Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	CE/E02 EK2.1	
	Importance Rating	3.3	

K/A Statement

EK2.1 Knowledge of the interrelations between the (Reactor Trip Recovery) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO 1 Rev: 0

Given:

- Plant is at 100% power
- Steam Generator 2 Steam Flow transmitter to Feedwater Control failed LOW
- The BOP operator took manual control of FWCS Master Controller #2 and raised the output to 40%, but the reactor tripped on Low Steam Generator Level
- All other Feedwater Controllers are in AUTOMATIC

On the Reactor Trip, Reactor Trip Override (RTO) will ____(1)____. Contingency actions required by the BOP will include ____(2)____.

	(1)	(2)
A.	align FWCS #2 components to their RTO position	raising the output of FWCS Master Controller #2 to reset the RTO
B.	align FWCS #2 components to their RTO position	lowering the output of FWCS Master Controller #2 to reset the RTO
C.	not align FWCS #2 components to their RTO position	manually closing MFRV #2 and positioning SUFRV #2 to 13 – 21% open
D.	not align FWCS #2 components to their RTO position	manually closing MFRV #2, positioning SUFRV #2 to 13 – 21% open and setting MFP B speed to minimum

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Reactor Trip override signal will close MFRVs, drive MFW Pumps to minimum speed, and close the SUFRVs to 13 – 21% Open. RTO sends signal directly to the individual component controllers. Adjusting FWCS Master Controller #2 output to less than 3.5% will reset the RTO signal. The output must be lowered.

B. **CORRECT**: Reactor Trip override signal will close MFRVs, drive MFW Pumps to minimum speed, and close the SUFRVs to 13 – 21% Open. RTO sends signal directly to the individual component controllers. Adjusting FWCS Master Controller #2 output to less than 3.5% will reset the RTO signal. The output must be lowered.

C. Incorrect. RTO sends signal directly to the individual component controllers. These actions would be correct for a SG level deviation signal when the MFRV and SUFRV are in manual.

D. Incorrect. RTO sends signal directly to the individual component controllers. The MFP speed controller is designed to go to minimum speed on a RTO signal but it goes to minimum speed if the controller was in manual or auto.

Technical Reference(s): OP-902-000 Revision 15
(Attach if not previously provided) OP-901-201 Revision 6
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-FWC Obj. 6 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS

Verify Core Heat Removal

___5. Determine Core Heat Removal acceptance criteria are met:

___a. Check at least one RCP is operating.

___b. Check operating loop ΔT is less than 13ºF.

___c. Check RCS subcooling is greater than or equal to 28ºF.

Verify RCS Heat Removal

___6. Determine RCS Heat Removal acceptance criteria are met:

___a. Check that at least one steam generator has **BOTH** of the following:

- Steam generator level is 10% to 76% NR.
- Main Feedwater is available to restore level within 55% to 70% NR **[60% to 80% NR].**

a.1 Verify Main Feedwater is restoring level in at least one steam generator within 55% to 70% NR **[60% to 80% NR]**.

a.2 Verify Emergency Feedwater is available to restore level in at least one steam generator within 55% to 70% NR **[60% to 80% NR]**.

(continue)

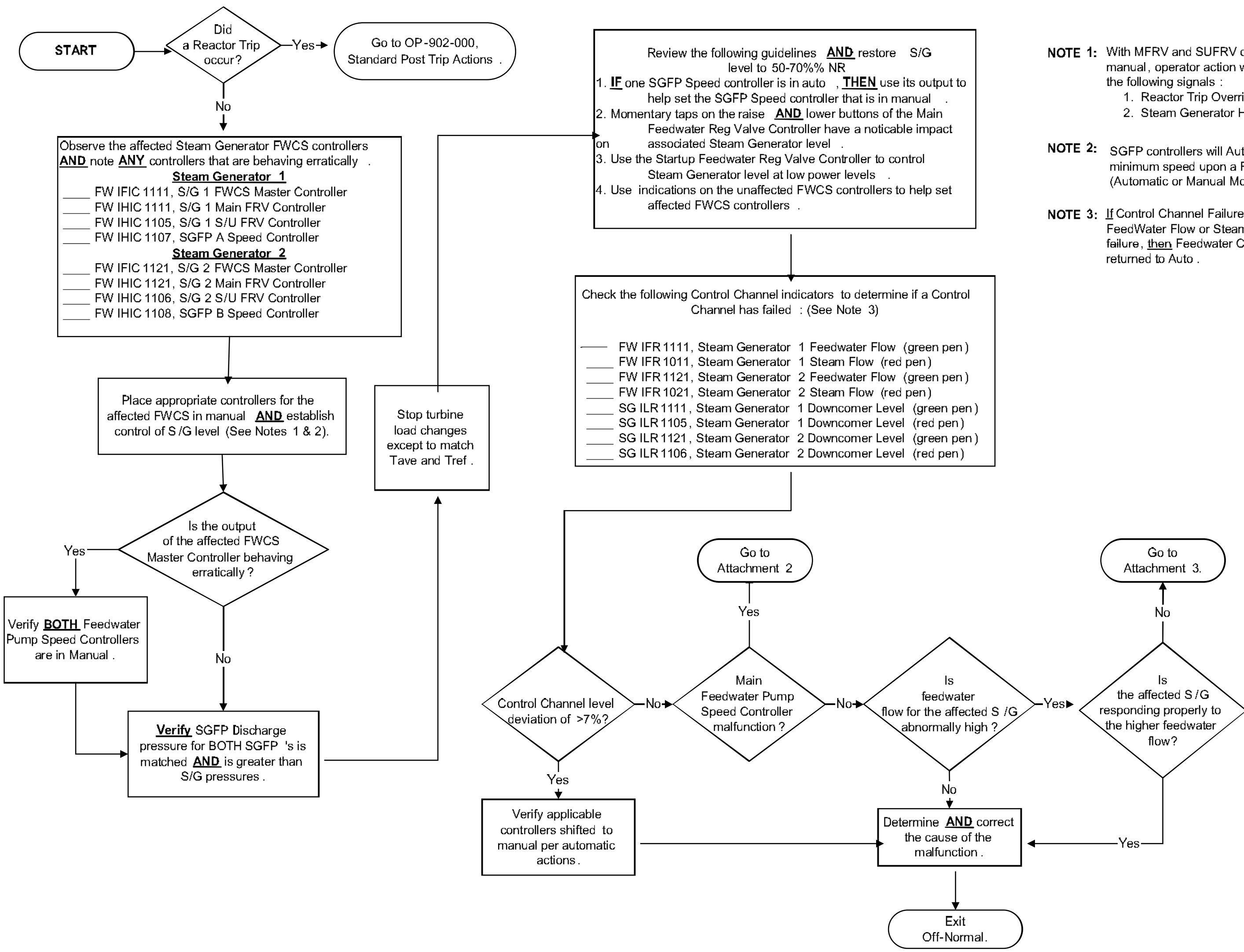
INSTRUCTIONS	CONTINGENCY ACTIONS
6. (continued)	
___ b. Check RCS T_C is 530ºF to 550ºF (continue)	b.1 **IF** RCS T_C is greater than 550ºF, **THEN** confirm that at least one steam generator is removing RCS heat: • Verify level is being restored to at least one steam generator • Verify Steam Bypass or ADVs are restoring RCS T_C 530ºF to 550ºF b.2 **IF** RCS T_C is less than 530ºF, **THEN** perform the following: • Verify feedwater flow is **NOT** excessive • Verify Steam Bypass or ADVs are restoring RCS T_C 530ºF to 550ºF • **IF** RCS T_C is less than 382ºF, **THEN** verify no more than two RCPs are operating • **IF** RCS T_C is being controlled by an ESD, **THEN** REFER TO Appendix 13, "Stabilize RCS Temperature" and stabilize RCS temperature using the least affected steam generator

INSTRUCTIONS	CONTINGENCY ACTIONS
6. (continued)	
___ c. Check steam generator pressure is 885 psia to 1040 psia.	c.1 **IF** steam generator pressure is less than 885 psia, **THEN** perform **ALL** of the following: 1) Verify steam bypass valves are closed. 2) Verify ADVs are closed. c.2 **IF** steam generator pressure is less than or equal to 666 psia, **THEN** verify MSIS is initiated. c.3 **IF** steam generator pressure is greater than 1040 psia, **THEN** verify that Steam Bypass or ADVs are restoring steam generator pressure to less than 1040 psia.
___ d. Check Feedwater Control in Reactor Trip Override: • MAIN FW REG valves are closed • STARTUP FW REG valves are 13% to 21% open • Operating main feedwater pumps are 3800 rpm to 4000 rpm	c.4 Manually operate the Feedwater Control system and restore level in at least one steam generator within 55% to 70% NR **[60%-80% NR]**.

(continue)

INSTRUCTIONS	CONTINGENCY ACTIONS
6. (continued)	
___ e. Reset moisture separator reheaters, and check the temperature control valves closed.	e.1 Isolate the moisture separator reheaters by locally closing the air operated temperature control valves.
Verify Containment Isolation	
___ 7. Determine Containment Isolation acceptance criteria are met:	
___ a. Check containment pressure is less than 16.4 psia.	a.1 **IF** containment pressure is greater than or equal to 17.1 psia, **THEN** verify the following: • CIAS is initiated • SIAS is initiated • MSIS is initiated
___ b. Check **NO** containment area radiation monitor alarms **OR** unexplained rise in activity.	
___ c. Check **NO** steam plant activity monitor alarms **OR** unexplained rise in activity.	

ATTACHMENT 1: GENERAL ACTIONS



NOTE 1: With MFRV and SUFRV controllers in manual , operator action will be required for the following signals :

1. Reactor Trip Override .
2. Steam Generator High Level Override

NOTE 2: SGFP controllers will Automatically go to minimum speed upon a Reactor Trip (Automatic or Manual Mode of Operation).

NOTE 3: If Control Channel Failure was due to FeedWater Flow or Steam Flow instrument failure , then Feedwater Controllers may be returned to Auto .

INTERLOCKS

High Level Override (HLO)
(Figure 22)

The purpose of the HLO is to aid in preventing moisture carryover into the main turbine caused by high SG water level. The HLO logic signal is generated in each FWCS when it senses a high water level in its associated SG. The initiating setpoint is increasing level at 74 percent of the measured narrow range span. The HLO condition will reset when level decreases to 72 percent, and control is returned to the four-element control system.

