placing the candidate in a condition with inadequate SDM per TS since fuel is in the process of being moved and a verification of fuel assembly location in the spent fuel pool following these moves has not yet been conducted.

The question meets the SRO Only criteria because it is testing the applicant's knowledge of TS bases and FSAR accident analysis of reactivity controls. Specifically:

SRO ONLY due to ES401 Figure 1 of NUREG 1021 as follows:

Can question be answered solely by knowing ≤ 1 hour TS/TRM Action? NO

Can question be answered solely by knowing the LCO/TRM information listed "above-the-line?" NO

Can question be answered solely by knowing the TS Safety Limits? NO

Does the question involve one or more of the following for TS, TRM, or ODCM? **YES**

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology. **YES**

The bounding criticality analyses are summarized in Table 9.1A-1 for the design basis MZTR storage configuration and in Table 9.1A-2 for the interim checkerboard storage configuration. In both cases, the single accident condition of the loss of all soluble boron is assumed. The calculated maximum reactivity of 0.943 (corresponding to the design basis MZTR storage configuration) is within the regulatory limit of 0.95. This maximum reactivity includes calculational uncertainties and uncertainties in reactivity due to manufacturing tolerances (95% probability at the 95% confidence level), an allowance for uncertainty in depletion calculations, and the evaluated effect of the axial distribution in burnup.

The value of k_{eff} in Table 9.1A-1 assumes no soluble boron to be present. For normal operations, a minimum soluble boron concentration of 2165 ppm is maintained in the Callaway fuel storage pool. This concentration of soluble boron provides a large safety margin for sub-criticality.

As cooling time increases in long-term storage, decay of 241Pu (and growth of 241Am) results in a continuous decrease in reactivity, which provides an increasing sub-criticality margin with time. No credit is taken for this decrease in reactivity other than to indicate conservatism in the calculations.

The burnup criteria identified in Figure 9.1A-3, for acceptable storage in Region 2 and Region 3, are used in appropriate administrative procedures to assure verified burnup as specified in the proposed Regulatory Guide 1.13, Revision 2 (Reference 4). Soluble poison is present in the pool water during fuel handling operations, and this serves as a further margin of safety and as a precaution in the event of fuel misplacement during fuel handling operations.

9.1A.2.1.2 Abnormal and Accident Conditions

Although credit for the soluble poison normally present in the fuel storage pool water is permitted under abnormal or accident conditions, most abnormal or accident conditions will not result in exceeding the limiting reactivity (k_{eff} of 0.95) even in the absence of soluble poison. The effects on reactivity of credible abnormal and accident conditions are discussed in Section 9.1A.2.2.5 and summarized in Table 9.1A-3. Of these abnormal or accident conditions, only two have the potential for a more than negligible positive reactivity effect. These include: (1) the inadvertent misplacement of a fresh fuel assembly and (2) the mis-location of a fresh fuel assembly into a position external and adjacent to a storage rack.

The inadvertent misplacement of a fresh fuel assembly has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. Assuring the presence of soluble poison will preclude the simultaneous occurrence of the two independent accident conditions during fuel handling operations. The largest reactivity increase would occur if a fresh fuel assembly of 5.0 wt% ^{235}U enrichment were to be inadvertently loaded into an empty cell in the checkerboard configuration with the remainder of the rack fully loaded with fuel of the highest permissible reactivity. For the MZTR configuration, when a fresh fuel assembly of 5.0 wt% ^{235}U enrichment is inadvertently loaded into a Region 2 location (with the remainder of the rack fully loaded with fuel of the highest permissible reactivity), the overall reactivity is slightly less reactive. However, it still exceeds the limiting value without the presence of soluble boron. Under these accident conditions, credit for the presence of soluble poison is permitted by the NRC guidelines. Calculations indicate that 500 ppm soluble boron would be adequate to reduce the k_{eff} to below the reference k_{eff} value (Table 9.1A-1). This soluble boron concentration bounds all other accidents and is well below the 2165 ppm soluble boron concentration that is maintained in both the Callaway fuel storoage pool.

It is possible for a fuel assembly to be dropped or mis-located in the fuel storage pool such that it may be situated outside and adjacent to a storage rack. The calculated k_{eff} value for the worst case situation exceeds the limit on reactivity in the absence of soluble boron. Because this case is less severe than the misplaced fresh fuel assembly accident, it requires less than 500 ppm soluble boron to reduce the k_{eff} to the reference value (Table 9.1A-1).

9.1A.2.2 Analytical Methodology

To assure the acceptability of the racks for storage of all fuel assembly design types, the most reactive assembly type was identified by independent criticality calculations. This most reactive assembly is the reference assembly used in the criticality calculations. In addition a nominal fuel storage cell is also used in the criticality calculations. This nominal fuel storage cell represents the fuel pool storage cells.

9.1A.2.2.1 Reference Fuel Assembly

The fuel storage pool racks are designed to accommodate any and all of the following Westinghouse fuel assembly types: 17x17 OFA, 17x17 Standard, and 17x17 Vantage 5H (V5H), with a maximum nominal initial enrichment of 5.0 wt% ^{235}U. Additional restrictions are specified to allow the storage of any of the aforementioned fuel assembly types without IFBA rods. Independent criticality calculations were performed to identify the most reactive assembly type. The results of these calculations show that at zero burnup the 17x17 OFA assembly has the greatest reactivity in the storage racks, and thus, is the design basis fuel assembly. The Westinghouse OFA is a 17 x 17 array of fuel rods with 25 rods replaced by 24 control rod guide tubes and 1 instrument thimble. Table 9.1A-4 summarizes the fuel assembly design specifications.

At burnups beyond approximately 25 MWd/kgU, the 17x17 Standard and 17x17 Vantage 5H become the most reactive assembly types. These two assembly types are essentially identical. Therefore, for the determination of the equivalent enrichments associated with Regions 2 and 3, the reactivity of the V5H assembly was related to an initial enrichment for the 17x17 OFA assembly.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Conduct of Operations	Group #	Generic		
	K/A #	2.1.5		
	Importance Rating	3.9		
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.				

Question # 94

Reactor Power is 100%.

- The shift is manned to the minimum required composition.
- The shift has 4 hours remaining.
- The Reactor Operator (RO) has become ill and must leave the site.

Which of the following describes the requirements regarding the shift composition and the MINIMUM required action in this situation?

A. The RO may not leave site until another qualified RO arrives and turnover of responsibilities is complete.

B. The RO may leave the site immediately after turnover of responsibilities to another qualified person on shift. A replacement MUST arrive within 1 hour.

C. The RO may leave the site immediately after turnover of responsibilities to another qualified person on shift. A replacement MUST arrive within 2 hours.

D. The RO may leave the site immediately after turnover of responsibilities to the BOP. No replacement is required for the remainder of the shift.

Answer: C

Explanation:

ODP-ZZ-00001 Step #4.4.1,c states "The shift complement may be one (1) less than the minimum requirements of Table 1 for a period of time NOT to exceed two (2) hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift complement to within the minimum requirements of Table 1. This provision does NOT permit any shift position to go unmanned upon-shift change due to an oncoming shift member being late or absent. [Ref: 5.2.54]"

A. Incorrect – Plausible as the shift is already at minimum manning or this may be confused with the SM,IA or SSO that must remain within the Owner Controlled Area (OCA).

B. Incorrect – the time aspect is wrong – 1 hour is not the requirement
C. Correct – see above explanation
D. Incorrect – the time aspect is wrong – 4 hours (i.e the remained of the shift) is not the requirement

Technical Reference(s):

1. ODP-ZZ-00001, Conduct of Operations, Rev 95 Section 4.4.1.c

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #66, Operations Code of Conduct, Objective A.7:
7. DISCUSS:
a. Minimum Shift Manning requirements
b. Unexpected absence requirements regarding shift complement
c. Whose permission is necessary for Control Room entry
d. Whose permission is necessary for 'At the Controls Area' entry
e. Personnel NOT requiring permission for 'At the Controls Area' entry

Question Source: Bank # __X__L16465__
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2007_________

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

10 CFR Part 55 Content:

10 CFR55.43(b)(1)

Comments:

Revised Distractor A per NRC Comments

k/a match as the candidate must have knowledge of the work hour limitations as listed in ODP-ZZ-00001 section 4.4.1 and then apply this knowledge / procedure requirements.

SRO ONLY per 10 CFR55.43(b)(1): Conditions and limitations in the facility license. Specifically, The required actions for not meeting administrative controls listed in Technical Specification (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements) as unit staffing is in Technical Specification 5.2.2 and this specific instance is listed in 5.2.2.b

4.4. Shift Operations

4.4.1. Shift Staffing

a. The minimum shift complement shall be as listed in Table 1: [Ref: 5.2.6, 5.2.36, 5.2.32, 5.2.35, 5.2.36, 5.2.41, 5.2.48, 5.2.53, 5.2.55]

Table 1: Minimum Shift Complement

Personnel On-shift	MODES						Fuel
	1	2	3	4	5	6 (2)	On-site (1)
Shift Manager	1	1	1	1	1	1	1
Control Room Supervisor (3)	1	1	1	1	1	1	1
Field Supervisor (3) (6)	1	1	1	1	1	1	1
Unit Reactor Operator	2	2	2	2	2	2	2
Operations Technician / Assistant Operations Technician (4) (5)	5	5	5	5	5	5	5
Shift Technical Advisor (3)	1	1	1	1	1	1	1
Additional Operations Personnel (7)	2	2	2	2	2	2	2
Chemistry Technician	1	1	1	1	1	1	1

NOTES

1. Fuel on site requires the minimum staffing of RERP Figure 5-1. [Ref: 5.2.48]

2. Core alterations shall be observed and directly supervised by a licensed SRO who has no other concurrent duties. [Ref: 5.2.27]

3. The CRS, FS, or a Shift Engineer shall meet the qualifications for the STA as required by the USNRC in NUREG 0737.

   The STA function may be fulfilled by the CRS or FS if they are STA-qualified. If the CRS and FS are not STA-qualified, a separate STA-qualified individual is required.

4. Minimum of five (5) OT or AOT are required to man the shift of which at least two (2) shall have completed Primary and Secondary watchstation classroom training.

5. Minimum of five Fire Brigade members forming a team consisting of a designated team leader, assistant team leader and team members. [Ref: 5.1.16]

6. The FS will normally be an SRO licensed individual or a non-licensed operating supervisor that is STA-qualified. If neither the CRS or FS is STA-Qualified, an additional STA-qualified individual is required to fill the IA function. [Ref: 5.2.19]

Step 4.4.1.a Cont'd

7. The additional personnel are required to fill the functions of RERP Control Room Communicator and the SSO. These are normally filled by an OT and an RO. [Ref: 5.2.19]

b. The following areas of confinement are established for shift personnel.

1. The on-shift CRS and 2 URO responsible for Control Room duties must remain within the Protected Area (PA).

2. The SM, IA or SSO may leave the PA for limited periods of time to support vital business needs but must remain within the Owner Controlled Area (OCA) fence. Time outside the PA should be minimized. Only one of these three is allowed outside the PA at any time. [Ref: 5.2.19]

3. The individuals designated as Fire Brigade Leader and Fire Brigade members must remain within the OCA fence and contractor parking lot. [Ref: 5.2.17 and 5.2.19]

c. The shift complement may be one (1) less than the minimum requirements of Table 1 for a period of time NOT to exceed two (2) hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift complement to within the minimum requirements of Table 1. This provision does NOT permit any shift position to go unmanned upon-shift change due to an oncoming shift member being late or absent. [Ref: 5.2.54]

d. The Control Room shall be occupied by at least one licensed URO and one licensed SRO. [Ref: 5.2.46, 5.2.47]

e. One URO is normally in the "IMMEDIATELY ADJACENT TO THE CONTROL PANELS AREA". (Attachment 1)

f. The SM normal watchstation shall be the Control Room. [Ref: 5.2.51]

1. During any absence of the SM from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid senior reactor operator license shall be designated to assume the Control Room command function. [Ref: 5.2.51]

2. During any absence of the SM from the Control Room while the unit is in MODE 5 and 6, an individual with a valid senior reactor operator or reactor operator license shall be designated to assume the Control Room command function. [Ref: 5.2.51]

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Equipment Control	Group #	Generic		
	K/A #	G2.2.7		
	Importance Rating	3.6		
Knowledge of the process for conducting special or infrequent tests.				

Question # 95

What is the MINIMUM position that is REQUIRED to conduct an Infrequently Performed Test (IPTE) Job Briefing?

A. Duty Manager

B. Shift Manager

C. IPTE Test Coordinator

D. Control Room Supervisor

Answer: A

Explanation:

Per APA-ZZ-0100A step 4.3.3 "Director, Nuclear Operations / Managers, Operations / Shift Outage Manager / Duty Manager or others designated in writing by the Director, Nuclear Operations, USING CA2680, IPTE Job Briefing / Critique and Attendance Sheet, CONDUCT the IPTE Job Briefing." Therefore the Duty Manager is required to conduct the IPTE Job Briefing

A. Correct – see above explanation.
B. Incorrect. Plausible as this individual has several responsibilities for IPTEs (see step 3.5 of APA-ZZ-0100A) including the safe operation of the plant and giving permission for the IPTE to begin but is not the conductor of the IPTE job brief.
C. Incorrect. APA-ZZ-0100A, Step 4.3.5. This individual is required to Provide any additional details during the brief but is not the "conductor" making this a plausible but incorrect choice.
D. Incorrect – The Control Room Supervisor is plausible as they hold a SRO license and responsible for the safe operation of the plant. Their responsibilities are listed in step 3.5 also.

Technical Reference(s):

1. APA-ZZ-0100A, Infrequently Performed Test Or Evolution Guidance, Rev 17

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, LP#14, Objective D: DISCUSS APA-ZZ-

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

0100a, Infrequently Performed Test Or Evolution to include the following:

1. Purpose and Scope
2. Shift Manager/Control Room Supervisor responsibilities
3. Operations Personnel responsibilities
4. DEFINE Infrequently Performed Tests or Evolutions (IPTE)

Question Source: Bank # _ _____
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(3)

Comments:

SRO ONLY level question based on applicant having to have the knowledge of administrative requirements concerning the conduct of Infrequently Performed Test or Evolution (IPTE) per 10 CFR 55.43(b)(3) – Facility licensee procedures required to obtain authority for design and operating changes in the facility

3.2.7. Resolves problems encountered during the performance of infrequently performed tests or evolutions.

3.2.8. Stops infrequently performed test or evolution when unexpected conditions arise.

3.2.9. Ensures that an adequate Margin of Safety has been determined and is maintained during performance of infrequently performed test or evolution.

3.3. Director, Engineering Services:

Designates a Test Lead or Coordinator for Engineering and I&C IPTE Procedures.

3.4. Director, Nuclear Operations:

Designates a Test Lead or Coordinator for Operations IPTE Procedures.

3.5. Shift Manager / Control Room Supervisor:

3.5.1. Is responsible for identifying procedures, evolutions or activities that may need to be evaluated as IPTE.

3.5.2. Is responsible for the safe operation of the plant under all conditions and at all times.

3.5.3. Is responsible for permitting, denying or discontinuing the performance of any infrequently performed test or evolution.

3.5.4. Ensures the test or evolution is terminated if any of the following occur:

- Plant parameter response is NOT as expected.
- Equipment does NOT respond as expected.
- Prerequisite conditions are NOT as expected.
- The Margin of Safety is significantly reduced.
- Test termination criteria is reached or exceeded.

3.6. Operations Personnel:

3.6.1. Are responsible to strictly adhere to, and fully understand the procedure steps of the test or evolution and to notify the SM/CRS/Test Coordinator when a step can NOT be performed as written.

3.6.2. Perform specifically assigned tasks as defined in the infrequently performed test or evolution procedure.

3.6.3. Immediately report any unexpected responses or conditions to the SM/CRS/Test Coordinator.

3.6.4. Maintain a questioning attitude during the performance of the test or evolution.

3.6.5. If acting as a member of the procedure team, participate in the IPTE Job Briefing and understand the following:

- Chain of command
- Applicable test termination criteria
- Applicable individual duties

3.7. Test Lead or Coordinator:

3.7.1. Properly coordinate the test in accordance with station procedures.

3.7.2. Assist with the briefing to ensure all support personnel duties, authorities and responsibilities are clearly stated.