Each SG (and each FWCS) has two level transmitters (Level 1 and Level 2) and a selector switch to determine if Level 1, Level 2, or both are to be used as inputs to the FWCS.

- If Level 1 or 2 is selected, then that level (1 or 2) must be equal to or greater than 74 percent to initiate HLO.
- If Both is selected, then both level channels must be equal to or greater than 74 percent to initiate HLO.

When the HLO logic is generated, a zero flow demand signal is imposed on both regulating valve programs. This causes both valves to close, and results in a water level decrease in the faulted SG.

The HLO logic also sends a zero flow demand signal to the High Select circuit for its pump speed setpoint program, where it is compared to the flow demand signal from the other FWCS. The larger is used to control the speed of both main feedwater pumps. This insures an adequate feedwater flow to the non-faulted SG.

The HLO zero flow demand is imposed on the pump speed control program and the resulting valve programs only if their respective M/A Stations are in the AUTO mode of operation.

Reactor Trip Override
(Figures 23 and 24)

The purpose of the Reactor Trip Override (RTO) is to prevent overcooling of the reactor coolant system after a reactor trip. This is accomplished by limiting the feedwater flow to a rate which will slowly refill the SGs.

The RTO condition is initiated by a reactor trip through four undervoltage relay status signals received from the Control Element Drive Mechanism Control System (CEDMCS). They are input to FWCS as follows:

- UV 1 and 2 are direct inputs to FWCS 1.
- UV 3 and 4 are direct inputs to FWCS 2.
- FWCS 1 sends UV 1 and 2 to FWCS 2.
- FWCS 2 sends UV 3 and 4 to FWCS 1.

As shown on Figure 23, at least 1 UV signal directly from CEDMCS coupled with one UV signal from the other FWCS is required to initiate RTO. This increases system reliability by requiring both FWCSs to recognize a reactor trip before starting control action.

If one or both pairs of UV signals are present a bistable output is generated to the three time delay networks. The time delay networks will hold an output for 10 seconds after receiving an input signal. This assures that the FWCS will maintain the reactor trip status long enough to switch to RTO. An output from two of three time delay networks generates the RTO logic.

Figure 24 shows that when the RTO logic signal exists, the following limitations are placed on the feedwater regulating valves and the pump speed program:

- Flow demand to the main feedwater pump is limited to 0 percent, holding it at minimum speed. If the M/A station is in Manual, RTO will force the controller to Auto for 5 seconds. This allows the speed to be reduced to minimum speed. After 5 seconds the Operator can place the M/A station in manual as required
- Flow demand to the Startup Regulating valve is limited to ~ 3.5 percent, holding it at its minimum opening.
- Flow demand to the Main Regulating valve is limited to ~ 0 percent, causing it to close.
- With the Feedwater Turbine in Local Manual Control upon receipt of a Reactor Trip Override signal the local station will be inhibited for 5 seconds to allow speed to go to 3900 RPM

When the reactor trips, a large level decrease in the SGs will occur due to shrink. This large level deviation will result in the FWCS generating a large FLOW DEMAND signal. With the RTO present, the FWCS should only supply enough feedwater to make up for decay heat removal and to slowly return the SG downcomer water level to normal. This is accomplished as follows:

- There are three comparators that are constantly comparing the actual FLOW DEMAND signal with minimum flow signal (~ 3.5 percent of rated flow).
- If the FLOW DEMAND signal is greater than ~ 3.5 percent signal, these three comparators seal in the RTO logic signal.
- When the SG levels are returned to a level that results in the FLOW DEMAND signal being less than ~ 3.5 percent, the RTO condition is reset and the FWCS functions as normal in AUTO.

The RTO programs are only activated for the respective M/A Valve Controller Stations that are in the AUTO mode of operation.

Level Deviation Condition

A deviation between the two level measurement channels on the same SG indicates an instrument problem and could result in the FWCS controlling at an improper level. On a deviation greater than 13 in. (approximately 7.2 percent), the following actions occur:

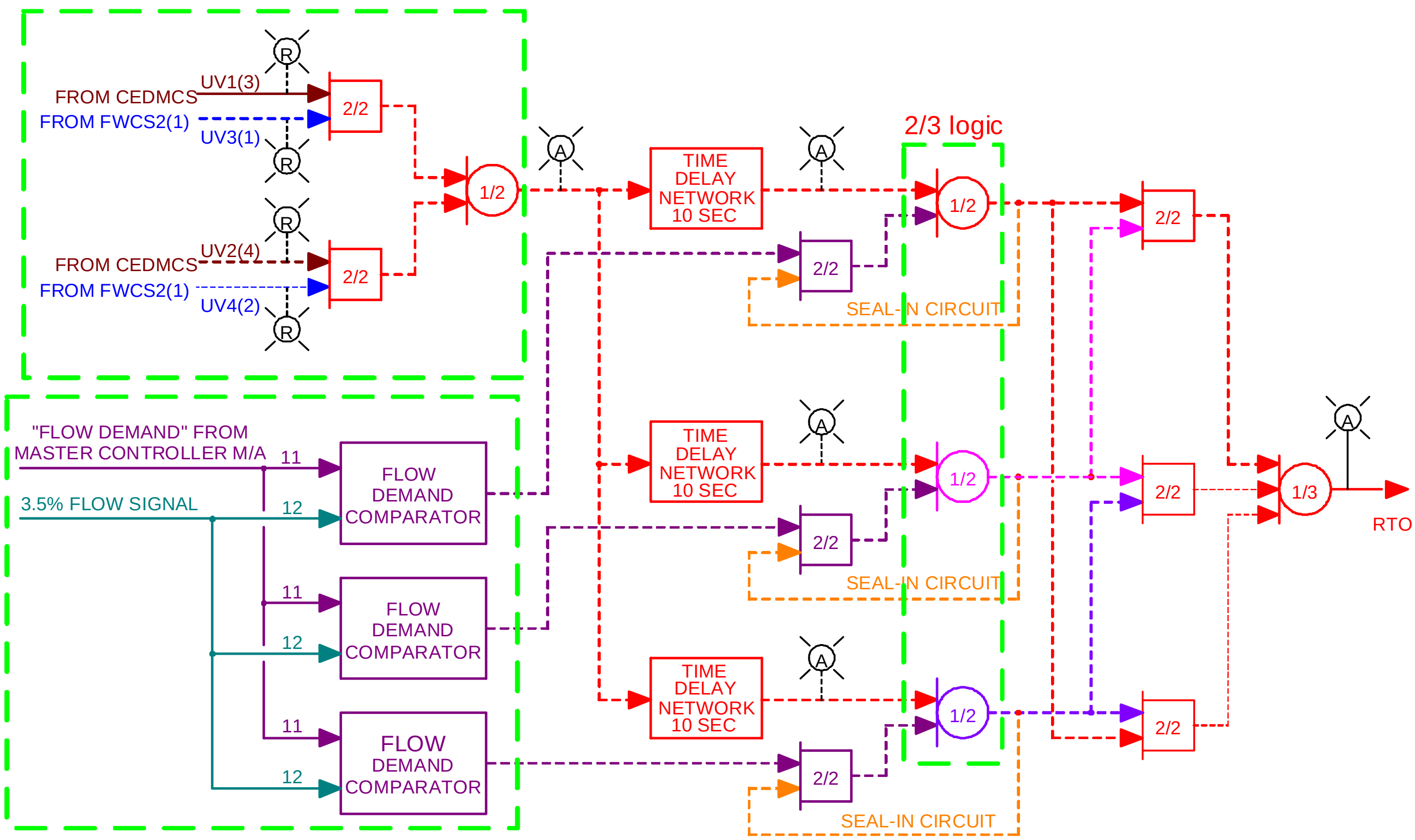
- SG LEVEL DEVIATION alarm - CP-1.
- Main regulating valve M/A Station shifts to MANUAL and holds last automatic output.
- Startup Regulating Valve M/A Station shifts to MANUAL.
- Pump speed M/A Station to MANUAL.

These actions occur regardless of the position of the Level Channel Selector Switch.

If the level deviation alarm cannot be cleared and the AUTOMATIC mode of operation is desired, a jumper must be placed across two front terminal points in the FWCS calculator to allow the M/A Stations to be returned to AUTO.

FIG. 23 REACTOR TRIP OVERRIDE LOGIC DIAGRAM

REF. 457000100



FOR FLOW DEMAND CONPARATOR:

IF 11 > 12, B.S. = 1

IF 11 < I2, B.S. = 0

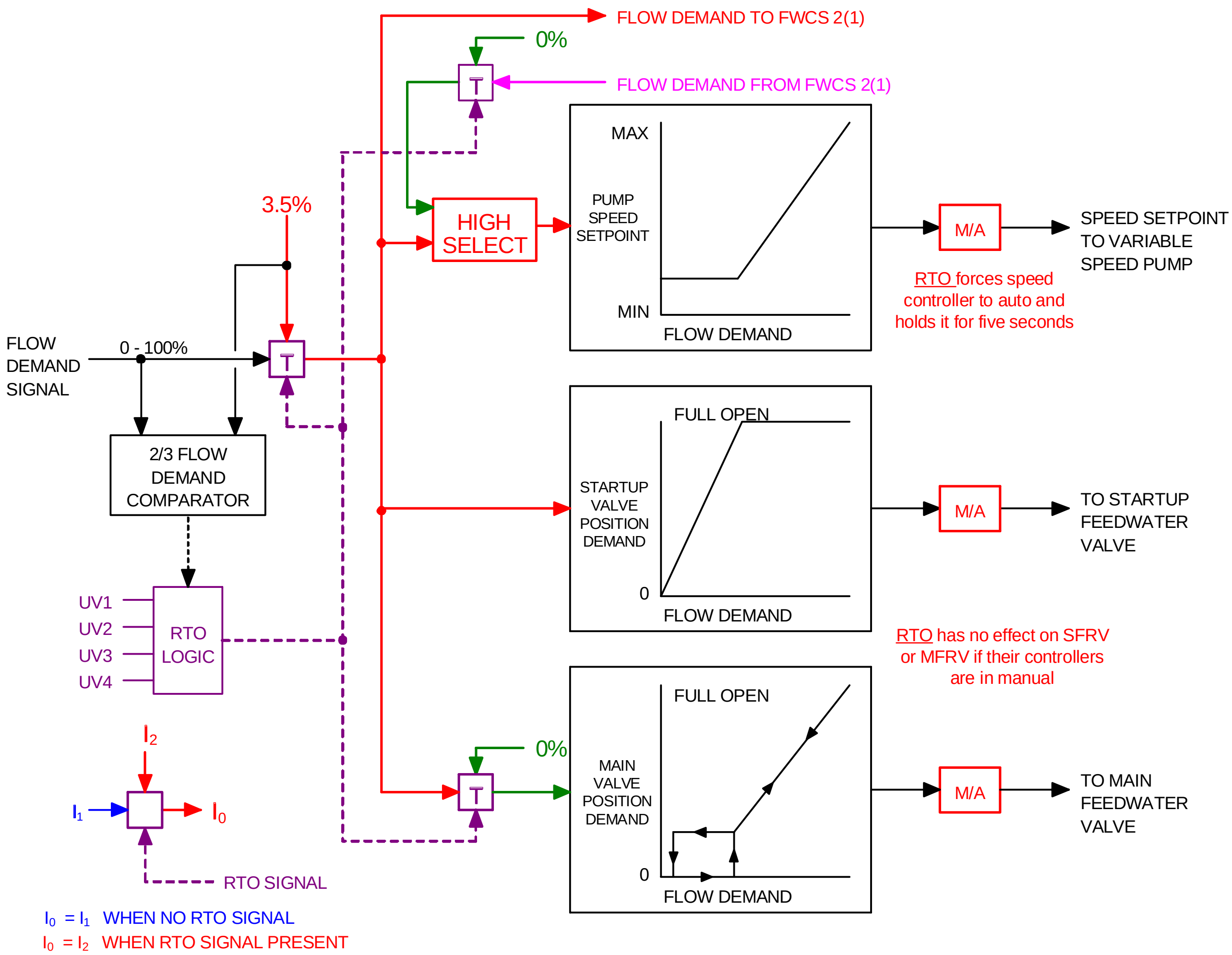
FOR TIME DELAY NETWORK:

WHEN INPUT IS 0, OUTPUT IS 0.

WHEN INPUT IS 1, OUTPUT IS 1 FOR 10 SECONDS, THEN OUTPUT IS 0.

FIG. 24 REACTOR TRIP OVERRIDE FUNCTIONAL BLOCK DIAGRAM

REF. 457000100



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000008 AA2.01	
	Importance Rating	3.9	

K/A Statement

AA2.01 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: RCS pressure and temperature indicators and alarms

Proposed Question: RO 2 Rev: 0

Given:

- Plant is at 100% power
- The following annunciators are in alarm:

  PRESSURIZER RELIEF LINE TEMP HI

  PRESSURIZER RELIEF VALVE OPEN

Which of the following statements is correct?

A. Quench Tank temperature will equal Tsat for the current Pressurizer pressure.

B. Quench Tank pressure will equal Psat for the current Pressurizer vapor space temperature.

C. The downstream Safety Relief temperature detector will indicate Tsat for the current Pressurizer pressure.

D. The downstream Safety Relief temperature detector will indicate Tsat for the current Quench Tank pressure.

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: The applicant must be aware that quench tank pressure will not be at pressurizer pressure and that the throttling process across the open relief valve is isenthalpic.
B. Incorrect: The quench tank pressure will increase until the rupture disc on the quench tank ruptures at 124 psig, but will never be Psat for the existing temperature in the pressurizer vapor space.
C. Incorrect: The downstream Safety Relief temperature will be at Tsat for current Quench tank pressure knowing that the throttling process across the open relief valve is isenthalpic.
D. **CORRECT:** The downstream Safety Relief temperature will be at Tsat for current Quench tank pressure knowing that the throttling process across the open relief valve is isenthalpic.

Technical Reference(s): WLP-OPS-THY04
(Attach if not previously provided)
(including version/revision number)

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

Learning Objective: WLP-OPS-RCS obj. 1 (As available)
WLP-OPS-THY04 obj. 21

Question Source: Bank # 1310-A
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 5
55.43 ______

Comments:

(Figure 29) The quench tank is designed to receive and condense the calculated discharges from the primary (Pressurizer) safety valves and to prevent the discharge from being released to the containment atmosphere.

The tank is sized to receive and condense the steam released by a loss of load event which is less than 1000 lbm. The loss of load event is considered to be the maximum normal discharge that the quench tank must withstand.

The 320 cubic foot (~2400 gal) quench tank and associated blowdown system are sized such that the maximum safety valve backpressure, 500 PSIG, is not reached during any anticipated transient steam release. The quench tank's 20 inch diameter rupture disc has a relief capacity greater than the combined relief capacities of the primary (Pressurizer) safety valves.

The quench tank is an austenitic stainless steel vessel suitable for prolonged contact with borated, demineralized water. Nozzles are provided for the safety valve discharge line, vents, drains, instrumentation, makeup water, nitrogen addition, and the rupture disc. The tank prevents the steam released from the primary (Pressurizer) safety valves from being released to the containment atmosphere.

The sparger, spray header, nozzles and rupture disc fittings are stainless steel.

To reduce the necessary volume and pressure requirements of the quench tank, the relief valve exhaust piping discharges under water. The normal (260 cubic foot) water volume is sufficient to condense the steam flow resulting from the safety valve actuation.

Including the 3 PSIG initial nitrogen pressure blanket (and assuming a 120°F containment ambient temperature), the peak pressure calculated for the limiting events is 85 PSIG which is well below the 124 PSIG setpoint of the 20-inch rupture disc. The 3 PSIG nitrogen pressure blanket is necessary to maintain a nonflammable, non-explosive atmosphere within the quench tank.

In the event that safety valve(s) discharge into the quench tank, provision is made to replace the hot water with cool water. Demineralized primary water is manually added from the Reactor Auxiliary Building to cool the tank water after a steam discharge. Non-condensable gases within the quench tank are vented to the containment vent header through an air-operated valve.

The quench tank is designed to handle the design basis steam releases. However, should the rupture disc burst, the primary coolant released to the containment would be minimal when compared to the LOCA for which the engineered safety features are designed to accommodate. Therefore, no inspection or testing requirements are imposed on the quench tank.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000009 EK2.03	
	Importance Rating	3.0	

K/A Statement

EK2.03 Knowledge of the interrelations between the small break LOCA and the following: S/Gs

Proposed Question: RO 3 Rev: 0

Given:

- Plant has experienced a Small Break LOCA
- Pressurizer Pressure is 1800 psia and lowering slowly
- Containment Pressure is 17.3 psia and lowering slowly
- Both Steam Generator pressures are 970 psia and steady
- The plant has commenced a controlled plant cooldown in accordance with OP-902-002, LOCA Recovery Procedure

To maintain the ability of the EFAS logic to automatically feed the Steam Generators during the controlled plant cooldown, the CRS will direct the ATC to lower the automatic initiation setpoints for which of the following ESFAS signals (if any)?

A. SIAS only

B. MSIS only

C. Both SIAS and MSIS

D. Neither SIAS or MSIS

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: OP-902-002 step 22 directs the crew to lower SIAS automatic initiation setpoints only if SIAS is not present. SIAS is present because containment pressure is >17.1 psia.
B. **CORRECT:** OP-902-002 step 21 directs the crew to lower MSIS automatic initiation setpoints if MSIS is not present or the MSIS is due to high containment pressure. In this case the MSIS is due to high containment pressure and SG pressures are stable. If the MSIS setpoint (low SG pressure) is not reduced, it may prevent automatic feeding of the Steam Generators by EFW.
C. Incorrect: OP-902-002 step 22 directs the crew to lower SIAS automatic initiation setpoints only if SIAS is not present. SIAS is present because containment pressure is >17.1 psia. MSIS initiation setpoints should be reset.
D. Incorrect: Resetting neither SIAS nor MSIS is plausible since both have actuated and the applicant could assume that neither should be reset.

Technical Reference(s): OP-902-002 step 21 and step 22
(Attach if not previously provided) TG OP-902-002 step 21 and step 22
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE02 obj. 17 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

9.0 AUTOMATIC FUNCTIONS

9.1	Hi Linear Power (RPS Trip)	108.0% Rated Thermal Power
9.2	Hi Log Power (RPS Trip)	.257% Rated Thermal Power
9.3	Hi Local Power Density (RPS Trip)	21 KW/FT
9.4	Lo DNBR (RPS Trip)	1.26
9.5	Hi Pzr Press (RPS Trip)	2350 PSIA
9.6	Lo Pzr Press (RPS Trip, SIAS/CIAS Actuation)	1684 PSIA
9.7	Lo SG 1 Level (RPS Trip, EFAS 1 Actuation)	27.4% NR
9.8	Lo SG 2 Level (RPS Trip EFAS 2 Actuation)	27.4% NR
9.9	Hi SG 1 Level (RPS Trip)	87.7% NR
9.10	Hi SG 2 Level (RPS Trip)	87.7% NR
9.11	Lo SG 1 Press (RPS Trip, MSIS Actuation)	666 PSIA
9.12	Lo SG 2 Press (RPS Trip, MSIS Actuation)	666 PSIA
9.13	Hi Cntmt Press (RPS Trip)	17.1 PSIA
9.14	SG Lo Flow (RPS Trip)	19.0 PSID
9.15	Loss Turb (RPS Trip)	45 PSIG Turb Cntrl Oil Press
9.16	Hi Cntmt Press (SIAS/CIAS, MSIS Actuation)	17.1 PSIA
9.17	Hi-Hi Cntmt Press (CSAS Actuation if SIAS is present)	17.7 PSIA
9.18	LO RWSP Level (RAS Actuation)	10% RWSP Level
9.19	Hi SG 1 ΔP (Allows EFAS 1 Actuation following MSIS)	SG 1 123 PSID above SG 2
9.20	Hi SG 2 ΔP (Allows EFAS 2 Actuation following MSIS)	SG 2 123 PSID above SG 1

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

The low steam generator pressure signal provides an input to the MSIS and the EFAS logic. Failure to reset the low steam generator pressure signal may prevent automatic feeding of the steam generators by the EFAS logic.

Reset MSIS Initiation Setpoints

* 21. **IF** MSIS is **NOT** present, **OR** MSIS initiation has occurred due to high containment pressure, **THEN** lower the automatic initiation setpoints as the cooldown and depressurization proceed for MSIS (low SG Pressure).

Reset SIAS Initiation Setpoints

* 22. **IF** SIAS is **NOT** present, **THEN** lower the automatic initiation setpoints as the cooldown and depressurization proceed for SIAS (low PZR Pressure).