3.8. Test Performer:

3.8.1. Is responsible to strictly adhere to, and fully understand the procedure steps of the test or evolution and to notify the SM/CRS/Test Coordinator when a step can NOT be performed as written.

3.8.2. Perform specifically assigned tasks as defined in the infrequently performed test or evolution procedure.

3.8.3. Immediately report any unexpected responses or conditions to the SM/CRS/Test Coordinator.

3.8.4. Participate in the IPTE Job Briefing and understand the following:

- Chain of command
- Applicable test termination criteria
- Applicable individual duties

3.9. Operations Training Manager and Technical Training Manager:

Responsible for assessing the training needs for infrequently performed tests or evolutions and providing training and/or Just In Time Training (JITT) for the activity, as required.

-END OF SECTION-

4.0 PROCEDURE INSTRUCTIONS

4.1. Evaluation Of Infrequently Performed Tests Or Evolutions

NOTE

Anyone involved in the evolution (Procedure Writer, Planner, Scheduler, Coordinator, Shift Manager, etc.) can use this procedure to evaluate evolution against IPTE criteria, however the management positions designated in Step 3.2 will make the final determination for IPTE applicability.

APA-ZZ-0100A Appendix A, Previously Evaluated IPTE Procedures, contains a list of previously evaluated IPTE procedures.

4.1.1. IF procedure is already identified as IPTE, AND all the requirements of Section 4.2 have NOT been incorporated into the procedure, Go To Section 4.2.

4.1.2. IF the procedure is already identified as IPTE, AND all the requirements of Section 4.2 have been incorporated into the procedure, PERFORM the following:

a. REVIEW procedure requirements against current plant conditions OR planned plant conditions.

b. REVISE as necessary using Section 4.2.

c. PERFORM or SCHEDULE procedure.

NOTE

Administrative Procedures (APA, *DP), Emergency Operating Procedures (EOP), Off Normal Procedures (OTO), and Annunciator Response Procedures (OTA) are NOT considered as meeting IPTE Guidelines due to the nature of the content and usage. [* = Department Designator]

4.1.3. IF the activity has NOT previously been evaluated as IPTE, DETERMINE IPTE applicability as follows:

a. IDENTIFY what actions or evolutions could become High Consequence Activities and why or how the High Consequences could occur.

Step 4.1.3 Cont'd

b. IF evolutions involve High Consequence Activities**,** EVALUATE the need for additional management oversight using criteria including, but NOT limited to the following:

- Evolutions NOT specifically covered by existing normal or Off Normal procedures
- Evolutions which are seldom performed even though covered by existing procedures (e.g., plant startup after a prolonged outage or after any outage that involves significant changes to systems, equipment or procedures related to the core, reactivity control, or reactor protection)
- Special, infrequently performed surveillance testing that involves complicated sequencing or placing the plant in unusual configurations (e.g., ECCS check valve leakage tests)
- Evolutions that require use of special test procedures in conjunction with existing procedures

4.1.4. IF the activity is determined to be an IPTE, Go To Section 4.2.

-END OF SECTION-

4.2. Infrequently Performed Test Or Evolution Procedure Development

NOTE

The following actions are to be performed by the Procedure Writer to incorporate the criteria into the IPTE procedure.

4.2.1. PREPARE procedure using the additional requirements listed on Attachment 1, Infrequently Performed Tests or Evolutions (IPTE) Procedure Development Checklist. [Ref: 5.2.1]

4.2.2. WRITE explanations for any requirements in Attachment 1 that are NOT applicable to the procedure being developed and append them to Attachment 1.

4.2.3. INCLUDE the completed Attachment 1 and any explanations for non applicable items on Attachment 1 with the CA0033 form.

-END OF SECTION-

4.3. Preparations and Performance Of Infrequently Performed Tests Or Evolutions

4.3.1. Director, Nuclear Operations / Managers, Operations – EVALUATE use of JITT in accordance with ODP-ZZ-00001, Operations Department - Code Of Conduct, and CTM-OPS, Callaway Training Manual - Operations Programs.

4.3.2. If JITT is performed, the personnel responsible for oversight (Shift Manager and the Manager, Nuclear Operations / Managers, Operations / Shift Outage Manager / Duty Manager) should observe the JITT if available.

4.3.3. Director, Nuclear Operations / Managers, Operations / Shift Outage Manager / Duty Manager or others designated in writing by the Director, Nuclear Operations, USING CA2680, IPTE Job Briefing / Critique and Attendance Sheet, CONDUCT the IPTE Job Briefing.

4.3.4. Director, Nuclear Operations / Managers, Operations / Shift Outage Manager / Duty Manager or others designated in writing by the Director, Nuclear Operations USING CA2423, Job Brief, ENSURE a job brief for the job is performed.

4.3.5. Test Lead or Coordinator – PROVIDE any additional details during the brief as indicated by the Director, Nuclear Operations / Managers, Operations / Shift Outage Manager / Duty Manager or others designated in writing by the Director, Nuclear Operations which includes:

- Precautions, prerequisites, and initial conditions
- Procedure timeline starting with current plant conditions and detailing expected changes in plant conditions
- Potential risks, High Consequence Activities, and Margin of Safety
- Termination criteria, contingency actions, and restart criteria
- Expected changes in plant status and deviations from normal plant parameters, setpoints, and limits
- Restoration or transitions
- Significant internal and external operating experience [Ref: 5.2.5]

4.3.6. Test Lead or Coordinator – OBTAIN SM/CRS permission to start.

4.3.7. Director, Nuclear Operations / Managers, Operations / Shift Outage Manager / Duty Manager or others designated in writing by the Director, Nuclear Operations PERFORM Control Room oversight during critical evolutions in accordance with ODP-ZZ-00001, Operations Department - Code Of Conduct.

4.3.8. Test Lead or Coordinator – DIRECT performance of the procedure.

4.3.9. Test Lead or Coordinator – WHEN the procedure is either completed or terminated, NOTIFY SM/CRS of completion or termination.

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	3		
Equipment Control	**Group #**	N/A		
	K/A #	G2.2.5		
	Importance Rating	3.2		
Knowledge of the process for making design or operating changes to the facility				

Question # 96

A modification is being prepared that will impact operation of the plant.

Per APA-ZZ-00143, 10 CFR 50.59 AND 10 CFR 72.48 REVIEWS, who provides the FINAL approval of the 50.59 evaluation after Onsite Review Committee (ORC) review?

A. Director, Nuclear Operation

B. Director, Engineering Design

C. Senior Director, Nuclear Operation

D. Senior Director, Nuclear Engineering

Answer: C

Explanation:
APA-ZZ-00143, 10 CFR 50.59 AND 10 CFR 72.48 REVIEWS Section 3 "Responsibilities" step 3.2. Senior Director, Nuclear Operations – Provides final approval of 50.59 or 72.48 Evaluations.

A. Incorrect, per above, Plausible due to the fact that the stem states modification impact operation of the plant and per APA-ZZ-00143 step 3.7 provides initial approval of the 50. 59 review
B. Incorrect, per above. Plausible due to the fact that the stem states a modification and per APA-ZZ-00143 step 3.7 provides initial approval of the 50. 59 review
C. Correct, per explanation above,
D. Incorrect, Plausible if the candidate incorrectly believes it is the Senior Director of Engineering vs. Operation who is the FINAL approval of 50.59 reviews.

Technical Reference(s):
1. *APA-ZZ-00143, 10 CFR 50.59 AND 10 CFR 72.48 REVIEWS, Rev 16*

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Learning Objective: 10 CFR 50.59 training LP1 - 10 CFR 50.59 Applicability Determination and Screening Objective , Define and identify the important terms used in 10 CFR 50.59 and the associated review process:
"Method of performing or controlling a design function,"

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(3)

Comments:

Facility licensee procedures required to obtain authority for design and operating changes in the facility. [10 CFR 55.43(b)(3)]
Some examples of SRO exam items for this topic include:

- 10 CFR 50.59 screening and evaluation processes.

APA-ZZ-00143

10 CFR 50.59 AND 10 CFR 72.48 REVIEWS

Major Revision 016

10 CFR 50.59 AND 10 CFR 72.48 REVIEWS

TABLE OF CONTENTS

10 CFR 50.59 AND 10 CFR 72.48 REVIEWS

1.0 PURPOSE

1.1. The purpose of this procedure is to establish and maintain the program for the performance and documentation of evaluations required per 10 CFR 50.59 and 10 CFR 72.48.

1.2. The ultimate goal of this program is to determine if NRC approval is required prior to implementation of certain activities.

1.3. Within this procedure “10 CFR 50.59” or “10 CFR 72.48” is often abbreviated as “50.59” or “72.48” consistent with common usage and frequent reference to these regulations.

1.4. Within this procedure “10 CFR 72.212 Evaluation Report” is often abbreviated as “212 Report”.

-END OF SECTION-

2.0 SCOPE

> **NOTE**
>
> "FSAR" as used in this procedure includes both the Callaway Plant Unit 1 FSAR and the HI-STORM UMAX FSAR unless otherwise stated. Proposed activities affecting the Independent Spent Fuel Storage Installation (ISFSI) facility or dry storage cask must also be considered for potential impact on the 10CFR72.212 Report.

This procedure applies to implementation of certain activities such as the following:

2.1. Changes to Final Safety Analysis Report (FSAR) including items incorporated by reference.

2.2. Permanent and temporary plant modifications

2.3. Changes to plant procedures outlined, summarized or completely described in the FSAR (and which affect or direct how FSAR-described design functions are performed or controlled).

2.4. Proposed tests or experiments NOT described in the FSAR.

2.5. Revisions to NRC-approved analysis methodologies or assumptions as described in the FSAR.

2.6. Proposed compensatory actions (temporary changes to facility or procedures) to address degraded or non-conforming conditions, which are typically processed as activities referenced in Step 2.2 and/or 2.3 above.

2.7. Site-specific deviation from the Independent Spent Fuel Storage Installation (ISFSI) or dry storage cask design, including ancillaries, as described in the dry storage cask FSAR, including items incorporated by reference.

2.8. Changes to the 10 CFR 72.212 Report.

2.9. This procedure is used to support 50.59 and 72.48 Reviews for, or as part of, the following proceduralized programs and processes (as applicable):

- APA-ZZ-00090, Nuclear Safety Review Board
- APA-ZZ-00091, Onsite Review Committee
- APA-ZZ-00101, Processing Procedures, Manuals, and Desktop Instructions
- APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process
- APA-ZZ-00109, Software Quality Assurance Program
- APA-ZZ-00140, Environmental and Other Licensing Evaluations
- APA-ZZ-00323, Configuration Management Process
- APA-ZZ-00330, Preventive Maintenance Program
- APA-ZZ-00500, Corrective Action Program
- APA-ZZ-00520, Reporting Requirements and Responsibilities
- APA-ZZ-00540, Commitment Management Program
- APA-ZZ-00600, Design Change Control
- APA-ZZ-00604, Requests For Resolution
- APA-ZZ-00605, Temporary System Modifications
- EDP-ZZ-04015, Evaluating and Processing Requests for Resolution (RFR)
- ETP-ZZ-00015, Preparation, Review, Approval and Control of the Curve Book
- FDP-ZZ-00100, Final Safety Analysis Report (FSAR) Change/Revision Process
- FDP-ZZ-00101, Technical Specification Bases Control Program
- FDP-ZZ-00103, License Document Change Process
- MDP-ZZ-FS001, Control of Freeze Seals
- ODP-ZZ-00008, Night Orders - Standing Orders - Operations Information Reports
- WDP-ZZ-00022, Deficiency Reporting

-END OF SECTION-

3.0 RESPONSIBILITIES

3.1. **Nuclear Generation Directors and Managers** – Ensure overall implementation of this procedure for activities performed within their areas of responsibility.

3.2. **Senior Director, Nuclear Operations** – Provides final approval of 50.59 or 72.48 Evaluations.

3.3. **Onsite Review Committee (ORC)** - Reviews all documents requiring a 50.59 or 72.48 Evaluation and recommends approval to Senior Director, Nuclear Operations in accordance with APA-ZZ-00091, Onsite Review Committee.

3.4. **Nuclear Safety Review Board (NSRB)** - Reviews activities related to the 50.59 or 72.48 review process as described in APA-ZZ-00090, Nuclear Safety Review Board. *NSRB review of 10 CFR 50.59 evaluations is primarily an after-the-fact review, as NSRB review is NOT required for completion/approval of 50.59 or 72.48 evaluations during the processing of such evaluations.*

3.5. **Director, Training** - Provides and/or supports 50.59 and 72.48 review process initial training and recurrent training to designated personnel, and for maintenance of training records.

3.6. **Manager, Regulatory Affairs -** Ensures the following are performed by the designated or responsible Regulatory Affairs and Licensing person(s):

- Maintaining the 10 CFR 50.59 Resource Manual and this procedure.
- Regulatory Affairs and Licensing approval of all 50.59 and 72.48 Evaluations.
- Maintaining a numerical log of all 50.59 and 72.48 Evaluations, including providing a log number to preparer of each Evaluation.
- Preparing and submitting the 50.59 and 72.48 Summary Report in accordance with APA-ZZ-00520, Reporting Requirements and Responsibilities, and this procedure.
- Requesting that the HI-STORM UMAX CoC holder (Holtec) obtain NRC approval of any changes to the HI-STORM UMAX CoC required prior to implementing activities that require prior NRC approval.
- Requesting that the HI-STORM UMAX CoC holder (Holtec) make any required changes to the HI-STROM UMAX FSAR that may be generic in nature
- Ensuring that site-specific changes to, or deviations from, the HI-STORM UMAX FSAR are tracked in the 72.212 Report.

3.7. **Nuclear Generation Directors, Managers, and Supervising Engineers**:

- Ensure personnel within their organizations, who prepare or review Applicability Determinations, 50.59 and/or 72.48 Screens, and 50.59 and/or 72.48 Evaluations, meet the qualification requirements.
- Check the application of the 50.59 and 72.48 Review process to those activities listed in Section 2.0 prior to approval of the activity for implementation, as appropriate.
- Provide initial approval (by signature) of 50.59 and 72.48 Evaluations initiated by personnel under their supervision.

3.8. Preparer (Applicability Determination, 50.59 Screen, 50.59 Evaluation, 72.48 Screen, or 72.48 Evaluation)

3.8.1. Ensures qualifications are current in QualMaster.

3.8.2. Ensures he/she has adequate expertise in the technical or administrative matters related to the activity being reviewed.

3.8.3. Ensures the appropriate Applicability Determinations/Screen/Evaluation is performed and the corresponding form is completed.

3.8.4. Uses the 50.59 and/or 72.48 Resource Manual(s) for guidance when performing activities in this procedure.

3.8.5. For Applicability Determinations and 50.59 and 72.48 Screens performed by outside organizations, ensures the outside organizations have access to sufficient as-built design and licensing documents to support proper performance of the Applicability Determinations and Screens, as needed.

3.9. Reviewer (Applicability Determination, 50.59 Screen, 50.59 Evaluation, 72.48 Screen, or 72.48 Evaluation)

3.9.1. Ensures qualifications are current in QualMaster

3.9.2. Ensures he/she has adequate expertise in the technical or administrative matters related to the activity being reviewed.

3.9.3. Ensures the appropriate Applicability Determinations/screen/evaluation is performed and the corresponding form is completed.

3.9.4. Verifies the documentation is thorough and fully covers the proposed activity.

3.9.5. Uses the 50.59 and/or 72.48 Resource Manual(s) for guidance when performing activities in this procedure.

-END OF SECTION-

4.0 PROCEDURE INSTRUCTIONS

4.1. General

4.1.1. WHEN the scope of the Applicability Determination, 50.59 and/or 72.48 Screen, or 50.59 and/or 72.48 Evaluation extends beyond the expertise of the preparer or reviewer, SEEK assistance from other appropriately qualified personnel.

4.1.2. The 10 CFR 50.59 Resource Manual is consistent with NEI 96-07 and provides useful guidance when answering the screen and evaluation questions.

4.1.3. NEI 96-07, Appendix B, provides useful guidance when answering the 10 CFR 72.48 screen and evaluation questions. NEI 12-04 provides additional guidance on answering 72.48 screens and evaluations; however, it is not endorsed by the NRC and should only be used as a reference.

4.1.4. Regardless of the result of a 50.59 or 72.48 Review, ENSURE affected Primary Licensing documents are updated/revised in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process, as necessary.