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000011 2.4.11	
	Importance Rating	4.0	

K/A Statement

2.4.11 Knowledge of abnormal condition procedures.

Proposed Question: RO 4 Rev: 0

OP-902-002, LOCA Recovery Procedure states that core uncovery and superheated conditions may be expected for up to ____(1)____ minutes. The crew is allowed to stay in OP-902-002 in this condition and are not required to exit to OP-902-008, Functional Recovery Procedure, provided _____(2)_____ .

	(1)	(2)
A.	15	at least one Charging pump is operating with flow > 40 gpm
B.	30	at least one Charging pump is operating with flow > 40 gpm
C.	15	safety injection flow is within the SI flow curves
D.	30	safety injection flow is within the SI flow curves

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: The note preceding safety function 3 and 5 in OP-902-002 indicates that core uncovery may be expected for up to 30 minutes (15 minutes is time for natural circulation to be established) and that transitioning to OP-902-008 is not required provided SI flow curves are met (Charging pumps will be injecting to the RCS but is not specified in the note).
B. Incorrect: The first part is correct. (Charging pumps will be injecting to the RCS but is not specified in the note).
C. Incorrect: The note preceding safety function 3 and 5 in OP-902-002 indicates that core uncovery may be expected for up to 30 minutes (15 minutes is time for natural circulation to be established). The second part is correct.
D. **CORRECT:** The note preceding safety function 3 and 5 in OP-902-002 indicates that core uncovery may be expected for up to 30 minutes and that transitioning to OP-902-008 is not required provided SI flow curves are met.

Technical Reference(s): OP-902-002 Safety Function 3 and 5 Revision 18
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS PPE02 obj. 17 (As available)

Question Source: Bank # 6148-A
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments:

SAFETY FUNCTION:

3. RCS Inventory Control (cont)

NOTE

Core uncovery and superheated conditions may be expected for up to 30 minutes for some LOCA events. If the SI flow is in accordance with the SI flow curves exiting this procedure to the OP-902-008, Functional Recovery Procedure, will not provide any additional guidance to restore inventory control.

PARAMETER	CRITERIA	CRITERIA SATISFIED
Condition 2 RAS NOT Actuated		
a. HPSI flow	Appendix 2-E, "HPSI Flow Curve"	____ ____ ____ ____
b. LPSI flow	Appendix 2-F, "LPSI Flow Curve"	____ ____ ____ ____
c. RVLMS LEVEL PLENUM	$\geq$ 20%	____ ____ ____ ____
Condition 3 RAS Actuated		
a. HPSI flow	Appendix 2-E, "HPSI Flow Curve"	____ ____ ____ ____
b. RVLMS LEVEL PLENUM	$\geq$ 20%	____ ____ ____ ____

SAFETY FUNCTION:

5. Core Heat Removal

------------------------------ **NOTE** ------------------------------

Core uncovery and superheated conditions may be expected for up to 30 minutes for some LOCA events. If the SI flow is in accordance with the SI flow curves exiting this procedure to the OP-902-008, Functional Recovery Procedure, will not provide any additional guidance to restore inventory control.

PARAMETER	CRITERIA	CRITERIA SATISFIED
a. RCS T_H and Representative CET temperature	< superheated	_____ _____ _____ _____

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000015/17 AK3.04	
	Importance Rating	3.1	

K/A Statement

AK3.04 Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Reduction of power to below the steady state power- to-flow limit

Proposed Question: RO 5 Rev: 0

The Steam Generator Low Flow trip is required to provide protection against a Reactor Coolant Pump ____(1)____ event because the DNBR _____(2)_____ .

	(1)	(2)
A.	sheared shaft	trip uses RCP speed sensors to measure flow
B.	sheared shaft	algorithm is only valid with four RCPs running
C.	locked rotor	trip uses RCP speed sensors to measure flow
D.	locked rotor	algorithm is only valid with four RCPs running

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** The CPC generated DNBR trip uses RCP speed sensors to measure RCS flow. The CPCs will not "see" a RCP sheared shaft event. Per TS 2.2.1 basis, this is the reason we have the SG low flow trip.

B. Incorrect: Part 1 is correct. The DNBR algorithm is only valid within limits specified in TS 2.2.1 basis (aux trips). One of the aux trips is RCP speed, but it is for less than 2 RCPs running and its purpose is not to protect against a RCP sheared shaft event.

C. Incorrect: The CPC generated DNBR trip uses RCP speed sensors to measure RCS flow. The CPCs will not "see" a RCP sheared shaft event. Per TS 2.2.1 basis, this is the reason we have the SG low flow trip.

D. Incorrect: The CPC generated DNBR trip uses RCP speed sensors to measure RCS flow. The CPCs will not "see" a RCP sheared shaft event. Per TS 2.2.1 basis, this is the reason we have the SG low flow trip. One of the aux trips is RCP speed, but it is for less than 2 RCPs running and its purpose is not to protect against a RCP sheared shaft event.

Technical Reference(s): TS 2.2.1 Basis
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPS00 Obj. 5 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 6
55.43

Comments:

Local Power Density - High (Continued)

• **(DRN 02-458)**

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the fuel centerline melt Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

• **(DRN 02-458)**

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1860 psia. At this pressure a DNBR - Low trip will automatically occur. This low pressure trip also provides protection against steam generator tube rupture events. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than the fuel design limit such that the decrease

DNBR - Low (Continued)
>(EC-18510, Ch. 64)
in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit of 1.24. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.
<(EC-18510, Ch. 64)

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	≥ 495°F
b.	RCS Cold Leg Temperature-High	< 580°F
c.	Axial Shape Index-Positive	Not more positive than +0.5
d.	Axial Shape Index-Negative	Not more negative than -0.5
e.	Pressurizer Pressure-Low	≥ 1860 psia
f.	Pressurizer Pressure-High	< 2375 psia
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	≤ 7.00
i.	Quality Margin-Low	> 0

>(DRN 04-1243, Ch. 38)
The CPCs contain several auxiliary trip functions which are credited in the safety analysis. These trips manifest themselves as DNBR trips however they are making the trip determination on parameters other than DNBR.

The CPC Variable Overpower Trip (VOPT) is provided to include a trip on power which is compensated for the decalibrating effects of changes in coolant temperature in the reactor vessel downcormer. Additionally, the trip setpoint is allowed to change with slow changes in plant power. Thus at intermediate steady state powers, the plant is protected by a power trip which is a small distance above steady state power levels. The rate at which the automatic increases and decreases in the setpoint may change are limited and accounted for in the safety analysis.

The CPCs contain a trip which detects asymmetries in cold leg loop temperatures resulting from as asymmetric steam generator transient. The trip occurs if the cold leg asymmetry exceeds 11 °F.

The CPCs contain a trip monitoring margin to saturation conditions in the hot legs. A trip will be generated if margin to saturation is less than 13 °F.

>(EC-22385, Ch. 75)
The CPCs contain an auxiliary trip on low RCP speed. The trip will occur if there are less than 2 RCPs running.
<(DRN 04-1243, Ch. 38; EC-22385, Ch. 75)

>(EC-22790, Ch. 66)

<(EC-22790, Ch. 66)

Reactor Coolant Flow - Low

>(DRN 03-6, Ch. 20; EC-8894, Ch. 75)

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a nominal setpoint of 19.00 psid. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

<(DRN 03-6, Ch. 20; EC-8894, Ch. 75)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000022 AA1.01	
	Importance Rating	3.4	

K/A Statement

AA1.01 Ability to operate and/or monitor the following as they apply to the Loss of Reactor Coolant Makeup: CVCS letdown and charging

Proposed Question: RO 6 Rev: 0

Given:

- The plant is at 100% power
- A rupture of the charging line has occurred
- Charging Pumps Header Isolation Valve (CVC 209) is isolated
- Letdown is isolated
- Charging flow can not be established through the normal charging pump discharge path and the crew has entered OP-901-112, Charging or Letdown Malfunction

The crew will align the Charging pumps to discharge through HPSI header ____(1)____ and restore pressurizer level using the ____(2)____ .

	(1)	(2)
A.	A	cold leg injection valves
B.	A	hot leg injection valves
C.	B	cold leg injection valves
D.	B	hot leg injection valves

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**. OP-901-112 section E1 step 6 directs the crew to align charging to HPSI header A and use cold leg injection valves to restore pressurizer level.

B. Incorrect. OP-901-112 section E1 step 6 directs the crew to align charging to HPSI header A and use cold leg injection valves to restore pressurizer level. Hot leg injection valves are plausible because the flow is directed to the hot leg injection line.

C. Incorrect. OP-901-112 section E1 step 6 directs the crew to align charging to HPSI header A and use cold leg injection valves to restore pressurizer level.

D. Incorrect. OP-901-112 section E1 step 6 directs the crew to align charging to HPSI header A and use cold leg injection valves to restore pressurizer level. Hot leg injection valves are plausible because the flow is directed to the hot leg injection line.

Technical Reference(s): OP-901-112 step 6 revision 5
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PP010 obj. 9 (As available)

Question Source: Bank # X Question #31
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2006 RO NRC Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

E_1 CHARGING MALFUNCTION (CONT'D)

CAUTION

IF HPSI PUMPS ARE OPERATING, THEN CHARGING PUMPS SHOULD NOT BE ALIGNED TO HPSI HEADER.

NOTE

Aligning Charging to HPSI Train A renders HPSI train A INOPERABLE and Charging Pumps INOPERABLE. Enter TS 3.5.2 and 3.1.2.4. Refer to TS 3.5.3.

	PLACEKEEPER START	DONE	N/A
6. IF flow can NOT be established through the normal Charging Pump discharge path, THEN align Charging Pumps to discharge through HPSI Header A as follows:	☐	☐	☐
6.1 Locally open CHARGING HEADER XCONN TO HPSI HEADER A ISOLATION (SI 504) (-35 Wing Area, Col. 6A & L).	☐	☐	
6.2 Open ONE of the following Train A HPSI COLD LEG INJECTION valves: • 1A (SI 225A) • 1B (SI 226A) • 2A (SI 227A) • 2B (SI 228A)	☐	☐	
6.3 Locally open CHARGING PUMPS DISCHARGE TO HPSI ISOLATION (CVC 199) (Charging Pump Room A).	☐	☐	

E1 CHARGING MALFUNCTION (CONT'D)

	PLACEKEEPER	
START	DONE	N/A

NOTE

Charging Header flow will not indicate with CHARGING PUMPS HEADER ISOLATION VALVE (CVC 209) closed.

6.4 Close Charging Pumps Header Isolation Valve (CVC 209). ☐ ☐

6.5 Operate Charging Pumps as necessary to maintain Pressurizer level within the limits of Attachment 1, Pressurizer Level Versus Tave Curve. ☐ ☐

7. WHEN repairs have been completed to the Charging Header, THEN restore Charging Pumps discharge alignment to normal as follows: ☐ ☐

7.1 Stop ALL Charging Pumps. ☐ ☐

7.2 Restore HPSI Header as follows: ☐ ☐

7.2.1 Locally close CHARGING HEADER XCONN TO HPSI HEADER A ISOLATION (SI 504) (-35 Wing Area, Col. 6A & L). ☐ ☐

7.2.2 Verify closed the following Train A HPSI COLD LEG INJECTION valves: ☐ ☐

- 1A (SI 225A)
- 1B (SI 226A)
- 2A (SI 227A)
- 2B (SI 228A)

7.2.3 Locally close CHARGING PUMP DISCHARGE TO HPSI ISOLATION (CVC 199) (Charging Pump Room A). ☐ ☐

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL

Tem(F)	90	90	90	90	90	90	90	90	90
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
1.0	11.1	13.5	15.2	16.6	69.1	79.6	95.3	111.1	121.6
2.0	13.8	16.7	18.8	20.6	85.6	98.6	118.1	137.6	150.6
3.0	15.7	18.9	21.3	23.4	97.1	111.9	134.0	156.1	170.9
4.0	17.6	21.2	23.9	26.2	108.9	125.5	150.3	175.1	191.7
5.0	19.5	23.6	26.6	29.1	121.2	139.6	167.2	194.8	213.2
6.0	21.1	25.5	28.7	31.4	130.6	150.4	180.2	210.0	229.8
7.0	22.8	27.6	31.0	34.0	141.4	162.9	195.1	227.3	248.8
8.0	24.1	29.1	32.8	35.9	149.3	172.0	206.0	240.1	262.7
9.0	25.5	30.8	34.7	38.1	158.2	182.3	218.3	254.4	278.4
10.0	26.7	32.3	36.3	39.8	165.7	190.8	228.6	266.3	291.5
11.0	27.9	33.7	37.9	41.6	172.8	199.1	238.5	277.9	304.1
12.0	29.0	35.0	39.4	43.2	179.5	206.8	247.7	288.6	315.9
13.0	29.8	36.1	40.6	44.5	185.1	213.2	255.4	297.6	325.7
14.0	30.8	37.2	41.9	45.9	191.0	220.0	263.5	307.1	336.1
15.0	31.8	38.5	43.3	47.5	197.3	227.3	272.2	317.2	347.2
16.0	32.9	39.8	44.8	49.1	204.1	235.1	281.5	328.0	359.0
17.0	34.1	41.2	46.4	50.8	211.3	243.4	291.5	339.6	371.7
18.0	35.0	42.3	47.6	52.2	217.0	249.9	299.4	348.8	381.8
19.0	35.8	43.2	48.7	53.3	221.8	255.5	306.0	356.6	390.3
20.0	36.6	44.2	49.8	54.6	226.9	261.3	313.0	364.7	399.2
21.0	37.4	45.3	50.9	55.8	232.2	267.4	320.3	373.2	408.5
22.0	38.3	46.3	52.2	57.2	237.7	273.8	328.0	382.1	418.2
23.0	39.3	47.5	53.4	58.6	243.5	280.5	336.0	391.5	428.5
24.0	39.9	48.2	54.2	59.4	247.1	284.7	341.0	397.3	434.8
25.0	40.4	48.8	55.0	60.2	250.4	288.5	345.5	402.6	440.7
26.0	40.9	49.5	55.7	61.0	253.8	292.4	350.2	408.1	446.6
27.0	41.5	50.2	56.5	61.9	257.3	296.4	355.0	413.7	452.7
28.0	42.1	50.9	57.2	62.7	260.9	300.5	360.0	419.4	459.0
29.0	42.7	51.6	58.1	63.6	264.6	304.8	365.1	425.3	465.5
30.0	43.3	52.3	58.9	64.5	268.4	309.1	370.3	431.4	472.2
31.0	43.9	53.1	59.7	65.5	272.3	313.6	375.7	437.7	479.0
32.0	44.6	53.9	60.6	66.4	276.3	318.2	381.2	444.1	486.1
33.0	45.2	54.7	61.5	67.4	280.4	323.0	386.9	450.8	493.4
34.0	45.9	55.5	62.5	68.5	284.7	327.9	392.8	457.6	500.9
35.0	46.6	56.3	63.4	69.5	289.1	333.0	398.8	464.7	508.6
36.0	47.3	57.2	64.4	70.6	293.6	338.2	405.1	472.0	516.6
37.0	48.1	58.1	65.4	71.7	298.3	343.6	411.5	479.5	524.8
38.0	48.9	59.1	66.5	72.9	303.1	349.1	418.2	487.2	533.3
39.0	49.7	60.0	67.6	74.1	308.1	354.8	425.0	495.2	542.0
40.0	50.5	61.1	68.7	75.3	313.2	360.8	432.1	503.5	551.1
41.0	51.4	62.1	69.9	76.6	318.5	366.9	439.5	512.1	560.4
42.0	52.3	63.2	71.1	77.9	324.0	373.2	447.1	520.9	570.1
43.0	53.2	64.3	72.3	79.3	329.7	379.8	454.9	530.1	580.2
44.0	54.1	65.4	73.6	80.7	335.6	386.6	463.1	539.6	590.5
45.0	55.1	66.6	75.0	82.2	341.8	393.7	471.5	549.4	601.3
46.0	56.1	67.9	76.4	83.7	348.1	401.0	480.3	559.6	612.5
47.0	56.8	68.6	77.3	84.7	352.2	405.7	485.9	566.2	619.7
48.0	57.3	69.3	78.0	85.4	355.3	409.3	490.2	571.2	625.1
49.0	57.8	69.9	78.7	86.2	358.5	412.9	494.6	576.3	630.7
50.0	58.3	70.5	79.4	87.0	361.7	416.6	499.0	581.5	636.4
51.0	58.9	71.1	80.1	87.8	365.0	420.4	503.6	586.7	642.2
52.0	59.4	71.8	80.8	88.6	368.3	424.3	508.2	592.1	648.1
53.0	59.9	72.5	81.6	89.4	371.7	428.2	512.9	597.6	654.1
54.0	60.5	73.1	82.3	90.2	375.2	432.2	517.7	603.2	660.2

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	90	90	90	90	90	90	90	90	90
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
55.0	61.1	73.8	83.1	91.1	378.7	436.3	522.6	608.9	666.4
56.0	61.7	74.5	83.9	92.0	382.3	440.4	527.5	614.7	672.7
57.0	62.3	75.2	84.7	92.8	386.0	444.7	532.6	620.6	679.2
58.0	62.9	76.0	85.5	93.7	389.8	449.0	537.8	626.6	685.8
59.0	63.5	76.7	86.4	94.7	393.6	453.4	543.1	632.7	692.5
60.0	64.1	77.5	87.2	95.6	397.5	457.9	548.4	639.0	699.4
61.0	64.7	78.3	88.1	96.5	401.5	462.4	553.9	645.4	706.4
62.0	65.4	79.0	89.0	97.5	405.5	467.1	559.5	651.9	713.5
63.0	66.1	79.8	89.9	98.5	409.6	471.9	565.2	658.5	720.8
64.0	66.7	80.7	90.8	99.5	413.9	476.7	571.0	665.3	728.2
65.0	67.4	81.5	91.8	100.6	418.2	481.7	577.0	672.3	735.8
66.0	68.1	82.4	92.7	101.6	422.6	486.8	583.0	679.3	743.5
67.0	68.9	83.2	93.7	102.7	427.1	491.9	589.2	686.5	751.4
68.0	69.6	84.1	94.7	103.8	431.7	497.2	595.6	693.9	759.5
69.0	70.4	85.1	95.7	104.9	436.3	502.6	602.0	701.5	767.7
70.0	71.0	85.8	96.6	105.8	440.0	506.9	607.2	707.4	774.3
71.0	71.4	86.3	97.2	106.5	442.9	510.2	611.2	712.1	779.4
72.0	71.9	86.9	97.8	107.2	445.9	513.6	615.2	716.8	784.5
73.0	72.4	87.5	98.5	107.9	448.8	517.0	619.3	721.6	789.8
74.0	72.9	88.1	99.1	108.7	451.9	520.5	623.5	726.4	795.1
75.0	73.4	88.7	99.8	109.4	454.9	524.0	627.7	731.3	800.4
76.0	73.9	89.3	100.5	110.1	458.0	527.6	631.9	736.3	805.9
77.0	74.4	89.9	101.2	110.9	461.1	531.2	636.3	741.3	811.4
78.0	74.9	90.5	101.9	111.7	464.3	534.9	640.7	746.5	817.0
79.0	75.4	91.1	102.6	112.4	467.6	538.6	645.1	751.6	822.7
80.0	75.9	91.8	103.3	113.2	470.8	542.3	649.6	756.9	828.4
81.0	76.5	92.4	104.0	114.0	474.1	546.2	654.2	762.2	834.2
82.0	77.0	93.1	104.8	114.8	477.5	550.0	658.8	767.6	840.2
83.0	77.6	93.7	105.5	115.7	480.9	554.0	663.5	773.1	846.2
84.0	78.1	94.4	106.3	116.5	484.4	557.9	668.3	778.7	852.2
85.0	78.7	95.1	107.0	117.3	487.9	562.0	673.2	784.3	858.4
86.0	79.3	95.8	107.8	118.2	491.4	566.1	678.1	790.0	864.7
87.0	79.8	96.5	108.6	119.1	495.1	570.3	683.0	795.8	871.0
88.0	80.4	97.2	109.4	119.9	498.7	574.5	688.1	801.7	877.5
89.0	81.0	97.9	110.2	120.8	502.4	578.8	693.2	807.7	884.0
90.0	81.6	98.7	111.1	121.7	506.2	583.1	698.5	813.8	890.7
91.0	82.3	99.4	111.9	122.7	510.1	587.5	703.7	820.0	897.4
92.0	82.9	100.2	112.8	123.6	513.9	592.0	709.1	826.2	904.3
93.0	83.4	100.9	113.5	124.4	517.4	596.0	713.9	831.8	910.4
94.0	83.9	101.4	114.1	125.1	520.2	599.2	717.7	836.3	915.3
95.0	84.3	101.9	114.8	125.8	523.0	602.5	721.6	840.8	920.2
96.0	84.8	102.5	115.4	126.5	525.9	605.7	725.6	845.4	925.3
97.0	85.3	103.1	116.0	127.2	528.7	609.1	729.5	850.0	930.3
98.0	85.7	103.6	116.7	127.9	531.7	612.4	733.5	854.7	935.4
99.0	86.2	104.2	117.3	128.6	534.6	615.8	737.6	859.4	940.6
100.0	86.7	104.8	118.0	129.3	537.6	619.2	741.7	864.2	945.9