4.1.5. IF the proposed activity being reviewed meets one of the following, INITIATE a change to the Callaway Plant, Unit 1 FSAR in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process:

- Existing information in the FSAR (e.g., design bases, safety analyses, or descriptions of existing structures, systems, components (SSCs) or functions, including how they are operated and controlled) is no longer correct as a result of the activity and needs to be updated.
- SSCs described in the FSAR are being removed.
- Functions or procedures described in the FSAR are eliminated.
- A new design bases or safety analyses (or associated description) is created which meets the criteria for addition to the FSAR in accordance with procedures APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process, and FDP-ZZ-00103, License Document Change Process, or FDP-ZZ-00100, Final Safety Analysis Report (FSAR) Change/Revision Process.

NOTE: Callaway will not be updating the HI-STORM UMAX FSAR, only authorizing site-specific deviations.

4.2. Limitations on the Implementation of a Proposed Activity

4.2.1. IF an activity within the scope of this procedure requires a 50.59 or 72.48 Screen, do NOT approve for implementation until the screen is completed.

4.2.2. IF an activity requires a 50.59 or 72.48 Evaluation, do NOT approve for implementation until ORC and the Senior Director, Nuclear Operations approve the evaluation with the conclusion that the activity does NOT require prior NRC approval.

4.2.3. IF an activity requires prior NRC approval or a change to Technical Specifications, do NOT approve for implementation until NRC approval is obtained.

NOTE: If Part 50 applies, a License Amendment Request (LAR) is submitted by Ameren to the NRC for review and approval.

NOTE: If Part 72 applies, a CoC Amendment is submitted by the CoC holder (Holtec International) to the NRC for review and approval.

4.2.4. PRIOR to implementation of an activity requiring NRC approval, SEEK approval from the NRC by processing the activity as either:

a. a license amendment in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process, or

b. a CoC amendment. Only the certificate holder (Holtec International) can request one. ENSURE someone from the Regulatory Affairs department (RARL group) is consulted to determine next steps.

4.3. Qualifications

4.3.1. Applicability Determination:

a. Personnel responsible for preparing or reviewing Applicability Determinations shall be competent in technical and administrative matters related to the proposed activity and shall have successfully completed the following training:

1. Prepare a 10 CFR 50.59 Applicability Determination and Screening: ESP/641A

   NOTE: ESP/641A requires completion of classroom training (T62.0482 6) and a Qualification Standard (T62.7641Q).

b. Personnel responsible for preparing or reviewing Applicability Determinations should complete recurrent training (T62.0482 8) once every three years to refresh or enhance their knowledge of the process.

4.3.2. **50.59 Screens:**

a. Personnel responsible for preparing or reviewing 50.59 Screens shall be competent in technical and administrative matters related to the proposed activity and shall be currently qualified per 4.3.1 above.

4.3.3. **50.59 Evaluations**:

a. Personnel responsible for preparing or reviewing 50.59 Evaluations shall be competent in technical and administrative matters related to the proposed activity, shall be currently qualified per 4.3.2 above, and shall have successfully completed the 50.59 evaluation process training:

1. Prepare a 10 CFR 50.59 Evaluation: ESP/648A

NOTE: This requires completion of classroom training (T62.0481 6) and a Qualification Standard (T62.7648 Q)

4.3.4. 72.48 Screens and Evaluations:

a. Personnel responsible for preparing or reviewing 72.48 Screens and/or Evaluations shall be competent in technical and administrative matters related to the proposed activity, shall be currently qualified per 4.3.1 above, and shall have successfully completed the 72.48 evaluation process training:

1. Prepare a 10 CFR 72.48 Screening and Evaluation: ESP/649A

NOTE: This requires completion of classroom training (T62.0649 6) and a Qualification Standard (T62.7649 Q)

-END OF SECTION-

4.4. Applicability Determination

NOTE

Activities that fall within the scope of this procedure are defined by 10 CFR 50.59 and 10 CFR 72.48 and clarified by guidance in the 50.59 and 72.48 Resource Manuals. Additional guidance may be available in departmental procedures for the specific activity being considered.

In some cases an activity may include a portion for which it is determined that a 50.59 or 72.48 screen **is NOT** needed, and a portion for which it is determined that a screen **is** needed. This situation is addressed on a CA2510, Applicability Determination Form. In such cases, an Applicability Determination would be sufficient for covering the portion of change that does NOT require a screen, but a screen would have to be prepared to cover the portion of change requiring a screen.

4.4.1. COMPLETE Applicability Determinations as follows:

a. For an activity of the type listed in Section 2.0, DETERMINE if the activity is within the scope of this procedure.

1. In certain cases, the applicable process procedure (or form) may in effect serve as the documented Applicability Determination to determine whether 50.59 or 72.48 applies.

   Eligible activities include those addressed by:

   - MDP-ZZ-FS001, Control of Freeze Seals.
   - APA-ZZ-00101, Processing Procedures, Manuals, and Desktop Instructions (or its associated CA0033, Procedure Review Form (PRF), or EMPRV procedure checklist).
   - ODP-ZZ-00008, Night Orders - Standing Orders - Operations Information Reports (or its associated CA0311, Night Order).

b. Using the guidance of the 10 CFR 50.59 Resource Manual to determine what change programs or processes apply to the activity, PERFORM an Applicability Determination by completing a CA2510, Applicability Determination Form.

c. Use the Storage Cask Certificate of Compliance (CoC) Checklist (Attachment 1) to determine which portion of the proposed activity involves other regulatory requirements and controls and is governed by the CoC, Cask FSAR, or 72.212 Report.

d. ASK a Qualified Reviewer to review the document and sign and date the Applicability Determination which also signifies his or her concurrence and resolution of comments.

e. IF the results of the Applicability Determination indicate that all portions of the activity do NOT require performance of a 50.59 or 72.48 Screen, ROUTE the completed Applicability Determination with the parent document.

f. IF results of the Applicability Determination indicate that a 10 CFR 50.59 and/or a 10 CFR 72.48 Review is required, the Applicability Determination shall be forwarded to qualified personnel to perform a 50.59 Screen and/or 72.48 Screen per Section 4.5 and/or 4.7, respectively.

g. IF the results of the Applicability Determination indicate that the activity requires processing in accordance with a program(s) other than 10 CFR 50.59, FOLLOW procedure(s) implementing the program(s) indicated by review.

NOTE

Applicability Determinations prepared by outside agencies, in accordance with approved Quality Assurance programs, may be used to fulfill the requirements of the Applicability Determination process specifically explained in this procedure. The specific Applicability Determination form may be one developed by either the outside agency or Callaway (CA2510).

4.4.2. WHEN using or applying Applicability Determinations prepared by others, ENSURE the following guidance is met:

a. IF using an Applicability Determination prepared by outside agencies, ENSURE the Applicability Determination prepared by an outside agency is approved by a Callaway Qualified Reviewer.

b. ENSURE outside agency has access to sufficient as-built design and licensing documents to perform an Applicability Determination.

c. IF the results of the review indicate that the activity requires processing in accordance with a program(s) other than the 10 CFR 50.59 and/or 10 CFR 72.48 Review program, FOLLOW procedure(s) implementing the program(s).

d. Indicate approval by signing the Applicability Determination.

-END OF SECTION-

4.5. 50.59 Screen (10 CFR 50.59(c)(1))

NOTE

In certain cases, an applicable process procedure (or form) may in effect serve as the documented 50.59 Screen. Eligible activities include those addressed by EDP-ZZ-04015, Evaluating and Processing Requests for Resolution (RFR), (or its associated CA2842, 10CFR50.71(e)/50.59 Screen for Replacement Item Equivalency (RIE)) for replacement item equivalencies only.

4.5.1. *Preparers* - WHEN directed by the Applicability Determination or the 10 CFR 50.59 Resource Manual, COMPLETE a CA2511, 50.59 Screen Form.

a. LIST documents (licensing basis/technical/other) reviewed in preparing the 50.59 Screen, including section numbers. *Relevant documents listed on parent document need NOT be listed on the 50.59 Screen.*

b. IDENTIFY relevant FSAR-described SSCs and associated design function(s). *Identification of design function is a prerequisite to being able to effectively answer screening questions on the 50.59 Screen.*

c. COMPLETE the screening questions listed on the form by checking the YES or NO box as appropriate.

d. ATTACH an additional page, if needed, to provide justification for any "YES" response.

e. IF overall conclusion is that a 50.59 Evaluation is NOT required, PROVIDE written justification in the section provided for this.

f. ENSURE justification contains sufficient discussion that any subsequent Qualified Reviewer, knowledgeable in the subject area, can recognize the essential argument leading to conclusion.

g. SIGN, DATE, and FORWARD the completed 50.59 Screen to the Qualified Reviewer for review and concurrence.

h. WHEN comments/questions are resolved, CHECK that the Qualified Reviewer documented his/her review with a signature and date which signifies his/her concurrence with the results and adequacy of justification.

i. IF results of the 50.59 Screen indicate the activity does NOT require performance of a 50.59 Evaluation, ROUTE the completed 50.59 Screen with the documentation associated with the proposed activity being screened.

Step 4.5.1 Cont'd

j. IF it is determined that a change to the Technical Specifications is required, PROCESS a license amendment request in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process (FDP-ZZ-00103, License Document Change Process, or FDP-ZZ-00102, Operating License (OL) Change/Revision Process). *The activity then requires NRC approval of the associated license amendment prior to implementation.*

k. IF a 50.59 Evaluation is required, FORWARD the 50.59 Screen to an Evaluator to have a 50.59 Evaluation performed in accordance with Section 4.6. *The preparer of the 50.59 Screen may perform the 50.59 Evaluation if qualified as an Evaluator.*

l. FORWARD a copy of the completed 50.59 Screen to the 50.59 program owner (Regulatory Affairs and Licensing).

4.5.2. WHEN using or applying 50.59 Screens prepared by others, ENSURE the following guidance is met:

NOTE

50.59 Screens prepared by outside agencies, in accordance with approved Quality Assurance programs, may be used to fulfill the requirements of 10 CFR 50.59 in lieu of the 50.59 Screen process specifically explained in this procedure. The specific 50.59 Screen form may be one developed by either the outside agency or Callaway (CA2511).

a. IF using a 50.59 Screen prepared by outside agencies, ENSURE the 50.59 Screen prepared by an outside agency is approved by a Callaway Qualified Reviewer, which may require resolution of comments and augmenting the outside agency's 50.59 Screen, as necessary, to provide a complete 50.59 screen for Callaway. Indicate approval by signing the 50.59 Screen.

b. ENSURE outside agency has access to sufficient as-built design and licensing documents to perform a 50.59 Screen.

c. IF results of the 50.59 Screen indicate the activity does NOT require performance of a 50.59 Evaluation, ROUTE the completed 50.59 Screen with the documentation associated with the proposed activity being screened.

Step 4.5.2 Cont'd

d. IF it is determined that a change to the Technical Specifications is required, PROCESS a license amendment request in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process (FDP-ZZ-00103, License Document Change Process, or FDP-ZZ-00102, Operating License (OL) Change/Revision Process). *The activity then requires NRC approval of the associated license amendment prior to implementation.*

e. IF a 50.59 Evaluation is required, FORWARD the 50.59 Screen to an Evaluator to have a 50.59 Evaluation performed in accordance with Section 4.6. *The preparer of the 50.59 Screen may perform the 50.59 Evaluation if qualified as an Evaluator.*

f. FORWARD a copy of the completed 50.59 Screen to the 50.59 program owner (Regulatory Affairs and Licensing).

-END OF SECTION-

4.6. 50.59 Evaluation (10 CFR 50.59(c)(2))

NOTE

Preparers and Qualified Reviewers of 50.59 Evaluations are qualified as Evaluators with respect to the 50.59 Review process.

The purpose of a 50.59 Evaluation is to determine if a proposed activity requires prior NRC approval via a License Amendment in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process.

4.6.1. *Preparers* - WHEN directed by a 50.59 Screen or the 10 CFR 50.59 Resource Manual, COMPLETE and PROCESS a CA2512, 50.59 Evaluation Form. [Ref: 5.2.33]

a. OBTAIN a log number for the 50.59 Evaluation from the 50.59 program owner (Regulatory Affairs and Licensing) and RECORD log number on the 50.59 Evaluation in the space provided.

b. In the "Activity Description" box, PROVIDE a written summary description of the proposed activity.

c. Using applicable Sections of the 10 CFR 50.59 Resource Manual as necessary, COMPLETE a separate written response, which includes sufficient discussion to support the conclusion reached, for each of the eight questions on the 50.59 Evaluation form.

   1. IF the proposed activity involves a new or modified operator action that supports a design function credited in the safety analysis, REVIEW questions on CA2512a, Questions for New or Modified Operator Action(s) Form, as part of the 10CFR50.59 Evaluation. [Ref: 5.1.30]

d. Based on the responses prepared for the eight questions addressed on the 50.59 Evaluation form, PROVIDE, in the "Summary of Evaluation" box, a written summary of the evaluation, including the conclusions reached.

e. SIGN, DATE and FORWARD the completed 50.59 Evaluation package to a Qualified Reviewer for review and concurrence.

f. WHEN comments/questions are resolved, CHECK that the Qualified Reviewer has documented his/her review with his/her signature and date signifying concurrence with the results including justification for each of the responses to the eight questions on the form.

g. OBTAIN concurrence (sign-off) from Regulatory Affairs and Licensing.

h. SUBMIT the completed 50.59 Evaluation to the cognizant Supervising Engineer for approval.

NOTE

A 50.59 Evaluation may incorporate other 50.59 Evaluations, Calculations, or Engineering Disposition Forms by reference. For example, detailed descriptions of components, functions, applicable codes, standards and specifications need NOT be described in a 50.59 Evaluation when this information is already detailed in a STARS-ENG-5001-8.1, Engineering Disposition Form prepared in accordance with APA-ZZ-00600, Design Change Control.

4.6.2. WHEN incorporating other 50.59 Evaluations, Calculations, or Engineering Disposition Forms by reference into the 50.59 Evaluation, REFERENCE the Engineering Disposition Form and augment the reference with what information is needed to provide an adequate description or complete reference.

4.6.3. IF the Engineering Disposition Form provides critical information for the 50.59 Evaluation and the Engineering Disposition Form is subsequently revised (except for minor or editorial changes), REVIEW the 50.59 Evaluation and revised Engineering Disposition Form with the same level of detail as the original 50.59 Evaluation.

4.6.4. IF an activity is changed such that the 50.59 Evaluation may no longer be valid, CONFIRM the existing 50.59 Evaluation remains valid, REVISE/REPLACE the existing Evaluation, or PREPARE a new/follow-up Evaluation, as appropriate.

4.6.5. IF the 50.59 Evaluation determines that NRC approval is required, as indicated by a “YES” response to any of the eight questions on the 50.59 Evaluation, OBTAIN a license amendment PRIOR to implementation of the change or activity:

a. IF it is decided to continue with the change or activity, PERFORM the following:

1. PROCESS a license amendment request in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process (FDP-ZZ-00103, License Document Change Process, or FDP-ZZ-00102, Operating License (OL) Change/Revision Process).

2. PROVIDE a copy of the 50.59 Evaluation (as processed to this point) to the 50.59 program owner (Regulatory Affairs and Licensing).

b. IF it is decided to NOT continue with the change or activity, FILE/DISCARD paperwork as desired.

4.6.6. WHEN using or applying 50.59 Evaluations prepared by others, ENSURE the following guidance is met:

NOTE

50.59 Evaluations prepared by outside agencies, in accordance with approved Quality Assurance programs, may be used to fulfill the requirements of 10 CFR 50.59 in lieu of 50.59 Evaluation process specifically explained in this procedure.

a. IF using a 50.59 Evaluation prepared by outside agencies to address specific issues within a 50.59 Evaluation prepared by AmerenUE, REFERENCE the 50.59 Evaluation prepared by the outside agency, within the AmerenUE 50.59 Evaluation.

b. ENSURE outside agency has access to sufficient as-built design and licensing documents to perform a 50.59 Evaluation.

c. ENSURE the 50.59 Evaluation prepared by an outside agency is approved by the following, which may require resolution of comments and augmenting the outside agency's 50.59 Evaluation, as necessary, to provide a complete evaluation for Callaway:

- Evaluator
- Qualified Reviewer
- Regulatory Affairs and Licensing
- Cognizant Supervising Engineer

d. ENSURE the completed 50.59 Evaluation prepared by an outside agency receives ORC review and approval (per Step 4.6.7 below), either by itself (if it is a stand-alone document) or in conjunction with the Callaway 50.59 Evaluation (CA2512) that references it.

e. FILE the in-house 50.59 Evaluation, which documents the above review process, along with the outside agency's 50.59 Evaluation. (O025 3.01 and O025 5.02)

4.6.7. COMPLETE review and approval of a 50.59 Evaluation by obtaining ORC review, as follows.

a. OBTAIN ORC review and Senior Director, Nuclear Operations approval in accordance with APA-ZZ-00091, Onsite Review Committee.

Step 4.6.7 Cont'd

b. ENSURE ORC review and Senior Director, Nuclear Operations approval is documented on the 50.59 Evaluation.

4.6.8. FORWARD copy of approved 50.59 Evaluation to the 50.59 program owner (Regulatory Affairs and Licensing).

4.6.9. IF a previously approved 50.59 Evaluation is to be used and referenced, ENSURE compliance with following steps or guidance:

a. IF the 50.59 Evaluation for the previous activity remains applicable, bounding, and valid for the current activity AND the current activity is part of a previous modification to facility, procedures, or same test/experiment (e.g., the previous activity might be a Design Modification and the current activity the FSAR Change Notice for that Design Modification), REFERENCE the previous 50.59 Evaluation on the AD or 50.59 Screen for the current activity instead of preparing a new 50.59 Evaluation, if appropriate.

b. IF the 50.59 Evaluation for the previous activity remains applicable and valid for the current activity, but not bounding, AND the current activity is essentially identical to the previous activity, either REVISE the previous 50.59 Evaluation to encompass the current activity within the 50.59 Evaluation's scope, OR REFERENCE the previous 50.59 Evaluation on a new 50.59 Evaluation.

c. CONSIDER whether substantial changes to the plant's licensing basis have been made since the previous Evaluation was performed in order to determine whether a new Evaluation needs to be performed for the more recent activity.

d. REVIEW references in the previous 50.59 Evaluation to ensure changes have NOT been made which could impact the validity of the previous 50.59 Evaluation for the current activity.

e. INCLUDE results of review and a brief discussion of applicability of the previous 50.59 Evaluation to the current activity with the 50.59 Evaluation.

f. If desired, PERFORM a new 50.59 Evaluation referencing previous 50.59 Evaluation in appropriate sections.

-END OF SECTION-

4.7. 72.48 Screen (10 CFR 72.48(c)(1))

NOTE

Preparers and Qualified Reviewers of 72.48 Screens are qualified as Screeners with respect to the 72.48 Review process, although 72.48 Screens may also be prepared and reviewed by Evaluators.

4.7.1. *Preparers* - WHEN directed by an Applicability Determination or the 10 CFR 50.59 Resource Manual, COMPLETE a CA3145, 72.48 Screening Form.

a. LIST documents (licensing basis/technical/other) reviewed in preparing the 72.48 Screen, including section numbers. *The COC amendment and HI-STORM UMAX FSAR revision number should be listed.*

b. IDENTIFY relevant UMAX FSAR-described SSCs and associated design function(s). *Identification of design function is a prerequisite to being able to effectively answer screening questions on the 72.48 Screen.*

c. COMPLETE the screening questions listed on the form by checking the YES or NO box as appropriate.

d. ATTACH an additional page, if needed, to provide justification for any "YES" response.

e. IF overall conclusion is that a 72.48 Evaluation is NOT required, PROVIDE written justification in the section provided for this.

f. ENSURE justification contains sufficient discussion that any subsequent Qualified Reviewer, knowledgeable in the subject area, can recognize the essential argument leading to conclusion.

g. SIGN, DATE, and FORWARD the completed 72.48 Screen to the Qualified Reviewer for review and concurrence.

h. WHEN comments/questions are resolved, CHECK that the Qualified Reviewer documented his/her review with a signature and date which signifies his/her concurrence with the results and adequacy of justification.

i. IF results of the 72.48 Screen indicate the activity does NOT require performance of a 72.48 Evaluation, ROUTE the completed 72.48 Screen with the documentation associated with the proposed activity being screened.

j. IF it is determined that a change to the Certificate of Compliance is required, CONTACT Licensing department for further guidance. *The activity then requires NRC approval of the associated license amendment prior to implementation.*

k. IF it is determined that a change to the 72.212 Report is required, PROCESS in accordance with FDP-ZZ-00105.

l. IF a 72.48 Evaluation is required, PERFORM the 72.48 Evaluation in accordance with Step 4.8.

m. FORWARD a copy of the completed 72.48 Screen to the 72.48 program owner (Regulatory Affairs and Licensing).

4.7.2. WHEN using or applying 72.48 Screens prepared by others, ENSURE the following guidance is met:

NOTE

72.48 Screens may be prepared by outside agencies (i.e., Holtec) or by Callaway staff.

The specific 72.48 Screen form may be one developed by either the outside agency or Callaway (CA3145).

a. IF using a 72.48 Screen prepared by an outside agency, REVIEW it for general quality and compliance, impacts on plant procedures, and the 72.212 Report.

b. ENSURE outside agency has access to the appropriate Callaway specific information/documents to perform a 72.48 Screen.

c. IF results of the 72.48 Screen indicate the activity does NOT require performance of a 72.48 Evaluation, ROUTE the completed 72.48 Screen with the documentation associated with the proposed activity being screened.

d. IF it is determined that a change to the Certificate of Compliance is required, CONTACT Regulatory Affairs department for further guidance. *The activity then requires NRC approval of the associated license amendment prior to implementation.*

e. IF it is determined that a change to the 72.212 Report is required, PROCESS in accordance with FDP-ZZ-00105.

f. IF a 72.48 Evaluation is required, PERFORM the 72.48 Evaluation in accordance with Step 4.8.

g. FORWARD a copy of the completed 72.48 Screen to the 72.48 program owner (Regulatory Affairs and Licensing).

-END OF SECTION-

4.8. 72.48 Evaluation (10 CFR 72.48(c)(2))

NOTE

The purpose of a 72.48 Evaluation is to determine if a proposed activity requires prior NRC approval via a CoC Amendment. Only the certificate holder (Holtec International) can request a CoC amendment.

4.8.1. *Preparer* - WHEN directed by a 72.48 Screen or the 10 CFR 50.59 Resource Manual, COMPLETE and PROCESS a CA3146, 72.48 Evaluation Form.

a. OBTAIN a log number for the 72.48 Evaluation from the 72.48 program owner (Regulatory Affairs and Licensing) and RECORD log number on the 72.48 Evaluation in the space provided.

b. In the "Activity Description" box, PROVIDE a written summary description of the proposed activity.

c. Using applicable Sections of the 72.48 Resource Manual as necessary, COMPLETE a separate written response, which includes sufficient discussion to support the conclusion reached, for each of the eight questions on the 72.48 Evaluation form.

d. Based on the responses prepared for the eight questions addressed on the 72.48 Evaluation form, PROVIDE, in the "Summary of Evaluation" box, a written summary of the evaluation, including the conclusions reached.

e. SIGN, DATE and FORWARD the completed 72.48 Evaluation package to a Qualified Reviewer for review and concurrence.

f. WHEN comments/questions are resolved, CHECK that the Qualified Reviewer has documented his/her review with his/her signature and date signifying concurrence with the results including justification for each of the responses to the eight questions on the form. [Ref: 5.2.30, 5.2.31, 5.2.34]

g. OBTAIN concurrence (sign-off) from Regulatory Affairs and Licensing.

h. SUBMIT the completed 72.48 Evaluation to the cognizant Supervising Engineer for approval.

4.8.2. IF the 72.48 Evaluation determines that NRC approval is required, as indicated by a "YES" response to any of the eight questions on the 72.48 Evaluation, CONTACT the Regulatory Affairs department (RARL group). A CoC amendment is required PRIOR to implementation of the change or activity.

a. RARL will coordinate actions with the CoC holder if the effects of the 72.48 Evaluation meet one of the following:

1. The existing generic information in the cask FSAR (e.g., the design bases, safety analysis, or descriptions of existing SSCs or functions) is no longer correct and needs to be updated.

2. Generic functions or procedures described in the cask FSAR are modified or eliminated.

3. Callaway Plant has made a change to the cask design.

4.8.3. The 72.48 Evaluator shall ensure that site-specific changes to, or deviations from the cask FSAR are tracked in the 212 Report. The CoC holder does not revise the generic cask FSAR to reflect site-specific changes.

4.8.4. WHEN using or applying 72.48 Evaluations prepared by others, ENSURE the following guidance is met:

a. IF using a 72.48 Evaluation prepared by an outside agency, REVIEW it for general quality and for compliance, impacts on plant procedures, and the 72.212 Report.

b. ENSURE outside agency has access to sufficient as-built design and licensing documents to perform a 72.48 Evaluation.

c. ENSURE the 72.48 Evaluation prepared by an outside agency is approved by the following, which may require resolution of comments and augmenting the outside agency's 72.48 Evaluation, as necessary, to provide a complete evaluation for Callaway:

- Evaluator
- Qualified Reviewer
- Regulatory Affairs and Licensing
- Cognizant Supervising Engineer

d. ENSURE the completed 72.48 Evaluation prepared by an outside agency receives ORC review and approval (per Step 4.8.5 below), either by itself (if it is a stand-alone document) or in conjunction with the Callaway 72.48 Evaluation (CA3146) that references it.

e. FILE the in-house 72.48 Evaluation, which documents the above review process, along with the outside agency's 72.48 Evaluation. (O025 3.01 and O025 5.02)

4.8.5. COMPLETE review and approval of a 72.48 Evaluation by obtaining ORC review, as follows.

a. OBTAIN ORC review and Senior Director, Nuclear Operations approval in accordance with APA-ZZ-00091, Onsite Review Committee.

b. ENSURE ORC review and Senior Director, Nuclear Operations approval is documented on the 72.48 Evaluation.

4.8.6. FORWARD copy of approved 72.48 Evaluation to the 72.48 program owner (Regulatory Affairs and Licensing).

4.8.7. IF a previously approved 72.48 Evaluation is to be used and referenced, ENSURE compliance with following steps or guidance:

a. IF the 72.48 Evaluation for the previous activity remains applicable, bounding, and valid for the current activity AND the current activity is part of a previous modification to facility, procedures, or same test/experiment, REFERENCE the previous 72.48 Evaluation on the AD or 72.48 Screen for the current activity instead of preparing a new 72.48 Evaluation, if appropriate.

b. IF the 72.48 Evaluation for the previous activity remains applicable and valid for the current activity, but not bounding, AND the current activity is essentially identical to the previous activity, either REVISE the previous 72.48 Evaluation to encompass the current activity within the 72.48 Evaluation's scope, OR REFERENCE the previous 72.48 Evaluation on a new 72.48 Evaluation. [Ref: 5.2.38]

c. CONSIDER whether substantial changes to the ISFSI licensing basis have been made since the previous Evaluation was performed in order to determine whether a new Evaluation needs to be performed for the more recent activity.

d. REVIEW references in the previous 72.48 Evaluation to ensure changes have NOT been made which could impact the validity of the previous 72.48 Evaluation for the current activity.

e. INCLUDE results of review and a brief discussion of applicability of the previous 72.48 Evaluation to the current activity with the 72.48 Evaluation.

f. If desired, PERFORM a new 72.48 Evaluation referencing previous 72.48 Evaluation in appropriate sections.

-END OF SECTION-

4.9. Revisions to 50.59 or 72.48 Screens and Evaluations

4.9.1. IF a revision to a previously approved 50.59 or 72.48 Screen/Evaluation is desired, ENTER the next revision number on 50.59 or 72.48 Screen/50.59 or 72.48 Evaluation and APPLY change bars.

4.9.2. PERFORM the same level of review and approval on the revised 50.59 or 72.48 Screen/Evaluation as on the initial 50.59 or 72.48 Screen/50.59 or 72.48 Evaluation.

4.9.3. IF a 50.59 or 72.48 Screen/Evaluation was prepared under a previous version of this procedure, UPDATE the portions being revised to be consistent with format and content required by the current revision of this procedure, including the current applicable form(s).

-END OF SECTION-

4.10. Reports

4.10.1. *Responsible Regulatory Affairs and Licensing Engineer/Specialist* - PREPARE and SUBMIT a periodic report to the NRC containing a brief description of all activities (changes, tests, and experiments) that required a 50.59 or 72.48 Evaluation, including a summary of each of those evaluations, for activities completed during the reporting period, if NOT previously reported.

a. PREPARE and SUBMIT the report in accordance with APA-ZZ-00520, Reporting Requirements and Responsibilities.

b. REVIEW all completed 50.59 and 72.48 Evaluations and SELECT those that are to be included in the scope of the periodic report.

c. DOCUMENT in the report those activities (design/procedure changes, tests and experiments) implemented within the reporting period for Callaway Plant and determined NOT to require a License Amendment in accordance with 10 CFR 50.59(d)(2) or a CoC Amendment in accordance with 10 CFR 72.48 (d)(2).

d. FILE (submit) report at frequency specified in APA-ZZ-00520, Reporting Requirements and Responsibilities, in accordance with 10 CFR 50.59(d)(2) or 10 CFR 72.48 (d)(2).

e. IF desired, FILE this report along with the Callaway Plant FSAR update as provided by:

- 10 CFR 50.59(d)(2)
- 10CFR 50.71(e)(4)
- APA-ZZ-00520, Reporting Requirements and Responsibilities
- FDP-ZZ-00110, Electronic Licensing Document Revision, or FDP-ZZ-00100, Final Safety Analysis Report (FSAR) Change/Revision Process

4.10.2. To allow sufficient time for report generation, review and approval, SET an administrative cutoff date no later than two months prior to the report due date.

4.10.3. WHEN separate 50.59 or 72.48 Evaluations exist for a single activity, REPORT only one in the periodic report and clearly REFERENCE additional 50.59 or 72.48 Evaluations in the description of the one reported. *Avoid double reporting of the same activity in the periodic report.*

-END OF SECTION-

4.11. <u>Control of 10 CFR 50.59 and 10 CFR 72.48 Resource Manuals</u>

4.11.1. *Designated Individual in Regulatory Affairs and Licensing Department* - PREPARE and ISSUE the 50.59 and 72.48 Resource Manuals (or revisions thereof).

4.11.2. ENSURE the 50.59 and 72.48 Resource Manuals provide appropriate guidance to conduct changes in accordance with 10 CFR 50.59 and 10 CFR 72.48.

4.11.3. *Manager, Regulatory Affairs* - REVIEW and APPROVE the 50.59 and 72.48 Resource Manuals.

4.11.4. FILE the 50.59 and 72.48 Resource Manuals in accordance with Section 6.0.

4.11.5. ISSUE the 50.59 and 72.48 Resource Manuals for use in accordance with APA-ZZ-00200, Document Control.

-END OF SECTION-

4.12. <u>10 CFR 50.59 And 10 CFR 72.48 Program Assessment/Monitoring</u>

Responsible Regulatory Affairs and Licensing Engineer/Specialist – MONITOR and ASSESS the 10 CFR 50.59 and 10 CFR 72.48 program performance, effectiveness, and conformance to regulatory and industry guidance by performing any of the following activities on a periodic or as-needed basis. Means of accomplishing this are NOT limited to just the following activities.

- Self-assessments using the appropriate self-assessment guidelines.
- Benchmarking Callaway's 10 CFR 50.59 And 10 CFR 72.48 program (procedures and products) against those from other facilities.
- Quality Review Team (QRT) meetings to review a sampling of 50.59 and 72.48 review products (i.e., completed ADs, Screens, and Evaluations) for completeness, thoroughness, accuracy and conformance to the guidance of this procedure and the 50.59 and 72.48 Resource Manuals.