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	100	100	100	100	100	100	100	100	100
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
1.0	10.2	12.3	13.9	15.2	63.3	72.9	87.3	101.8	111.4
2.0	12.7	15.3	17.2	18.9	78.4	90.3	108.2	126.1	138.0
3.0	14.4	17.3	19.5	21.4	89.0	102.5	122.8	143.0	156.6
4.0	16.1	19.5	21.9	24.0	99.8	115.0	137.7	160.5	175.6
5.0	17.9	21.6	24.4	26.7	111.0	127.9	153.2	178.5	195.3
6.0	19.3	23.3	26.3	28.8	119.7	137.8	165.1	192.4	210.6
7.0	20.9	25.3	28.4	31.2	129.6	149.2	178.7	208.3	227.9
8.0	22.1	26.7	30.0	32.9	136.8	157.6	188.8	220.0	240.8
9.0	23.4	28.3	31.8	34.9	145.0	167.0	200.0	233.1	255.1
10.0	24.5	29.6	33.3	36.5	151.8	174.9	209.4	244.0	267.1
11.0	25.5	30.9	34.8	38.1	158.4	182.4	218.5	254.6	278.7
12.0	26.5	32.1	36.1	39.6	164.5	189.5	227.0	264.5	289.4
13.0	27.4	33.1	37.2	40.8	169.6	195.4	234.0	272.7	298.4
14.0	28.2	34.1	38.4	42.1	175.0	201.6	241.5	281.4	308.0
15.0	29.2	35.3	39.7	43.5	180.8	208.3	249.5	290.7	318.1
16.0	30.2	36.5	41.0	45.0	187.0	215.4	258.0	300.6	329.0
17.0	31.2	37.7	42.5	46.6	193.6	223.0	267.1	311.2	340.6
18.0	32.1	38.8	43.6	47.8	198.8	229.0	274.3	319.6	349.8
19.0	32.8	39.6	44.6	48.9	203.3	234.1	280.4	326.7	357.6
20.0	33.5	40.5	45.6	50.0	207.9	239.5	286.8	334.2	365.8
21.0	34.3	41.5	46.7	51.2	212.7	245.0	293.5	342.0	374.3
22.0	35.1	42.5	47.8	52.4	217.8	250.9	300.5	350.1	383.2
23.0	36.0	43.5	49.0	53.7	223.1	257.0	307.9	358.7	392.6
24.0	36.5	44.2	49.7	54.5	226.5	260.9	312.5	364.0	398.4
25.0	37.0	44.7	50.4	55.2	229.5	264.3	316.6	368.9	403.8
26.0	37.5	45.3	51.0	55.9	232.6	267.9	320.9	373.9	409.2
27.0	38.0	46.0	51.7	56.7	235.8	271.6	325.3	379.0	414.9
28.0	38.6	46.6	52.5	57.5	239.1	275.4	329.8	384.3	420.6
29.0	39.1	47.3	53.2	58.3	242.4	279.3	334.5	389.7	426.6
30.0	39.7	47.9	54.0	59.2	245.9	283.3	339.3	395.3	432.7
31.0	40.2	48.6	54.8	60.0	249.5	287.4	344.2	401.1	439.0
32.0	40.8	49.4	55.6	60.9	253.2	291.6	349.3	407.0	445.4
33.0	41.4	50.1	56.4	61.8	256.9	296.0	354.5	413.1	452.1
34.0	42.1	50.9	57.2	62.7	260.8	300.5	359.9	419.3	458.9
35.0	42.7	51.6	58.1	63.7	264.9	305.1	365.4	425.8	466.0
36.0	43.4	52.5	59.0	64.7	269.0	309.9	371.2	432.5	473.3
37.0	44.1	53.3	60.0	65.7	273.3	314.8	377.1	439.3	480.9
38.0	44.8	54.1	60.9	66.8	277.7	319.9	383.2	446.4	488.6
39.0	45.5	55.0	62.0	67.9	282.3	325.2	389.5	453.8	496.7
40.0	46.3	56.0	63.0	69.0	287.0	330.6	396.0	461.4	505.0
41.0	47.1	56.9	64.1	70.2	291.9	336.2	402.7	469.2	513.5
42.0	47.9	57.9	65.2	71.4	296.9	342.0	409.7	477.3	522.4
43.0	48.7	58.9	66.3	72.7	302.1	348.0	416.9	485.7	531.6
44.0	49.6	60.0	67.5	74.0	307.5	354.3	424.3	494.4	541.1
45.0	50.5	61.1	68.7	75.3	313.2	360.7	432.1	503.4	551.0
46.0	51.5	62.2	70.0	76.7	319.0	367.4	440.1	512.8	561.2
47.0	52.1	62.9	70.8	77.6	322.7	371.7	445.3	518.8	567.8
48.0	52.5	63.5	71.5	78.3	325.6	375.0	449.2	523.4	572.8
49.0	53.0	64.0	72.1	79.0	328.5	378.4	453.2	528.0	577.9
50.0	53.5	64.6	72.7	79.7	331.4	381.8	457.3	532.8	583.1
51.0	53.9	65.2	73.4	80.4	334.4	385.2	461.4	537.6	588.4
52.0	54.4	65.8	74.1	81.2	337.5	388.8	465.7	542.6	593.8
53.0	54.9	66.4	74.8	81.9	340.6	392.4	470.0	547.6	599.3
54.0	55.5	67.0	75.5	82.7	343.8	396.0	474.4	552.7	604.9
55.0	56.0	67.7	76.2	83.5	347.1	399.8	478.8	557.9	610.6
56.0	56.5	68.3	76.9	84.3	350.4	403.6	483.4	563.2	616.4

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	100	100	100	100	100	100	100	100	100
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
57.0	57.1	69.0	77.6	85.1	353.7	407.5	488.0	568.6	622.4
58.0	57.6	69.6	78.4	85.9	357.2	411.4	492.8	574.2	628.4
59.0	58.2	70.3	79.2	86.8	360.7	415.4	497.6	579.8	634.6
60.0	58.8	71.0	79.9	87.6	364.2	419.5	502.5	585.5	640.8
61.0	59.3	71.7	80.7	88.5	367.9	423.7	507.6	591.4	647.2
62.0	59.9	72.4	81.5	89.4	371.6	428.0	512.7	597.3	653.8
63.0	60.6	73.2	82.4	90.3	375.4	432.4	517.9	603.4	660.4
64.0	61.2	73.9	83.2	91.2	379.2	436.8	523.2	609.6	667.2
65.0	61.8	74.7	84.1	92.2	383.2	441.4	528.7	616.0	674.2
66.0	62.5	75.5	85.0	93.1	387.2	446.0	534.2	622.5	681.3
67.0	63.1	76.3	85.9	94.1	391.3	450.8	539.9	629.1	688.5
68.0	63.8	77.1	86.8	95.1	395.5	455.6	545.7	635.8	695.9
69.0	64.5	78.0	87.7	96.2	399.8	460.6	551.7	642.8	703.5
70.0	65.0	78.6	88.5	97.0	403.2	464.5	556.3	648.2	709.5
71.0	65.5	79.1	89.1	97.6	405.9	467.5	560.0	652.5	714.1
72.0	65.9	79.7	89.7	98.3	408.6	470.6	563.7	656.8	718.9
73.0	66.3	80.2	90.3	98.9	411.3	473.8	567.5	661.2	723.6
74.0	66.8	80.7	90.9	99.6	414.1	476.9	571.3	665.6	728.5
75.0	67.2	81.3	91.5	100.3	416.8	480.2	575.1	670.1	733.4
76.0	67.7	81.8	92.1	101.0	419.7	483.4	579.1	674.7	738.4
77.0	68.2	82.4	92.7	101.6	422.6	486.7	583.0	679.3	743.5
78.0	68.6	83.0	93.4	102.3	425.5	490.1	587.0	684.0	748.6
79.0	69.1	83.5	94.0	103.1	428.4	493.5	591.1	688.7	753.8
80.0	69.6	84.1	94.7	103.8	431.4	497.0	595.2	693.5	759.1
81.0	70.1	84.7	95.3	104.5	434.5	500.5	599.4	698.4	764.4
82.0	70.6	85.3	96.0	105.2	437.5	504.0	603.7	703.4	769.8
83.0	71.1	85.9	96.7	106.0	440.7	507.6	608.0	708.4	775.3
84.0	71.6	86.5	97.4	106.8	443.8	511.3	612.4	713.5	780.9
85.0	72.1	87.2	98.1	107.5	447.1	515.0	616.8	718.7	786.6
86.0	72.6	87.8	98.8	108.3	450.3	518.7	621.3	723.9	792.3
87.0	73.2	88.4	99.6	109.1	453.6	522.5	625.9	729.2	798.1
88.0	73.7	89.1	100.3	109.9	457.0	526.4	630.5	734.6	804.1
89.0	74.3	89.8	101.0	110.7	460.4	530.3	635.2	740.1	810.1
90.0	74.8	90.4	101.8	111.6	463.9	534.3	640.0	745.7	816.1
91.0	75.4	91.1	102.6	112.4	467.4	538.4	644.8	751.3	822.3
92.0	76.0	91.8	103.4	113.3	470.9	542.5	649.8	757.1	828.6
93.0	76.5	92.4	104.1	114.0	474.1	546.1	654.1	762.2	834.2
94.0	76.9	92.9	104.6	114.7	476.7	549.1	657.7	766.3	838.7
95.0	77.3	93.4	105.2	115.3	479.2	552.0	661.2	770.4	843.2
96.0	77.7	93.9	105.8	115.9	481.9	555.0	664.8	774.6	847.8
97.0	78.2	94.5	106.3	116.5	484.5	558.1	668.5	778.9	852.5
98.0	78.6	95.0	106.9	117.2	487.2	561.2	672.2	783.1	857.1
99.0	79.0	95.5	107.5	117.8	489.9	564.3	675.9	787.5	861.9
100.0	79.5	96.0	108.1	118.5	492.6	567.4	679.6	791.9	866.7