-END OF SECTION-

5.0 REFERENCES

5.1. Implementing

5.1.1. APA-ZZ-00090, Nuclear Safety Review Board

5.1.2. APA-ZZ-00091, Onsite Review Committee

5.1.3. APA-ZZ-00101, Processing Procedures, Manuals, and Desktop Instructions

5.1.4. APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process

5.1.5. APA-ZZ-00109, Software Quality Assurance Program

5.1.6. APA-ZZ-00140, Environmental and Other Licensing Evaluations

5.1.7. APA-ZZ-00200, Document Control

5.1.8. APA-ZZ-00323, Configuration Management Process

5.1.9. APA-ZZ-00330, Preventive Maintenance Program

5.1.10. APA-ZZ-00500, Corrective Action Program

5.1.11. APA-ZZ-00520, Reporting Requirements and Responsibilities

5.1.12. APA-ZZ-00540, Commitment Management Program

5.1.13. APA-ZZ-00600, Design Change Control

5.1.14. APA-ZZ-00604, Requests For Resolution

5.1.15. APA-ZZ-00605, Temporary System Modifications

5.1.16. EDP-ZZ-04015, Evaluating and Processing Requests for Resolution (RFR)

5.1.17. ETP-ZZ-00015, Preparation, Review, Approval and Control of the Curve Book

5.1.18. FDP-ZZ-00100, Final Safety Analysis Report (FSAR) Change/Revision Process

5.1.19. FDP-ZZ-00101, Technical Specification Bases Control Program

5.1.20. FDP-ZZ-00102, Operating License (OL) Change/Revision Process

5.1.21. FDP-ZZ-00103, License Document Change Process

5.1.22. FDP-ZZ-00110, Electronic Licensing Document Revision

5.1.23. FDP-ZZ-00105, 10 CFR 72.212 Report Maintenance

5.1.24. MDP-ZZ-FS001, Control of Freeze Seals

5.1.25. ODP-ZZ-00008, Night Orders - Standing Orders - Operations Information Reports

5.1.26. WDP-ZZ-00022, Deficiency Reporting

5.1.27. CA2510, Applicability Determination Form

5.1.28. CA2511, 50.59 Screen Form

5.1.29. CA2512, 50.59 Evaluation Form

5.1.30. CA2512a, Questions for New or Modified Operator Action(s) Form

5.1.31. CA3145, 72.48 Screen Form

5.1.32. CA3146, 72.48 Evaluation Form

5.2. Developmental

5.2.1. APA-ZZ-00315, Configuration Risk Management Program

5.2.2. APA-ZZ-00320, Work Execution

5.2.3. APA-ZZ-00340, Surveillance Program Administration

5.2.4. APA-ZZ-00356, Pump and Valve Inservice Test Program

5.2.5. EDP-ZZ-01003, Inservice Inspection Program

5.2.6. EDP-ZZ-01129, Callaway Plant Risk Assessment

5.2.7. GDP-ZZ-00600, Processing of Revisions and Change Notices to the Operating Quality Assurance Manual (OQAM)

5.2.8. KDP-ZZ-00400, RERP Impact Evaluations and Changes

5.2.9. SDP-ZZ-00030, Security Plan Revision Process

5.2.10. 10 CFR 50.59, Changes, Tests & Experiments

5.2.11. 10 CFR 50.59(d)(2)

5.2.12. 10CFR72.48, Changes, Tests & Experiments

5.2.13. 10CFR 50.71(e)(4)

5.2.14. NEI 96-07, Guidelines for 10 CFR 50.59 Implementation

5.2.15. NEI 96-07 Appendix B, Guidelines for 10 CFR 72.48 Implementation

5.2.16. NEI 12-04, Guidelines for 10 CFR 72.48 Implementation

a. NEI 12-04 has been submitted by NEI to the NRC for endorsement. It is intended to replace NEI 96-07 Appendix B when and if it is endorsed. Until that time, it is a worthwhile desktop guide.

5.2.17. NRC Regulatory Guide 1.187, Guidelines for implementation of 10 CFR 50.59

5.2.18. NRC Regulatory Guide 3.72, Guidelines for implementation of 10 CFR 72.48

5.2.19. 10 CFR 50.59 Resource Manual

5.2.20. 10CFR 72.48 Resource Manual

5.2.21. OQAM 1.20

5.2.22. OQAM 1.20.7

5.2.23. OQAM 1.20.8

5.2.24. OQAM 1.21

5.2.25. OQAM 1.21.10

5.2.26. OQAM 2.10

5.2.27. OQAM 3.14

5.2.28. OQAM 3.18

5.2.29. OQAM 3.19

5.2.30. OQAM 5.6.1

5.2.31. OQAM 5.6.2

5.2.32. OQAM 5.6.4

5.2.33. OQAM 17.11.2

5.2.34. OQAM Appendix A (*Obligation 01791*)

5.2.35. Technical Specification AC 5.1.1

5.2.36. FSAR Appendix 3A Regulatory Guide 1.187 Commitment

5.2.37. CARS 199700220

5.2.38. CARS 200101272

5.2.39. COMN 43356

5.2.40. NEI 96-07 Appendix E, “User’s Guide for NEI 96-07, Revision 1, ‘Guidelines for 10 CFR 50.59 Implementation’ ”

5.2.41. 10 CFR 72.48, “Changes, Tests and Experiments”

5.2.42. NEI 96-07, Appendix B, “Guidelines for 10 CFR 72.48 Implementation”

5.2.43. NRC Regulatory Guide 3.72, “Guideline for implementation of 10 CFR 72.48

5.2.44. HI-STORM UMAX Certificate of Compliance No. 1040

5.2.45. HI-STORM UMAX Final Safety Analysis Report

6.0 RECORDS

6.1. File completed documents listed below as a QA Record with applicable parent document: [Ref: 5.2.33]

- CA2510, Applicability Determination Form
- CA2511, 50.59 Screen Form
- CA2512, 50.59 Evaluation Form

6.2. 10 CFR 50.59 Resource Manual - File Number A230.0080.

6.3. 10 CFR 72.48 Resource Manual - File Number A230.0080

6.4. Records generated by performance of 50.59 Reviews are filed with the parent documents processed by the programs listed in Step 2.9 (as implemented in departmental procedures), except that copies of completed 50.59 Screens and Evaluations are also provided to the 50.59 program owner (Regulatory Affairs and Licensing). (10 CFR 50.59(d)(3))

6.5. The CA2510, Applicability Determination Form, and the CA2511, 50.59 Screen Form, are provided to the Regulatory Affairs Clerk for QA Record processing to File Number A170.0010.

-END OF SECTION-

7.0 DEFINITIONS/ACRONYMS

7.1. **50.59 And 72.48 Review** – A process for reviewing a proposed activity by the 50.59 and/or 72.48 Screening criteria and as necessary, subsequent Evaluation criteria to determine if NRC approval is required prior to implementation of an activity. A 50.59 or 72.48 Review may consist of an Applicability Determination, a 50.59 and/or 72.48 Screen, a 50.59 and/or 72.48 Evaluation, as appropriate.

7.2. **Applicability Determination** – The process for determining if a proposed activity has portions controlled by more specific regulations, thereby, excluding those portions from review under 10 CFR 50.59 and/or 10 CFR 72.48.

7.3. **50.59 Screen** – An assessment to determine whether an activity requires evaluation in accordance with 10 CFR 50.59. 50.59 Screens are documented via a CA2511, 50.59 Screen Form. 50.59 Screens are only prepared and reviewed by personnel who are qualified.

7.4. **50.59 Evaluation** – A documented evaluation against the eight criteria in 10 CFR 50.59(c)(2) to determine if a proposed change, test or experiment requires prior NRC approval via a license amendment under 10 CFR 50.90. 50.59 Evaluations are documented via a CA2512, 50.59 Evaluation Form. 50.59 Evaluations are only prepared and reviewed by personnel who are qualified.

7.5. **72.48 Screen** – An assessment to determine whether an activity requires evaluation in accordance with 10 CFR 72.48. A 72.48 screen is also used to determine if a “direct” CoC amendment is required, if a 72.212 Report change is required, or if there is a Part 72 impact on the canister from the proposed activity. 72.48 Screens are documented via a CA3145, 72.48 Screen Form. 72.48 Screens are only prepared and reviewed by personnel who are qualified.

7.6. **72.48 Evaluation** – A documented evaluation against the eight criteria in 10 CFR 72.48(c)(2) to determine if a proposed change, test or experiment requires prior NRC approval via a CoC amendment under 10 CFR 72.48. <u>NRC approval must be obtained by Holtec, the CoC holder.</u> 72.48 Evaluations are documented via a CA3146, 72.48 Evaluation Form. 72.48 Evaluations are only prepared and reviewed by personnel who are qualified.

7.7. **Final Safety Analysis Report (FSAR)** – The Callaway Plant Unit 1 Final Safety Analysis Report referred to in this procedure is the most recently updated FSAR submitted to the NRC as required by 10 CFR 50.71(e), including approved FSAR change notices issued in accordance with APA-ZZ-00108, Primary Licensing Documents: Change/Revision Process, and FDP-ZZ-00103, License Document Change Process, or FDP-ZZ-00100, Final Safety Analysis Report (FSAR) Change/Revision Process.

- In accordance with 10 CFR 50.34(b) the FSAR was part of the original application for Callaway's Operating License, but it continues to be maintained and periodically updated in accordance with 10 CFR 50.71(e). The FSAR contains information that describes the facility, sets forth the facility's design bases and limits on its operation, and presents a safety analysis.
- The FSAR includes information on site evaluation factors, information on organizational responsibilities, administrative controls, and plans for conducting normal operations and for coping with emergencies.
- The FSAR includes documents that are referenced as part of the description, but not documents merely listed as references. The FSAR description includes the text, tables, figures and drawings.
- When performing 10 CFR 50.59 Reviews, and in cases where controlled drawings are used as FSAR figures, the most recent revision of the controlled drawing is used instead of the associated FSAR figure. This is because FSAR figures are updated only at periodic intervals and consequently may not reflect the latest controlled drawings. For purposes of 50.59 Screens and 50.59 Evaluations, the latest issue of the applicable controlled drawing(s) represents the "updated FSAR" instead of the associated figure(s). A list of FSAR figures that are also plant controlled drawings are listed in an Attachment to FDP-ZZ-00110, Electronic Licensing Document Revision.

7.8. **Certificate of Compliance (CoC) –** The certificate issued by the NRC that approves the design of a spent fuel storage facility in accordance with the provisions of Subpart L of 10 CFR 72. The CoC may include appendices containing technical specifications, approved contents, design features, and other information controlled as part of the CoC (NRC approval is required for changes to these appendices.)

7.9. **UMAX FSAR -** The FSAR referred to in 10CFR72.48 is the HI-STORM UMAX System Final Safety Analysis Report (i.e., cask FSAR) revision that is the licensing basis for the ISFSI as documented in the 72.212 Report, including any generic changes or site-specific deviations applicable to the ISFSI. The applicable cask FSAR revision for the ISFSI may not be the latest issued by the CoC holder.

It is important to note that Holtec maintains this document.

7.10. **UMAX CoC Safety Analysis Report (SER) -** In support of issuing a dry spent fuel storage system certificate of compliance or amendment thereto, the NRC staff performs a safety evaluation and issues a safety evaluation report (SER). The SER serves as the basis to confirm that all the descriptions, analyses, commitments, programs, etc., in the certificate of compliance's final safety analysis report are acceptable. The SER is a source of information that may be useful in determining acceptance criteria for provisions of the cask FSAR. The NRC issues an original SER for initial certification and supplemental SERs for amendments covering only the scope of the amendment.

It is important to note that Holtec maintains this document.

7.11. **Outside Agency/Organization** – An organization consisting of individuals NOT employed directly by Ameren.

7.12. **Qualified Reviewer** – A qualified person who performs an independent review of a document(s) prepared pursuant to the 50.59 or 72.48 Review process. A Qualified Reviewer of an Applicability Determination, a 50.59 or 72.48 Screen, or a 50.59 or 72.48 Evaluation must ensure the proper qualifications as delineated in Section 4.3.

7.13. **Screener** – A person who is qualified to prepare or review Applicability Determinations and 50.59 or 72.48 Screens. Section 4.3 delineates qualifications.

7.14. **Evaluator** - A person who is qualified to prepare or review 50.59 or 72.48 Evaluations. Section 4.3 delineates qualifications.

7.15. **50.59 and 72.48 Resource Manuals** – Documents developed and maintained by the Utilities Service Alliance (USA) which have been adopted for use at Callaway Plant. The 50.59 and 72.48 Resource Manuals are consistent with NEI 96-07 and provide useful guidance and information for answering the questions listed on the Applicability Determination, 50.59 or 72.48 Screen and 50.59 or 72.48 Evaluation forms, and may be updated occasionally as changes are made to the generic USA document. The current revision of the 50.59 and 72.48 Resource Manuals are available from Document Control.

7.16. **10 CFR 72.212 Evaluation Report (72.212 Report)** – The document required by 10 CFR 72.212 to be developed and maintained, which: a) describes how the licensee complies with the requirements of the cask CoC and b) describes the evaluations performed as a condition of use for the 10 CFR 72 general license for the storage of spent nuclear fuel at the ISFSI.

-END OF SECTION-

8.0 SUMMARY OF CHANGES

Page(s)	Section or Step Number	Description
throughout		Revised entire procedure to include elements of the 10CFR72.48 rule as a result of the Callaway ISFSI project.

-END OF SECTION-

Attachment 1

Storage Cask Certificate Of Compliance (10CFR72.244) Checklist

Sheet 1 of 2

The following questions should be considered in light of possible impact to regulatory requirements and controls associated with Dry Cask Storage. If the governing documents (i.e., CoC, Cask FSAR, Cask Tech Specs, 72.212 Report, etc.) are unaffected by the proposed activity, then "NO" should be marked. Any "YES" should result in either a "Part" or "All" on the AD.

Does the proposed activity:

1. ☐ Yes ☐ No Involve a change to 1) the ISFSI facility, 2) spent fuel storage cask design or 3) procedures that affect a design function, a method of performing or controlling a design function, or an evaluation that demonstrates that intended functions will be accomplished?
2. ☐ Yes ☐ No Involve a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?
3. ☐ Yes ☐ No Involve spent fuel storage cask design basis accident or an ISFSI facility external event (i.e. flood, fire, earthquake, tornado missile, etc.) described in the FSAR?
4. ☐ Yes ☐ No Involve Maintenance activities impacting the ISFSI facility or Storage casks? Maintenance activities include troubleshooting, calibration, refurbishment, maintenance-related testing, identical replacements, housekeeping and similar activities that do not permanently alter the design, performance requirements, operation, or control of SSCs. *NOTE: 10 CFR 50.65 [Maintenance Rule] does not apply to an ISFSI facility or spent fuel storage cask certified under 10 CFR Part 72.*
5. ☐ Yes ☐ No Involve a Temporary Change to the ISFSI facility, spent fuel storage cask design or procedures for the purpose of 1) compensatory measures to address degraded or non-conforming conditions, or 2) implementation of temporary alterations that directly relate to and are necessary to support Maintenance? Temporary alterations include jumpering terminals, lifting leads, installing temporary lead shielding on pipes and equipment, removal of barriers, and use of temporary blocks, bypasses, scaffolding and supports.
6. ☐ Yes ☐ No Involve a change to the physical characteristics of the ISFSI storage pad which could impact a design bases function or evaluation (e.g. seismic, liquefaction, settlement, sliding, tip-over, flooding, etc.)?
7. ☐ Yes ☐ No Involve new buildings, structures, or fire/ explosive sources located in the proximity of the ISFSI storage pad or the Haul Path?
8. ☐ Yes ☐ No Involve a change to ITS-A, ITS-B, or ITS-C equipment utilized for spent fuel cask loading or storage operations?
9. ☐ Yes ☐ No Involve a change to the design or operation of Special Lifting Devices utilized with the Fuel Handling Building Crane in spent fuel cask handling operations?
10. ☐ Yes ☐ No Involve a change to off-site radiological release projections from ISFSI or spent fuel cask sources?
11. ☐ Yes ☐ No Involve a change to design bases documents associated with spent fuel cask loading or storage (CoC, Cask FSAR, Cask Tech Specs, 72.212 Report, etc.)

Attachment 1

Storage Cask Certificate Of Compliance (10CFR72.244) Checklist

Sheet 2 of 2

<u>10 CFR 72.48 Notes</u>

10 CFR 72.48 establishes the conditions under which the ISFSI licensee (Ameren) or the spent fuel storage cask certificate holder (Holtec) may make changes 1) in the ISFSI facility, 2) in the spent fuel storage cask design, or 3) in procedures described in the UFSAR; and conduct tests or experiments not described in the UFSAR without prior NRC approval. Proposed changes, tests and experiments (hereafter referred to collectively as activities) that satisfy the definitions and one or more of the criteria in the rule must be reviewed and approved by the NRC before implementation. Thus, 10 CFR 72.48 provides a threshold for regulatory review-not the final determination of safety-for proposed activities.

NEI 96-07 Appendix B provides guidelines for 10CFR 72.48 Evaluations.

In addition to controlling changes to the ISFSI facility, spent fuel storage cask design, and procedures described in the UFSAR under 10 CFR 72.48 as required by the rule, general licensees must also control changes to their 10 CFR 72.212 evaluations using the 10CFR 72.48 process in accordance with 10 CFR 72.212(b)(7).
10 CFR 72.48 controls changes to both 10 CFR 72.3 design bases and supporting design information contained in the UFSAR.

Activities described below are NOT associated with 10 CFR 72.48:

1. Changes to Administrative procedures governing the conduct of operations. NOTE: Changes to how design functions are performed or controlled do require a 72.48 review.

2. Spent Fuel selection and MPC loading

3. Cask Loading and Storage procedures (parameters are based on FSAR values which are Part 72)

4. Changes to plant systems that support dry cask loading and storage activities (Helium, Service Air, Demineralized Water, Borated Water, Radiation Monitoring, etc.).

5. Changes to the Fuel Handling Building structure or building systems that support dry cask loading and storage activities (ventilation, electrical power, SFP water level and chemistry, SFP gates, SFP cooling water sources, etc.).

6. Changes to the design or operation of the Fuel Handling Building Crane as it relates to spent fuel cask handling operations (e.g. capacity, travel limits, limit switches, heavy load paths, single-failure-proof designation, etc.).

7. Changes to design of the Haul Path or operation of cask transport equipment on the Haul Path.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Radiation control	**Group #**	N/A		
	K/A #	2.3.4		
	Importance Rating	3.7		
Knowledge of radiation exposure limits under normal or emergency conditions.				

Question #97

Given the following plant conditions:

- A General Emergency exists.
- The TSC's Emergency Response Organization is manned and functional.
- An operator that was sent to close a valve in the Auxiliary Building was severely injured.
- His injuries appear to be life threatening.
- A Rescue Team is being organized to attempt to remove the operator from the area.
- Radiation levels are approximately 60 Rem/hr in the area.

What is MAXIMUM dose that is allowed to perform this operation and whose permission is required?

An individual may receive up to …….

A. 10 Rem DDE with authorization from the Duty Manager.

B. 100 Rem DDE with authorization from the Duty Manager.

C. 10 Rem DDE with authorization from the Emergency Coordinator.

D. 100 Rem DDE with authorization from the Emergency Coordinator.

Answer: D

Explanation:
Attachment 1 of HDP-ZZ-01450 shows the dose limits allowed. Category 1 for life saving is 100 DDE while Category 2: Protection of large populations, personnel safety and accident mitigation is 10 DDE making 10 DDE plausible.

Step 3.3.1 of HDP-ZZ-01450 states whom can approve dose exposure in excess of the limits of 10 CFR and those people are:

- *Senior Vice President Generation and Chief Nuclear Officer*
- *Vice President Nuclear Operations*
- *Emergency Coordinator*

- *Recovery Manager*

Duty Manager is a plausible distractor as the duty manager is contacted / notified by the Shift Manager per ODP-ZZ-00001 ADD 13 for "Events involving an unplanned radioactive release or unplanned personnel exposure" but this in no way means that this person approves the exposure.

A. Incorrect. – both parts are wrong
B. Incorrect. – wrong person for approval
C. Incorrect – wrong dose limit
D. Correct.

Technical Reference(s):
1. HDP-ZZ-01450, Authorization to Exceed Federal Occupational Dose, Rev 11

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #75 – ALARA/RB Entry Objective I: HDP-ZZ-01450, Authorization To Exceed Federal Occupational Dose

1. IDENTIFY who can authorize dose exposure in excess of 10CFR20.1201 dose limits.
2. DISCUSS the limits for plant emergencies and the selection criteria associated with these limits

Question Source: Bank # _X__L16757___
Modified Bank # ______
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(4)

Comments:

SRO level question due to radiation hazards that may arise during normal and abnormal suituations, including maintenance activities and various contamination conditions; 10 CFR 55.43(b)(4)

3.3. Exposure Authorization/Documentation

3.3.1. One of the following can authorize dose exposure in excess of the limits of 10CFR20.1201, Occupational Dose Limits for Adults.

- Senior Vice President Generation and Chief Nuclear Officer
- Vice President Nuclear Operations
- Emergency Coordinator
- Recovery Manager

3.3.2. Authorization to exceed the occupational dose limits of 10CFR20.1201, Occupational Dose Limits for Adults is documented utilizing CA0276, Authorization to Exceed Federal Occupational Radiation Dose Limits.

3.3.3. Verbal authorization may be granted and the documentation completed at a later time, if necessary

3.4. Follow Up Actions For Doses In Excess Of 10CFR20.1201 Limits

3.4.1. Immediately notify one of the following of any dose potentially in excess of the limits of 10CFR20.1201, Occupational Dose Limits for Adults.

- Senior Vice President Generation and Chief Nuclear Officer
- Vice President Nuclear Operations
- Emergency Coordinator
- Recovery Manager

NOTE

The emergency phone directory contains a listing for the Radiation Emergency Assistance Center (REACTS) which provides medical assistance in the event of a potential radiation injury.

3.4.2. All individuals who have received a dose in excess of twice any limit of 10CFR20.1201, Occupational Dose Limits for Adults must be examined by a physician experienced in the treatment of radiation injuries.

3.4.3. All doses in excess of the routine occupational dose limits received during emergency situations are determined and subtracted from the dose limits for Planned Special Exposures Per 10CFR20.1201(b), Occupational Dose Limits for Adults.

3.4.4. All doses in excess of any limit of 10CFR20.1201, Occupational Dose Limits for Adults are assessed with respect to the reporting requirements of 10CFR20.2202, Notification of Incidents and 10CFR20.2203, Reports of Exposures, Radiation Levels, and Concentration of Radioactive Material Exceeding Constraints or Limits.

3.4.5. For reporting guidance, Refer To APA-ZZ-00520, Reporting Requirements and Responsibilities.

3.5. Estimating TEDE And CDE-Thyroid Dose For Emergency Team Personnel

3.5.1. Accident Type is determined by contacting either the Dose Assessment Tech in the Control Room or the Dose Assessment Coordinator in the EOF.

3.5.2. Calculate the TEDE dose using the following formula:

Dose Rate (in R/Hr) x 3 x Exposure Time (in Hrs) = TEDE (in REM)

3.5.3. Calculate CDE-Thyroid dose using the following formula:

Dose Rate (in R/Hr) x 40 x Exposure Time (in Hrs) = CDE-Thyroid (in REM)

4.0 REFERENCES

4.1. Implementing

4.1.1. APA-ZZ-00520, Reporting Requirements and Responsibilities

4.1.2. APA-ZZ-01000 Attachment 1, Callaway Plant Administrative Dose Guidelines

4.1.3. CA0276, Authorization to Exceed Federal Occupational Radiation Dose Limits

4.1.4. 10CFR20.1001(b), Purpose

4.1.5. 10CFR20.1201, Occupational Dose Limits for Adults

4.1.6. 10CFR20.1201(b), Occupational Dose Limits for Adults

4.1.7. 10CFR20.1206, Planned Special Exposures

4.1.8. 10CFR20.1208, Dose to an Embryo/Fetus

4.1.9. 10CFR20.2202, Notification of Incidents

4.1.10. 10CFR20.2203, Reports of Exposures, Radiation Levels, and Concentration of Radioactive Material Exceeding Constraints or Limits

4.1.11. 10CFR50.47(b)(11), Emergency Plans

4.1.12. 10CFR50.54(x), Conditions of Licenses

4.2. Developmental

4.2.1. Callaway Plant Radiological Emergency Response Plan Section 6.8.1

4.2.2. Regulatory Guide 8.13, Revision 3, Instruction Concerning Prenatal Radiation Exposure, USNRC, June, 1999.

4.2.3. Regulatory Guide 8.35, Planned Special Exposures Section C, "Regulatory Position", June, 1993.

4.2.4. NUREG/CR-6204, Questions and Answers Based on the Revised 10 CFR Part 20, May, 1994, questions 97 and 407.

4.2.5. DOE/NV/11718--80-Rev2, FRMAC Operations Manual, December 2005

4.2.6. HPCI 01-05, Estimation Of TEDE And CDE-Thyroid Dose For Emergency Team Personnel.

5.0 RECORDS

CA0276, Authorization to Exceed Federal Occupational Radiation Dose Limits (File H230.0064)

6.0 SUMMARY OF CHANGES

Page(s)	Section or Step Number	Description
6	3.5.3	Removed reference to attachment 4, since attachment 4 was removed in revision 9.

Attachment 1

Emergency Occupational Dose Limits

Sheet 1 of 1

Category 1: Life Saving	
Dose Limit (1)	**Personnel Selection Criteria**
100 rem DDE	1. Women of childbearing capacity are strongly encouraged to not volunteer. (See Note 2, below.) 2. Preferably has a lifetime dose (in rem) of less than his/her age and has not previously received a dose in excess of 10CFR20 limits. 3. Shall not be a declared pregnant woman. 4. Individual must be informed and fully aware of the risks involved.

Category 2: Protection of large populations, personnel safety, and accident mitigation	
Dose Limit	**Personnel Selection Criteria**
10 rem DDE 30 rem Lens of the Eye 100 rem SDE 100 rem TODE	1. Voluntary for women of child bearing capacity. (See Note 2, below.) 2. Preferably has a lifetime dose (in rem) of less than his/her age and has not previously received a dose in excess of 10CFR20 limits. 3. Shall not be a declared pregnant woman.

NOTES:

(1) These are recommended dose limits. Doses in excess of these limits should be avoided to the maximum extent possible. However, it is not possible to prejudge the amount of risk that one should be allowed to take to save the lives of others. Personnel electing to perform lifesaving actions shall do so on a voluntary basis and with full awareness 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. (See Attachment 2.)

(2) Women of childbearing capacity are strongly encouraged to not volunteer for assignments which may result in a dose of this magnitude. Refer to Attachment 3 for additional information.

Attachment 2

Health Effects From Acute Radiation Doses (1)

Sheet 1 of 1

Health Effects from a Single Acute Whole Body Absorbed Dose (TEDE)		
Dose (rem)	**% Prodromal Effects (2)**	**% Early Fatalities (3)**
50	2	< 5
100	15	< 5
140	50	5
200	85	15
250	98	35
300		50
400		85
460		95

Range of Acute Radiation Doses (TEDE) and Associated Radiation Induced Syndromes Causing Death		
Dose Range (rem TEDE)	**Principal Effect Contributing to Death**	**Time of Death (days post exposure)**
300 - 500	Hemopoietic Syndrome (damage to bone marrow)	30 - 60
500 - 1500	Gastro Intestinal Syndrome (damage to lungs and/or lining of intestinal tract	10 - 20
> 1500	Central Nervous System Syndrome (damage to brain)	1 - 5

Acute Radiation Doses (TODE) Causing Injury to Organs			
Organ	**Risk of Injury Within 5 Years**		**Type of Injury**
	5%	**50%**	
Bone Marrow	250 rem	450 rem	aplasia and pancytopenia
Lens of the Eye	500	1200	cataracts
Ovary	200 - 300	625 - 1200	permanent sterilization
Testes	500 - 1500	2000	permanent sterilization
Skin	5500	7000	ulcers, fibrosis
Liver	2500	4000	acute and chronic hepatitis
Lung	1500	2500	acute and chronic pneumonia.
Stomach, Intestines	4500	5500	ulcer, perforation, hemorrhage
Kidney	2000	2500	acute and chronic nephrosclerosis
Thyroid	3000		hypothyroidism

Approximate Cancer Risk to Average Individuals from a single acute dose of 25 rem TEDE		
Age at Exposure (years.)	**Approximate Risk of Premature Death (deaths per 1000 persons exposed)**	**Average Years of Life Lost if Premature Death Occurs (years)**
20 to 30	9.1	24
30 to 40	7.2	19
40 to 50	5.3	15
50 to 60	3.5	11

NOTES:

(1) These values assume a healthy young adult. Others are assumed to have an increased risk of acute mortality at high dose.

(2) Prodromal effects are early effects such as skin erythema, non-malignant skin damage, loss of appetite, nausea, fatigue, and diarrhea. Other pathophysiological effects such as hemotologic deficiencies, temporary infertility, and chromosome changes may also occur.

(3) Without medical treatment

Attachment 3

Effects Of Radiation Dose On The Developing Human Embyro / Fetus

Sheet 1 of 1

All women of child bearing capacity must read and ensure that the following is well understood before volunteering for an assignment which could result in a dose in excess of the occupational dose limits of 10CFR20.1201. The International Commission on Radiation Protection (ICRP) and the National Council on Radiation Protection (NCRP) and other national and international scientific advisory bodies on radiation protection recommend that women of child bearing capacity not exceed the occupational dose limits of 10CFR20.1201 because of the risk to the embryo/ fetus and the period of time that can elapse between conception and the point at which the pregnancy becomes known to the woman. Doses in excess of the occupational dose limits of 10CFR20.1201 have been shown to significantly increase the risk to the fetus of mental retardation and cancer induction.

It is the fundamental responsibility of the woman to decide if she is of childbearing capacity, and whether she will volunteer for an assignment which may result in a dose in excess of the occupational dose limits of 10CFR20.1201. However, in light of the significant risk to the embryo/ fetus, AmerenUE strongly encourages all women of childbearing capacity to not volunteer for such assignments. No woman shall experience a loss in wages, benefits, employment, or promotional opportunity as a result of electing to not volunteer for an assignment which may result in a dose in excess of the occupational dose limits of a 10CFR20.1201.

The developing human brain has been shown to be especially sensitive to radiation. Mental retardation has been observed in the survivors of the atomic bombings in Japan exposed *in utero* during sensitive periods of development. Additionally, some other groups exposed to radiation *in utero* have shown lower than average intelligence scores and poor performance in school.

No developmental effects caused by radiation have been observed at doses below the 5 rem occupational dose limit of 10CFR20.1201. Scientists are uncertain whether there are developmental effects at doses below the 5 rem dose limit. It may be that the effects are present but are too mild to measure because of the normal variability from one person the next and because the tools to measure the effects are not sensitive enough. Or, it may be that there is a threshold dose below which there are no developmental effects.

In addition to the developmental effects, scientific advisory groups assume that radiation exposure before birth may be 2 or 3 times more likely to induce cancer over a person’s lifetime than the same amount of radiation received as an adult.

In view of the possibility of developmental effects, even if very mild, at doses below 5 rem, and the potential risk in the induction of cancer, scientific advisory groups consider it prudent to limit the dose to the embryo/ fetus to 0.5 rem. At doses greater than 5 rem, such as might be received during an accident or during emergency response activities, the possibility of developmental effects and cancer induction increases significantly. Therefore, these same advisory groups recommend that women who are capable of conception limit their dose to the occupational dose limit of 5 rem in order to prevent excessive dose to the embryo/ fetus before the pregnancy becomes known to the woman.

Reference: Regulatory Guide 8.13, Revision 3.

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	3		
Radiation Control	**Group #**	N/A		
	K/A #	G 2.3.13		
	Importance Rating	3.8		
Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc				

Question # 98

Given the following timeline:

Date	Time	Activity
1/31	0000	Plant Shutdown to Mode 3 commenced.
1/31	0630	Mode 3 Entry.
1/31	1330	Mode 4 Entry.
1/31	2230	Mode 5 Entry.
2/01	2230	Mode 6 Entry.

What is the EARLIEST time that irradiated fuel movements within the vessel may commence?

A. 2/2 at 0630.

B. 2/3 at 0630.

C. 2/3 at 2230.

D. 2/4 at 2230.

Answer: B

Explanation:
OSP-SF-00003, PRE-CORE ALTERATION VERIFICATIONS*, Attachment 5, Actions Required Prior To Movement Of Irradiated Fuel Assemblies Within The Reactor Vessel Attachment 4 must be current. Attachment 4, Actions Required Prior To Removing The Upper Internals, states that “Reactor has been Subcritical for at least 72 hours.”*

Based on the conditions in the stem the applicant cannot assume the reactor is subcritical until 1/31 at 0630.