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	110	110	110	110	110	110	110	110	110
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
1.0	9.3	11.2	12.6	13.8	57.5	66.3	79.4	92.5	101.2
2.0	11.5	13.9	15.6	17.1	71.3	82.1	98.3	114.6	125.4
3.0	13.0	15.8	17.8	19.5	80.9	93.1	111.6	130.0	142.3
4.0	14.6	17.7	19.9	21.8	90.7	104.5	125.2	145.8	159.6
5.0	16.3	19.7	22.1	24.3	100.9	116.2	139.2	162.2	177.5
6.0	17.6	21.2	23.9	26.2	108.8	125.3	150.1	174.8	191.3
7.0	19.0	23.0	25.8	28.3	117.7	135.6	162.4	189.3	207.1
8.0	20.1	24.3	27.3	29.9	124.4	143.2	171.6	199.9	218.8
9.0	21.3	25.7	28.9	31.7	131.8	151.8	181.8	211.8	231.8
10.0	22.3	26.9	30.3	33.2	138.0	158.9	190.3	221.8	242.7
11.0	23.2	28.1	31.6	34.6	143.9	165.8	198.6	231.4	253.3
12.0	24.1	29.2	32.8	36.0	149.5	172.2	206.3	240.3	263.0
13.0	24.9	30.1	33.8	37.1	154.1	177.5	212.7	247.8	271.2
14.0	25.7	31.0	34.9	38.3	159.1	183.2	219.5	255.7	279.9
15.0	26.5	32.0	36.1	39.5	164.3	189.3	226.7	264.1	289.1
16.0	27.4	33.1	37.3	40.9	169.9	195.7	234.5	273.2	299.0
17.0	28.4	34.3	38.6	42.3	175.9	202.7	242.7	282.8	309.5
18.0	29.2	35.2	39.7	43.5	180.7	208.1	249.3	290.5	317.9
19.0	29.8	36.0	40.5	44.4	184.7	212.8	254.9	296.9	325.0
20.0	30.5	36.8	41.5	45.5	188.9	217.6	260.7	303.7	332.4
21.0	31.2	37.7	42.4	46.5	193.3	222.7	266.7	310.8	340.1
22.0	31.9	38.6	43.5	47.6	197.9	228.0	273.1	318.2	348.3
23.0	32.7	39.5	44.5	48.8	202.8	233.6	279.8	326.0	356.8
24.0	33.2	40.1	45.2	49.5	205.8	237.1	284.0	330.8	362.1
25.0	33.7	40.7	45.8	50.2	208.6	240.2	287.7	335.3	366.9
26.0	34.1	41.2	46.4	50.9	211.4	243.5	291.6	339.8	371.9
27.0	34.6	41.8	47.0	51.6	214.3	246.8	295.6	344.5	377.0
28.0	35.1	42.4	47.7	52.3	217.3	250.3	299.8	349.3	382.3
29.0	35.6	43.0	48.4	53.0	220.3	253.8	304.0	354.2	387.7
30.0	36.1	43.6	49.1	53.8	223.5	257.4	308.3	359.3	393.2
31.0	36.6	44.2	49.8	54.5	226.7	261.2	312.8	364.5	398.9
32.0	37.1	44.9	50.5	55.4	230.1	265.0	317.4	369.8	404.8
33.0	37.7	45.5	51.3	56.2	233.5	269.0	322.2	375.4	410.8
34.0	38.3	46.2	52.0	57.0	237.1	273.1	327.1	381.1	417.1
35.0	38.8	46.9	52.8	57.9	240.7	277.3	332.1	387.0	423.5
36.0	39.5	47.7	53.7	58.8	244.5	281.6	337.3	393.0	430.1
37.0	40.1	48.4	54.5	59.8	248.4	286.1	342.7	399.3	437.0
38.0	40.7	49.2	55.4	60.7	252.4	290.7	348.2	405.7	444.1
39.0	41.4	50.0	56.3	61.7	256.5	295.5	353.9	412.4	451.4
40.0	42.1	50.9	57.3	62.8	260.8	300.4	359.9	419.3	458.9
41.0	42.8	51.7	58.2	63.8	265.3	305.5	366.0	426.4	466.7
42.0	43.5	52.6	59.2	64.9	269.8	310.8	372.3	433.8	474.8
43.0	44.3	53.6	60.3	66.1	274.6	316.3	378.8	441.4	483.1
44.0	45.1	54.5	61.4	67.2	279.5	322.0	385.6	449.3	491.8
45.0	45.9	55.5	62.5	68.5	284.6	327.8	392.7	457.5	500.7
46.0	46.8	56.5	63.6	69.7	289.9	333.9	399.9	466.0	510.0
47.0	47.3	57.2	64.4	70.6	293.3	337.8	404.6	471.5	516.0
48.0	47.7	57.7	65.0	71.2	295.9	340.8	408.2	475.6	520.6
49.0	48.2	58.2	65.5	71.8	298.5	343.9	411.9	479.9	525.2
50.0	48.6	58.7	66.1	72.5	301.2	347.0	415.6	484.2	529.9
51.0	49.0	59.3	66.7	73.1	303.9	350.1	419.3	488.6	534.8
52.0	49.5	59.8	67.3	73.8	306.7	353.3	423.2	493.1	539.7
53.0	50.0	60.4	68.0	74.5	309.6	356.6	427.1	497.6	544.7
54.0	50.4	60.9	68.6	75.2	312.5	359.9	431.1	502.3	549.7
55.0	50.9	61.5	69.2	75.9	315.4	363.3	435.2	507.0	554.9
56.0	51.4	62.1	69.9	76.6	318.4	366.8	439.3	511.8	560.2

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	110	110	110	110	110	110	110	110	110
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after									
Shutdown	Time To	Time To	Time To	Time To	Time To	Time To	Time To	Time To	Time To
(days)	Boil (min)	Boil (min)	Boil (min)	Boil (min)	Boil (min)	Boil (min)	Boil (min)	Boil (min)	Boil (min)
57.0	51.9	62.7	70.6	77.3	321.5	370.3	443.5	516.8	565.6
58.0	52.4	63.3	71.3	78.1	324.6	373.9	447.8	521.8	571.1
59.0	52.9	63.9	72.0	78.9	327.8	377.5	452.2	526.9	576.7
60.0	53.4	64.6	72.7	79.6	331.0	381.3	456.7	532.1	582.4
61.0	53.9	65.2	73.4	80.4	334.3	385.1	461.3	537.4	588.2
62.0	54.5	65.9	74.1	81.2	337.7	389.0	465.9	542.8	594.1
63.0	55.0	66.5	74.9	82.1	341.1	392.9	470.7	548.4	600.2
64.0	55.6	67.2	75.7	82.9	344.6	397.0	475.5	554.0	606.4
65.0	56.2	67.9	76.4	83.8	348.2	401.1	480.5	559.8	612.7
66.0	56.8	68.6	77.2	84.7	351.9	405.3	485.5	565.7	619.1
67.0	57.4	69.4	78.1	85.6	355.6	409.7	490.7	571.7	625.7
68.0	58.0	70.1	78.9	86.5	359.5	414.1	495.9	577.8	632.4
69.0	58.6	70.9	79.8	87.4	363.4	418.6	501.3	584.1	639.3
70.0	59.1	71.5	80.4	88.2	366.5	422.1	505.6	589.1	644.7
71.0	59.5	71.9	81.0	88.7	368.9	424.9	508.9	593.0	649.0
72.0	59.9	72.4	81.5	89.3	371.3	427.7	512.3	596.9	653.3
73.0	60.3	72.9	82.1	89.9	373.8	430.6	515.7	600.9	657.6
74.0	60.7	73.4	82.6	90.5	376.3	433.4	519.2	604.9	662.1
75.0	61.1	73.9	83.2	91.1	378.8	436.4	522.7	609.0	666.5
76.0	61.5	74.4	83.7	91.8	381.4	439.3	526.2	613.1	671.1
77.0	62.0	74.9	84.3	92.4	384.0	442.3	529.8	617.3	675.7
78.0	62.4	75.4	84.9	93.0	386.7	445.4	533.5	621.6	680.3
79.0	62.8	75.9	85.5	93.7	389.4	448.5	537.2	625.9	685.0
80.0	63.3	76.5	86.1	94.3	392.1	451.6	541.0	630.3	689.8
81.0	63.7	77.0	86.7	95.0	394.8	454.8	544.8	634.7	694.7
82.0	64.2	77.6	87.3	95.7	397.6	458.0	548.6	639.2	699.6
83.0	64.6	78.1	87.9	96.4	400.5	461.3	552.5	643.8	704.6
84.0	65.1	78.7	88.5	97.0	403.4	464.6	556.5	648.4	709.7
85.0	65.6	79.2	89.2	97.7	406.3	468.0	560.6	653.1	714.8
86.0	66.0	79.8	89.8	98.5	409.2	471.4	564.6	657.9	720.0
87.0	66.5	80.4	90.5	99.2	412.3	474.9	568.8	662.7	725.3
88.0	67.0	81.0	91.2	99.9	415.3	478.4	573.0	667.6	730.7
89.0	67.5	81.6	91.8	100.7	418.4	482.0	577.3	672.6	736.2
90.0	68.0	82.2	92.5	101.4	421.6	485.6	581.6	677.7	741.7
91.0	68.5	82.8	93.2	102.2	424.8	489.3	586.0	682.8	747.3
92.0	69.1	83.5	94.0	103.0	428.0	493.0	590.5	688.0	753.0
93.0	69.5	84.0	94.6	103.7	430.9	496.3	594.5	692.6	758.1
94.0	69.9	84.5	95.1	104.2	433.2	499.0	597.7	696.4	762.2
95.0	70.3	84.9	95.6	104.8	435.5	501.7	600.9	700.1	766.3
96.0	70.7	85.4	96.1	105.4	437.9	504.4	604.2	704.0	770.5
97.0	71.1	85.9	96.7	105.9	440.3	507.2	607.5	707.8	774.7
98.0	71.4	86.3	97.2	106.5	442.7	510.0	610.8	711.7	779.0
99.0	71.8	86.8	97.7	107.1	445.2	512.8	614.2	715.7	783.3
100.0	72.2	87.3	98.3	107.7	447.7	515.7	617.6	719.6	787.6