Per the FSAR " The first step in fuel handling is the safe shutdown of the reactor. After a radiation survey of the containment, the disassembly of the reactor vessel is started. After disassembly is complete, the first fuel handling is started. It is estimated that the earliest time to first fuel transfer *after shutdown is 72 hours."*

A. Incorrect, Plausible if the candidate incorrectly applies 48 hours to the time the reactor was subcritical.
B. Correct per explanation above
C. Incorrect, Plausible if the candidate incorrectly applies 48 hours to the time the reactor entered Mode 6. This is also 72 hours from the Mode 5 entry time.
D. Incorrect, Plausible if the candidate incorrectly applies 72 hours to the time the reactor entered Mode 6.

Technical Reference(s):
1. OSP-SF-00003, PRE-CORE ALTERATION VERIFICATIONS, Rev 28
2. FSAR 15.7.4.2 Sequence of Events and Systems Operations

References to be provided to applicants during examination: none

Learning Objective: T61.003B, Off-Normal Operations, LP #60, Objective C: STATE the major steps for OTG-ZZ-00007:
1. Cooldown from cold shutdown to refueling conditions.
2. Preparation for fuel movement.

Question Source: Bank # _X_L16752____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A _______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(7)

Comments:
Matches the KA by testing the applicant knowledge radiological safety procedures pertaining to fuel handling found in FSAR 15.7.4.2 and OSP-SF-00003.

SRO Only met by:
Fuel handling facilities and procedures. [10 CFR 55.43(b)(7)]
Some examples of SRO exam items for this topic include:
Prerequisites for vessel disassembly and reassembly.
Assessment of surveillance requirements for the refueling mode.

Attachment 4

Actions Required Prior To Removing The Upper Internals

Sheet 1 of 1

Person Performing Attachment   Initials

____________________   ________   Date/Time Started ________________

____________________   ________   Date/Time Completed ________________

Step		Initial
6.5.1	Attachment 3 is CURRENT.	
6.5.2	Reactor has been Subcritical for at least 72 hours.	
6.5.3.a	CTMT Equipment Hatch Missile Shield is closed.	
	OR	OR
6.5.3.b	Administrative requirements of APA-ZZ-00750 are in place for the equipment hatch missile shield.	

Attachment 5

Actions Required Prior To Movement Of Irradiated Fuel Assemblies Within The Reactor Vessel

Sheet 1 of 1

Person Performing Attachment	Initials		
______________________	________	Date/Time Started	________________
______________________	________	Date/Time Completed	________________

Step		Initial
6.6.1	Attachment 4 is CURRENT.	
6.6.2	Refueling Machine Overload Cutoff Test is complete.	

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	3		
Emergency Procedures / Plan	**Group #**	N/A		
	K/A #	G2.4.28		
	Importance Rating	4.1		
Knowledge of procedures relating to a security event (non-safeguards information).				

Question # 99

Reactor Power is 100%.

The NRC just called the Control Room and supplied the following information:

- A Boeing 737 was hijacked and is on path toward the site.
- The estimated time of arrival is 25 minutes.
- A valid authentication code was provided.

(1) What Attachment of OTO-SK-00002, Plant security Event – Aircraft Threat, should the CRS enter?

And

(2) Per OTO-SK-00002 and the above conditions, what actions are REQUIRED at this point in time?

A. (1) Attachment A, Airborne Threat - Imminent
(2) Actuate FBIS, and Check CRVIS is Actuated ONLY

B. (1) Attachment A, Airborne Threat - Imminent
(2) Manually trip the Reactor, Actuate FBIS, and Check CRVIS is Actuated.

C. (1) Attachment B, Airborne Threat - Probable
(2) Actuate FBIS, and Check CRVIS is Actuated ONLY

D. (1) Attachment B, Airborne Threat - Probable
(2) Manually trip the Reactor, Actuate FBIS, and Check CRVIS is Actuated.

Answer: C

Explanation:
Per OTO-SK-00002, Plant security Event – Aircraft Threat and the conditions time to the site being 25 minutes, the CRS will direct the actions of Attachment B, Airborne Threat – Probable.

Attachment A would be applicable if the time to the site was less than 5 minutes but plausible if the candidate does not apply 25 minutes and 'Imminent' correctly.

Per the conditions given in the stem, 25 minute arrival time, a Reactor TRIP is NOT required. Requiring a reactor trip is plausible as it is required in Attachment A step #A11. Furthermore, the candidate may believe / falsely remember that it is required in both Attachments A and B or only required in Attachment B if they attachment priority is confused because the actions in Attachment B mirror that of Attachment A. The candidate may believe a trip is required due to the sense of urgency; the arrival time is only 25 minutes from now. Requiring a reactor trip when the arrival time is 25 minutes is plausible for these reasons.

Both Attachment A and B steps #13 and 14 direct manually actuating FBIS and verifying CRVIS actuation.

A. Incorrect, the procedure attachment is wrong
B. Incorrect, both are wrong
C. Correct, see above
D. Incorrect, a reactor trip is not required

Technical Reference(s):
1. OTO-SK-00002, Plant security Event – Aircraft Threat, Rev 20

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP#16, OTO-SK-00001 Plant Security Event – Hostile Intrusion, OTO-SK-00002, Plant Security Event – Aircraft Threat, OTO-SK-00004, Plant Security Event – Electrical Threat. Obj J Given a set of plant conditions or parameters indicating a Plant Security Event – Aircraft Threat, IDENTIFY the correct procedure(s) to be utilized and OUTLINE the high level actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam _____NA_______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

Revised part 2 of question per NRC Comments about number of EAL questions on exam.

SRO ONLY Justification:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **NO**
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures **NO**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures **NO**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**CHECK That Size Of Aircraft Is -** • Large Aircraft (long-distance) OR • Small aircraft that presents a greater threat than its size would indicate	Go To Attachment C, Airborne Threat - Informational.
# 5.	**CHECK Estimated Time To Site - 5 MINUTES OR LESS**	IF the estimated time to the site is less than 30 minutes, THEN Go To Attachment B, Airborne Threat - Probable. IF the estimated time to the site is 30 minutes or greater, THEN Go To Attachment C, Airborne Threat - Informational.
6.	**Go To Attachment A, Airborne Threat - Imminent**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 1 of 7)
Aircraft Threat - Imminent

NOTE

IMMINENT (All are normally required)
1. Notification and/or verification by NRC or NORAD that the site is in the flight path of a Track of Interest.
2. Estimated time to site is 5 minutes or less.
3. Altitude or heading changes align the aircraft with the site.

# A1.	**CHECK Aircraft Threat - IMMINENT**	Return to Procedure Steps, Step 2.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 2 of 7)
Aircraft Threat - Imminent

A2. CHECK Notifications Of Aircraft Threat Received From The NRC

- Authentication Code Provided By The NRC Matches The Current Alphanumeric Authentication Code

  AND

- Maintain continuous communication with the NRC Operations Center

  AND

- If lost, reestablish continuous communication with the NRC Operations Center as soon as practical after an on-site aircraft impact or within 5 minutes of previously anticipated impact time if no impact occurs

PERFORM the following:

a. While Continuing With The Procedure, VALIDATE The Authenticity Of The Incoming Call AND NOTIFY NRC Operations Center Using The ENS Line:
   - Primary:
     - 9-1-301-816-5100
   - Backup:
     - 9-1-301-951-0550
     - 9-1-301-415-0550
     - 9-1-301-415-0553

b. PROVIDE the following information:
   - Site Name - Callaway
   - Current Alphanumeric Authentication Code
   - Emergency Classification - IF Determined
   - Nature of the Threat - Airborne Threat Imminent

c. Maintain continuous communication with the NRC Operations Center.

d. If lost, reestablish continuous communication with the NRC Operations Center as soon as practical after an on-site aircraft impact or within 5 minutes of previously anticipated impact time if no impact occurs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 3 of 7)
Aircraft Threat - Imminent

A3. DIRECT Each Watchstander To Perform Their Assigned Steps:

- SM - Steps A4 through A9
- BOP - Attachment D, BOP Actions
- CRS/RO - Steps A10 through A15

NOTE

Shift Manager performs Steps A4 through A9.

A4. DECLARE An Alert

NOTE

A normal Sentry notification will still be required after the event declaration is made. Selecting the "Imminent Aircraft Threat" icon will bring up a notification message with information for ONLY the Counties and State.

A5. CONDUCT County And State Notifications Using SENTRY:

a. ENSURE SENTRY computer ON

b. SELECT IMMINENT AIRCRAFT THREAT Icon on the desktop

c. SEND message

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 4 of 7)
Aircraft Threat - Imminent

A6. NOTIFY Plant Security Of The Following:

- Aircraft Threat is Imminent
- If the event is at night NOTIFY Security a loss of Outside Lighting will occur
- NOTIFY Security to take actions for Code Black Tornado and adjust as directed by SSS

A7. ACTIVATE Callaway Plant Emergency Response Organization(ERO) Via The Internet(Everbridge) Using Instructions Kept In The Shift Manager's Desk

- Send Message 12

A8. Perform Sentry Notifications per EIP-ZZ-00201, Notifications

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 5 of 7)
Aircraft Threat - Imminent

NOTE

The requirements for the Line of Sight Two-Person Rule must be implemented as soon as practicable and when safe for personnel involved. It is recognized that additional personnel may need to be called in to implement this activity. Routine essential duties such as Operations Technician Rounds and Security Patrols may continue until personnel are available on-site to implement the Line of Sight Two-Person Rule.

A9. CONSIDER Implementing Line of Sight Two-Person Rule:

- Any exceptions to the Line of Sight Two-Person Rule are approved by the Shift Manager:
  - Those exceptions are only made to a allow for essential plant personnel duties in the interim period until necessary personnel have arrived on site to support the Two-Person Rule

    OR

  - When it is NOT practicable to implement due to personal safety concerns
- Brief available personnel if using the Line of Sight Two-Person Rule per Attachment F, Security Briefing For Credible Threat

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 6 of 7)
Aircraft Threat - Imminent

NOTES

- CRS/RO are responsible for the remaining steps of this attachment.
- Any Fuel stored in an Independent Spent Fuel Storage Installation(ISFSI) Multi-Purpose Canister(MPC) should remain in the MPC.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A10.	**CHECK Fuel Handling - SECURED**	PERFORM the following: a. PLACE fuel assembly in transit in the nearest safe location: • Reactor Vessel <u>OR</u> • Spent Fuel Pool b. DISCONTINUE fuel handling operations.
A11.	**CHECK Reactor - SHUTDOWN**	PERFORM the following: a. Manually TRIP the Reactor. b. PERFORM E-0, Reactor Trip Or Safety Injection, while continuing with this procedure. c. MAINTAIN Steam Generator Level In The Upper End Of The Operating Band.
A12.	**NOTIFY NRC That On-site Personnel And Off-site Response Organizations Have Been Notified And The Reactor Is Tripped.**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT A
(Page 7 of 7)
Aircraft Threat - Imminent

A13.	**Manually ACTUATE Fuel Building Isolation (FBIS):** • SA HS-10 • SA HS-14	
A14.	**CHECK CRVIS ACTUATED:**	Manually ACTUATE CRVIS: • SA HS-9 • SA HS-13
A15.	**Go To Attachment B, Aircraft Threat - Probable, Step B15 And PERFORM Additional Items, At The Discretion Of The SM/CRS**	

-END-

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 1 of 10)
Aircraft Threat - Probable

NOTE

PROBABLE (All are normally required)
1. Notification and/or verification by NRC or NORAD that the site is in the flight path of a Track of Interest.
2. Estimated time to site is less than 30 minutes.
3. Large aircraft or a Small aircraft that presents a greater threat than its size would indicate.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
# B1.	**CHECK Airborne Threat - PROBABLE**	IF the estimated time to the site is 5 minutes or less, THEN Go To Attachment A, Aircraft Threat - Imminent. IF the estimated time to the site is 30 minutes or greater, THEN Go To Attachment C, Aircraft Threat - Informational. If the threat is a small Aircraft, THEN Go To Attachment C, Aircraft Threat - Informational

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 2 of 10)
Aircraft Threat - Probable

NOTE

The goal of the Accelerated Notification to the NRC is to allow the NRC to warn other licensees and initiate Federal response in accordance with the National Response Plan.

B2. CHECK Notifications Of Aircraft Threat Received From The NRC

- Authentication Code Provided By The NRC Matches The Current Alphanumeric Authentication Code

  AND

- Maintain continuous communication with the NRC Operations Center

  AND

- If lost, reestablish continuous communication with the NRC Operations Centers as soon as practical after an on-site aircraft impact or within 5 minutes of previously anticipated impact time if no impact occurs

RESPONSE NOT OBTAINED:

PERFORM the following:

a. While Continuing With The Procedure, VALIDATE The Authenticity Of The Incoming Call AND NOTIFY NRC Operations Center Using The ENS Line:

- Primary:
  - 9-1-301-816-5100
- Backup:
  - 9-1-301-951-0550
  - 9-1-301-415-0550
  - 9-1-301-415-0553

b. PROVIDE the following information:

- Site Name - Callaway
- Current Alphanumeric Authentication Code
- Emergency Classification - IF Determined
- Nature of the Threat - Airborne Threat Probable

c. Maintain continuous communication with the NRC Operations Center.

d. If lost, reestablish continuous communication with the NRC Operations Center as soon as practical after an on-site aircraft impact or within 5 minutes of previously anticipated impact time if no impact occurs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 3 of 10)
Aircraft Threat - Probable

B3. DIRECT Each Watchstander To Perform Their Assigned Steps:

- SM - Steps B4 through B10
- BOP - Attachment D, BOP Actions
- CRS/RO - Steps B11 through B27

NOTE

Shift Manager performs Steps B4 through B10.

B4. DECLARE An Alert

B5. CHECK Aircraft Arrival Onsite - LESS THAN 15 MINUTES

Go To Step B7

NOTE

A normal Sentry Notification will still be required after the event declaration is made. Selecting the "Imminent Aircraft Threat" icon will bring up a notification message with information for ONLY the Counties and State.

B6. CONDUCT County And State Notifications Using SENTRY:

a. ENSURE SENTRY computer ON

b. SELECT IMMINENT AIRCRAFT THREAT Icon on the desktop

c. SEND message by pressing the SEND button

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 4 of 10)
Aircraft Threat - Probable

B7. NOTIFY Plant Security Of The Following:

- Aircraft Threat is Probable
- Security Protected Area Vehicle Gate may be required to be opened if needed for the event
- If the event is at night NOTIFY Security a loss of Outside Lighting will occur
- NOTIFY Security to take actions for Code Black Tornado and adjust as directed by SSS

B8. ACTIVATE Callaway Plant Emergency Response Organization (ERO) Via The Internet(Everbridge) Using Instructions Kept In The Shift Manager's Desk

- Send Message 12

B9. PERFORM Sentry Notification Per EIP-ZZ-00201, Notifications

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 5 of 10)
Aircraft Threat - Probable

NOTE

The requirements for the Line of Sight Two-Person Rule must be implemented as soon as practicable and when safe for personnel involved. It is recognized that additional personnel may need to be called in to implement this activity. Routine essential duties such as Operations Technician Rounds and Security Patrols may continue until personnel are available on-site to implement the Line of Sight Two-Person Rule.

B10. CONSIDER Implementing Line of Sight Two-Person Rule:

- Any exceptions to the Line of Sight Two-Person Rule are approved by the Shift Manager:
  - Those exceptions are only made to a allow for essential plant personnel duties in the interim period until necessary personnel have arrived on site to support the Two-Person Rule

    OR

  - When it is NOT practicable to implement due to personal safety concerns
- Brief available personnel if using the Line of Sight Two-Person Rule per Attachment F, Security Briefing For Credible Threat

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 6 of 10)
Aircraft Threat - Probable

NOTES

- CRS/RO are responsible for the remaining steps of this attachment.
- Any Fuel stored in an Independent Spent Fuel Storage Installation(ISFSI) Multi-Purpose Canister(MPC) should remain in the MPC.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B11.	**CHECK Fuel Handling - SECURED**	PERFORM the following: a. PLACE fuel assembly in transit in the nearest safe location: • Reactor Vessel OR • Spent Fuel Pool b. DISCONTINUE fuel handling operations.
B12.	**CHECK Fuel Transfer Tube Isolation Valve - CLOSED** • EC-V995	PERFORM the following: a. ENSURE Fuel Transfer Cart is in the Fuel Building. b. DIRECT Operations Technician to close Fuel Transfer Tube Isolation Valve: • EC-V995
B13.	**Manually ACTUATE Fuel Building Isolation (FBIS):** • SA HS-10 • SA HS-14	
B14.	**CHECK CRVIS ACTUATED**	Manually ACTUATE CRVIS • SA HS-9 • SA HS-13

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 7 of 10)
Aircraft Threat - Probable

B15.	**CHECK Event Occurring - DURING DAYLIGHT HOURS**	SECURE Exterior Plant Lighting:
		• OPEN 13.8 KV Bus PA02 To Site Breaker PA0209:
		  • PA HIS-11
		• DIRECT Operations Technician to OPEN the following breakers from the Switchyard:
		  • Safeguard Transformer 13.8 KV PCB 52-2 using MD52CS2, Breaker MD522 Control Switch.
		  • Safeguard Transformer 13.8 KV PCB 52-4 using MD52CS4, Breaker MD524 Control Switch.
		  • PPPG17303 Supply To PPPG175, AC Power Panel (SWYD House North Wall Panel)
		• DIRECT Operations Technician to OPEN the following breakers from the Site Switchgear Building:
		  • LPPG118 Main Panel Breaker
		  • LPPG119 Main Panel Breaker
		  • PPPG12103 Breaker, Cool TWR Obstruction LTS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 8 of 10)
Aircraft Threat - Probable

NOTE

If a Code RED or BLACK is in progress personnel movement must be coordinated with the Security Shift Supervisor.

B16. DISPATCH An Operator To NB02 Switchgear To Standby For Direction On Starting The Diesel Generator(s)If Needed

B17. SECURE Any Surveillances Or Maintenance Activities In Progress

B18. RETURN Equipment To An Operable Status

B19. RESTORE Inoperable ECCS Equipment To Operable Status:

- RHR pumps
- Charging pumps
- SI pumps

B20. CHECK Containment Purge - SECURED

Response not obtained: Manually ACTUATE CPIS:

- SA HS-11
- SA HS-15

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 9 of 10)
Aircraft Threat - Probable

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B21.	**CHECK Containment Integrity - ESTABLISHED**	INITIATE actions for Containment closure: a. ENSURE Containment Equipment Hatch is closed. b. ENSURE at least one door for Containment Emergency Personnel Airlock is closed. c. ENSURE at least one door for the Containment Personnel Airlock is closed. d. CLOSE any open Containment Penetration.
B22.	**ENSURE Makeup Water Sources Are Filled To The Upper Limit Of The Operating Band:** • CST • DWST • RWST • Fire Water Tanks	
B23.	**RESTORE Decay Heat Removal Systems To Service:** • AFW • RHR • Main Condenser	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT B
(Page 10 of 10)
Aircraft Threat - Probable

B24. SECURE Lineup For Both Of The Following:

- RWST or SFP recirculation per OTN-EC-00001, Fuel Pool Cooling And Cleanup System
- SFP Cleanup and Skimmer lineup per OTN-EC-00001, Fuel Pool Cooling And Cleanup System

NOTE

The Security Power Block Supervisor will make the request for additional assistance from the Callaway EOC following a Code RED or BLACK announcement.

B25. Contact Callaway EOC For Additional Outside Agencies (e.g. Fire Department, Medical Services, Law Enforcement) Assistance

- 573-592-2485 (Primary)
- 573-592-2486 (Backup)
- 573-592-2487 (Backup)

RESPONSE NOT OBTAINED: Use Emergency Telephone Directory

B26. CHECK Security Threat - HAS BEEN TERMINATED OR AIRCRAFT HAS IMPACTED SITE.

RESPONSE NOT OBTAINED: Do NOT continue until Security Threat has been terminated OR Aircraft has impacted site.

B27. Go To Attachment E, Restoration

-END-

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	**Group #**	N/A		
	K/A #	G2.4.25		
	Importance Rating	3.7		
Knowledge of fire protection procedures.				

Question # 100

Reactor Power is 100% and it is a weekday day shift. An unplanned Fire Protection Impairment is required due to an inoperable sprinkler system.

(1) Who is required to initiate the Fire Protection Impairment Permit?

And

(2) If an hourly fire watch is required, what department will perform the required compensatory actions?

A. (1) Shift Technical Advisor
(2) Security

B. (1) Shift Technical Advisor
(2) Maintenance

C. (1) Fire Marshall
(2) Security

D. (1) Fire Marshall
(2) Maintenance

Answer: A

Explanation:
Per APA-ZZ-00701, section 4.3, Planning Unscheduled Impairment, step 4.3.1 "Control Room, INITIATE an FPIP for all work activities identified to require an impairment permit." Additionally in a Note in the responsibilities section after step 3.7.5 :

- *Shift Manager/Control Room Supervisor/Shift Technical Advisor/Shift Engineer (SM/CRS/STA/SE) are identified as "CONTROL ROOM" throughout this procedure.*
- *On backshifts and weekends the STA performs the Control Room functions.*
- *On day shift during the week the Fire Protection Impairment Permit (FPIP) Coordinator in the Work Control Center performs the Control Room functions.*

Section 3.7 details the Fire Marshals responsibilities. Per step 3.7:4 the Fire Marshal "Reviews upcoming impairment activities and compensatory actions other than hourly fire watches, and coordinates needs with the Control Room." and step 3.7.5 "Reviews Fire Protection Impairment Permits (FPIP) after planning is complete to ensure compensatory actions are correct." Additionally in step 4.3.5, after the unplanned impairment is planned, the Fire Marshall will "REVIEW impairment for correctness." Therefore the Fire Marshal is plausible as he/she is involved in the process but wrong as they do not initiate the FPIP – that's the Shift Technical Advisor responsibility.

Per step 4.4.1 of the APA-ZZ-00701, "Control Room, COORDINATE with the following, as necessary, to establish necessary compensatory actions or measures:

- *Security to ensure hourly or continuous fire watches are posted"*

Maintenance department is plausible as they are often the impairer listed in the procedure and are required to notify the control room when impairments will occur or are created. Furthermore this is an unplanned impairment (not the normal work planning process) and it is plausible that any support department may be required to support the compensatory measures. Also, Maintenance is responsible for hot work (welding / grinding/ etc.) watches.

A. Correct – See above explanation
B. Incorrect – wrong FPIP comp action performer
C. Incorrect – wrong initiator
D. Incorrect – both are wrong

Technical Reference(s):
1. APA-ZZ-00700, Fire Protection Program, Rev 21
2. APA-ZZ-00701, Control of Fire Protection Impairments, Rev 21
3. APA-ZZ-00703, fire Protection Operability Criteria and Surveillance Requirements, Rev 25

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #72, FPIP, HAZMAT and Injuries, Objective A & B:
A. In accordance with APA-ZZ-00701; Control of Fire Protection Impairments
1. DEFINE Fire Protection Impairment
2. DEFINE Fire Protection Impairment Permit
3. DESCRIBE the Shift Manager/Control Room Supervisor/Shift Technical Advisor/Shift Engineer responsibilities.

B. In accordance with APA-ZZ-00703, Fire Protection Operability Criteria and Surveillance Requirements
1. DESCRIBE the purpose and scope.
2. DISCUSS the SM/CRS responsibilities.
3. Utilizing Attachments 1 through 7:
    a. DETERMINE the action if operability requirements are not met.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(1)

Comments:

SRO Only due to Conditions and limitation in the facility license 10 CFR 55.43(b)(1) specifically "Administration of fire protection program requirements such as compensatory actions associated with inoperable sprinkler systems, fire doors, etc."

Revised question per NRC comments.

3.3.4. Adds Security resources to jobs IF Security resources are required for continuous fire watch support.

3.4. Plant Personnel:

3.4.1. Notifies the Control Room immediately IF an unplanned impairment to a fire protection system or component is discovered.

3.4.2. Complies with SDP-KC-00001, Requirements for and Duties of Compensatory Fire Watches, when posted as a continuous fire watch including completing post documentation and returning documentation to the Shift Security Supervisor (SSS) once the post has been terminated.

3.5. Restorer:

3.5.1. Restores impairments created by the work activity before signing off the work package.

3.5.2. Coordinates with the Control Room to ensure Control Room Personnel know when impairments are restored.

3.6. Shift Security Supervisor (SSS):

3.6.1. Ensures compensatory fire watches are stationed according to SDP-KC-00001, Requirements for and Duties of Compensatory Fire Watches, after receiving a request for a fire watch from the Control Room.

3.6.2. Informs the Control Room IF on-shift Security Personnel are unavailable to post a fire watch.

3.6.3. Processes fire watch activities according to SDP-KC-00001, Requirements for and Duties of Compensatory Fire Watches.

3.6.4. Performs call-out of off-duty Security Personnel to post a continuous fire watch when deemed necessary by the Control Room.

3.7. Fire Marshal:

3.7.1. Provides notification to Nuclear Electric Insurance Limited (NEIL) of impairments and elimination of impairments, when required.

3.7.2. Monitors fire protection impairments and identifies scheduled work dates for restoring impairments exceeding 35 days.

3.7.3. Notifies lead craft they are approaching 35 day completion expectation for jobs restoring impairments.

3.7.4. Reviews upcoming impairment activities and compensatory actions other than hourly fire watches, and coordinates needs with the Control Room.

3.7.5. Reviews Fire Protection Impairment Permits (FPIP) after planning is complete to ensure compensatory actions are correct.

NOTE

Shift Manager/Control Room Supervisor/Shift Technical Advisor/Shift Engineer (SM/CRS/STA/SE) are identified as "CONTROL ROOM" throughout this procedure.

On backshifts and weekends the STA performs the Control Room functions.

On day shift during the week the Fire Protection Impairment Permit (FPIP) Coordinator in the Work Control Center performs the Control Room functions.

During Outages, the FPIP Coordinator in the Outage Control Center fulfills the Control Room function.

3.8. Shift Manager/Control Room Supervisor/Shift Technical Advisor/Shift Engineer/FPIP Coordinator:

3.8.1. Maintains Impairment Permit and impairment status up-to-date when notified by plant personnel.

3.8.2. Generates and activates FPIP(s)/Impairment Points upon identification/notification of any unscheduled or unplanned impairment.

3.8.3. Ensures action to restore unscheduled impairments is initiated by the individual who reports the condition to the Control Room.

3.8.4. Coordinates with Security and the Impairer to ensure watches are stationed as required and to ensure equipment required for support of compensatory measure and required actions are installed or staged as required.

3.8.5. Ensures compensatory watches are established for impairments, when necessary.

3.8.6. Reviews Impairment Points when the associated impairment is restored and close out the permit IF all impairments have been resorted.

3.8.7. Grants authorization to implement the impairment.

3.8.8. Ensures an FPIP correctly documents impairments associated with work packages and WPA based on plant conditions at the time of the impairment before granting authorization to begin work on a work package requiring an FPIP.

3.8.9. Ensures Compensatory Measures are terminated when an impairment point is restored.

3.8.10. Notifies the Fire Marshal of any unplanned impairment as soon as practical. IF the Fire Marshal is NOT available on site, Fire Marshal is notified the next normally scheduled work day.

3.8.11. Reviews upcoming impairments and coordinating compensatory actions that could require the following:

- Extra manpower (e.g., continuous fire watch)
- Prearranged equipment (e.g., backup hoses)
- Special Compensatory measures or cautions

3.8.12. Provides on-shift Plant Personnel to post compensatory fire watches required by FPIPs when Security Personnel are unavailable.

-END OF SECTION-

4.0 PROCEDURE INSTRUCTIONS

NOTE

Fire protection equipment is designed and installed to protect plant equipment and personnel from the hazards of fire. The impairment of fire protection equipment places the plant in a greater risk to the consequences of a fire. Impaired fire protection equipment should be restored promptly.

Smoke detectors that use a small radioactive source (typically Am-241) are not opened and repaired. These devices are replaced if found to have failed.

4.1. Establishing Priority of Restoring Impaired Fire Protection Equipment

4.1.1. Schedulers, using the guidance of PDP-ZZ-00023, Work Screening and Processing, ESTABLISH the priority of work activities for fire protection equipment.

4.1.2. Fire Protection Engineer or Fire Marshal, PERFORM the following:

- Periodically REVIEW active fire protection system impairments.
- IF work activities are NOT correctly prioritized, PROVIDE feedback to scheduling.
- RESOLVE issues that prevent the prompt restoration of fire protection systems and components.
- MONITOR and IDENTIFY IF scheduled work dates for restoring fire protection impairments exceed 35 days.
- NOTIFY the lead craft they are approaching the 35 day completion expectation.

4.2. Planning Impairments

NOTE

APA-ZZ-00701 Appendix A, Electronic FPIP Processing, provides details regarding employee security levels, impairment and impairment point status codes, and automatic e-mail messaging for impairment processing.

IF the computer program is unavailable, this record can be manually generated using a CA1270, Fire Protection Impairment Permit Form.

4.2.1. Planner, PERFORM the following:

a. INITIATE an FPIP for all work activities identified during the planning process that creates an impairment.

b. INPUT the actual fire protection impaired item using the "Location" field. [Ref: 5.2.3]

NOTE

The computer will automatically supply impairment data for most of the fire protection components listed in APA-ZZ-00703, Fire Protection Operability Criteria and Surveillance Requirements, IF that location is the component of the associated work document.

IF the impaired item is known but is NOT the job location, the impaired fire protection item can still be entered into the FPIP and impairment data automatically supplied by the computer program.

c. IF the location is NOT known, PROVIDE the following information, as a minimum, so the Fire Protection Reviewer can determine what fire protection location is being impaired:

- Description of work activity
- IF applicable, work activity type and number including retest documents

d. IF any continuous fire watches will be required, ADD resource loading to the job for Security Resources.

e. CHANGE FPIP status to “Planned,” which initiates engineering review.

4.2.2. Fire Marshal or Designee, PERFORM the following:

a. REVIEW FPIP to ensure the following:

- Impaired fire protection location is identified
- Proper compensatory measures are provided
- Impact on Fire Risk Systems and Components – Fire Risk Management Actions (RMA) is identified. IF the FP system or component is credited in ODP-ZZ-00002 Appendix 3, Risk Management Actions for Fire Risk Systems Components, ADD note to the FPIP and consider the impact on the Fire Risk Management Action.
- Sufficient notes are provided for special concerns
- Retests are identified, IF applicable
- NEIL notification is made, IF required by Attachment 1, Nuclear Electric Insurance Limited (NEIL) Notifications.

b. COMPLETE applicable portion of FPIP.

c. IF FPIP requires continuous fire watch support, NOTIFY Planner to add Security resource to job.

4.3. Planning Unscheduled Impairments

NOTE

The Control Room generates impairments for unscheduled (jump-up) impairments to meet the requirements stated in APA-ZZ-00703, Fire Protection Operability Criteria and Surveillance Requirements.

4.3.1. Control Room, INITIATE an FPIP for all work activities identified to require an impairment permit.

4.3.2. Control Room, IDENTIFY impairment concerns and required compensatory actions.

4.3.3. Control Room, REVIEW FPIP for impact on Fire Risk Systems and Components – Fire Risk Management Actions. IF the FP system or component is credited in ODP-ZZ-00002 Appendix 3, Risk Management Actions for Fire Risk Systems Components, ADD note to the FPIP and consider the impact on the Fire Risk Management Action.

NOTE

The Fire Marshal will automatically receive an e-mail notification when an electronic FPIP, which is unplanned or requires NEIL notification, is made active.

4.3.4. Control Room, NOTIFY Fire Marshal of unscheduled impairment.

4.3.5. Fire Marshal, PERFORM the following:

a. REVIEW impairment for correctness.

b. NOTIFY NEIL of impairment when required by Attachment 1, Nuclear Electric Insurance Limited (NEIL) Notifications.

4.4. Implementing Impairments

NOTE

Except for unscheduled impairments, requirements for continuous fire watches are expected to be coordinated with Security before the need date.

4.4.1. Control Room, COORDINATE with the following, as necessary, to establish necessary compensatory actions or measures:

- Security to ensure hourly or continuous fire watches are posted
- Other groups, depending on required compensatory measures

4.4.2. Impairer, NOTIFY Control Room when FPIP associated work documents indicate work has started.

4.4.3. Control Room, ENSURE identified compensatory actions or measures are still applicable to the work activity.

4.4.4. Worker or Supervisor, NOTIFY Control Room before creating an impairment to ensure necessary compensatory actions are taken or compensatory measures are in place.