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	120	120	120	120	120	120	120	120	120
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
1.0	8.4	10.1	11.4	12.5	51.8	59.6	71.4	83.2	91.1
2.0	10.4	12.5	14.1	15.4	64.1	73.9	88.5	103.1	112.8
3.0	11.7	14.2	16.0	17.5	72.8	83.8	100.4	117.0	128.0
4.0	13.2	15.9	17.9	19.6	81.6	94.0	112.6	131.2	143.6
5.0	14.7	17.7	19.9	21.9	90.8	104.6	125.3	146.0	159.7
6.0	15.8	19.1	21.5	23.6	97.9	112.7	135.0	157.3	172.2
7.0	17.1	20.7	23.3	25.5	105.9	122.0	146.2	170.3	186.4
8.0	18.1	21.8	24.6	26.9	111.9	128.9	154.4	179.9	196.9
9.0	19.1	23.1	26.0	28.5	118.6	136.6	163.6	190.6	208.6
10.0	20.0	24.2	27.3	29.9	124.1	143.0	171.3	199.6	218.4
11.0	20.9	25.3	28.4	31.2	129.5	149.2	178.7	208.2	227.9
12.0	21.7	26.2	29.5	32.4	134.5	155.0	185.6	216.3	236.7
13.0	22.4	27.1	30.5	33.4	138.7	159.8	191.4	223.0	244.0
14.0	23.1	27.9	31.4	34.4	143.1	164.9	197.5	230.1	251.8
15.0	23.9	28.8	32.5	35.6	147.9	170.3	204.0	237.7	260.1
16.0	24.7	29.8	33.6	36.8	152.9	176.1	211.0	245.8	269.0
17.0	25.6	30.9	34.8	38.1	158.3	182.4	218.4	254.5	278.5
18.0	26.2	31.7	35.7	39.1	162.6	187.3	224.3	261.4	286.1
19.0	26.8	32.4	36.5	40.0	166.2	191.5	229.3	267.2	292.4
20.0	27.4	33.2	37.3	40.9	170.0	195.8	234.6	273.3	299.1
21.0	28.1	33.9	38.2	41.9	174.0	200.4	240.0	279.7	306.1
22.0	28.8	34.7	39.1	42.9	178.1	205.2	245.7	286.3	313.4
23.0	29.5	35.6	40.1	43.9	182.5	210.2	251.8	293.3	321.0
24.0	29.9	36.1	40.7	44.6	185.2	213.3	255.5	297.7	325.8
25.0	30.3	36.6	41.2	45.2	187.7	216.2	258.9	301.7	330.2
26.0	30.7	37.1	41.8	45.8	190.2	219.1	262.4	305.8	334.6
27.0	31.1	37.6	42.3	46.4	192.8	222.1	266.0	310.0	339.2
28.0	31.6	38.1	42.9	47.0	195.5	225.2	269.7	314.3	344.0
29.0	32.0	38.7	43.5	47.7	198.3	228.4	273.5	318.7	348.8
30.0	32.5	39.2	44.2	48.4	201.1	231.6	277.5	323.3	353.8
31.0	32.9	39.8	44.8	49.1	204.0	235.0	281.5	328.0	359.0
32.0	33.4	40.4	45.5	49.8	207.0	238.5	285.6	332.8	364.2
33.0	33.9	41.0	46.1	50.6	210.1	242.0	289.9	337.8	369.7
34.0	34.4	41.6	46.8	51.3	213.3	245.7	294.3	342.9	375.3
35.0	35.0	42.3	47.6	52.1	216.6	249.5	298.8	348.2	381.1
36.0	35.5	42.9	48.3	52.9	220.0	253.4	303.5	353.6	387.1
37.0	36.1	43.6	49.1	53.8	223.5	257.4	308.4	359.3	393.2
38.0	36.7	44.3	49.9	54.7	227.1	261.6	313.3	365.1	399.6
39.0	37.3	45.0	50.7	55.6	230.8	265.9	318.5	371.1	406.1
40.0	37.9	45.8	51.5	56.5	234.7	270.3	323.8	377.3	412.9
41.0	38.5	46.6	52.4	57.4	238.7	274.9	329.3	383.7	419.9
42.0	39.2	47.4	53.3	58.4	242.8	279.7	335.0	390.3	427.2
43.0	39.9	48.2	54.3	59.5	247.1	284.6	340.9	397.2	434.7
44.0	40.6	49.1	55.2	60.5	251.5	289.7	347.0	404.3	442.5
45.0	41.3	50.0	56.2	61.6	256.1	295.0	353.3	411.7	450.6
46.0	42.1	50.9	57.3	62.8	260.9	300.5	359.9	419.3	458.9
47.0	42.6	51.5	57.9	63.5	263.9	304.0	364.1	424.2	464.3
48.0	43.0	51.9	58.5	64.1	266.2	306.7	367.3	428.0	468.4
49.0	43.4	52.4	59.0	64.6	268.6	309.4	370.6	431.8	472.6
50.0	43.8	52.9	59.5	65.2	271.0	312.2	373.9	435.7	476.9
51.0	44.2	53.4	60.1	65.8	273.5	315.0	377.3	439.7	481.2
52.0	44.6	53.8	60.6	66.4	276.0	317.9	380.8	443.7	485.6
53.0	45.0	54.3	61.2	67.0	278.6	320.9	384.3	447.8	490.1
54.0	45.4	54.9	61.7	67.7	281.2	323.9	387.9	452.0	494.7
55.0	45.8	55.4	62.3	68.3	283.8	326.9	391.6	456.2	499.3
56.0	46.3	55.9	62.9	69.0	286.5	330.0	395.3	460.6	504.1

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	120	120	120	120	120	120	120	120	120
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
57.0	46.7	56.4	63.5	69.6	289.3	333.2	399.1	465.0	508.9
58.0	47.2	57.0	64.1	70.3	292.1	336.4	403.0	469.5	513.9
59.0	47.6	57.5	64.8	71.0	294.9	339.7	406.9	474.1	518.9
60.0	48.1	58.1	65.4	71.7	297.9	343.1	410.9	478.8	524.0
61.0	48.6	58.7	66.1	72.4	300.8	346.5	415.1	483.6	529.3
62.0	49.1	59.3	66.7	73.1	303.9	350.0	419.2	488.5	534.6
63.0	49.6	59.9	67.4	73.9	307.0	353.6	423.5	493.5	540.1
64.0	50.1	60.5	68.1	74.6	310.1	357.2	427.9	498.5	545.6
65.0	50.6	61.1	68.8	75.4	313.4	361.0	432.3	503.7	551.3
66.0	51.1	61.8	69.5	76.2	316.7	364.7	436.9	509.0	557.1
67.0	51.7	62.4	70.3	77.0	320.0	368.6	441.5	514.4	563.0
68.0	52.2	63.1	71.0	77.8	323.5	372.6	446.3	520.0	569.1
69.0	52.8	63.8	71.8	78.7	327.0	376.6	451.1	525.6	575.3
70.0	53.2	64.3	72.4	79.4	329.8	379.8	455.0	530.1	580.2
71.0	53.6	64.8	72.9	79.9	331.9	382.3	457.9	533.6	584.0
72.0	53.9	65.2	73.4	80.4	334.1	384.9	461.0	537.1	587.8
73.0	54.3	65.6	73.9	80.9	336.4	387.4	464.1	540.7	591.8
74.0	54.7	66.1	74.4	81.5	338.6	390.0	467.2	544.3	595.7
75.0	55.0	66.5	74.9	82.0	340.9	392.7	470.3	548.0	599.8
76.0	55.4	67.0	75.4	82.6	343.2	395.3	473.5	551.7	603.8
77.0	55.8	67.4	75.9	83.2	345.6	398.0	476.8	555.5	608.0
78.0	56.2	67.9	76.4	83.7	347.9	400.8	480.1	559.3	612.2
79.0	56.6	68.4	76.9	84.3	350.4	403.6	483.4	563.2	616.4
80.0	57.0	68.8	77.5	84.9	352.8	406.4	486.8	567.1	620.7
81.0	57.4	69.3	78.0	85.5	355.3	409.3	490.2	571.1	625.1
82.0	57.8	69.8	78.6	86.1	357.8	412.2	493.7	575.2	629.5
83.0	58.2	70.3	79.1	86.7	360.4	415.1	497.2	579.3	634.0
84.0	58.6	70.8	79.7	87.3	363.0	418.1	500.8	583.5	638.6
85.0	59.0	71.3	80.3	88.0	365.6	421.1	504.4	587.7	643.2
86.0	59.5	71.8	80.9	88.6	368.3	424.2	508.1	592.0	647.9
87.0	59.9	72.4	81.5	89.3	371.0	427.3	511.8	596.3	652.7
88.0	60.3	72.9	82.1	89.9	373.7	430.5	515.6	600.8	657.5
89.0	60.8	73.5	82.7	90.6	376.5	433.7	519.5	605.2	662.4
90.0	61.2	74.0	83.3	91.3	379.3	437.0	523.4	609.8	667.4
91.0	61.7	74.6	83.9	92.0	382.2	440.3	527.3	614.4	672.5
92.0	62.2	75.1	84.6	92.7	385.1	443.6	531.4	619.1	677.6
93.0	62.6	75.6	85.1	93.3	387.7	446.6	534.9	623.3	682.1
94.0	62.9	76.0	85.6	93.8	389.8	449.0	537.8	626.6	685.8
95.0	63.3	76.5	86.1	94.3	391.9	451.4	540.7	630.0	689.5
96.0	63.6	76.9	86.5	94.8	394.1	453.9	543.7	633.4	693.3
97.0	64.0	77.3	87.0	95.3	396.2	456.4	546.7	636.9	697.1
98.0	64.3	77.7	87.5	95.9	398.4	458.9	549.7	640.4	700.9
99.0	64.7	78.2	88.0	96.4	400.6	461.4	552.7	644.0	704.8
100.0	65.0	78.6	88.5	96.9	402.8	464.0	555.8	647.6	708.7

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	130	130	130	130	130	130	130	130	130
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
1.0	7.4	9.0	10.1	11.1	46.0	53.0	63.5	74.0	81.0
2.0	9.2	11.1	12.5	13.7	57.0	65.7	78.7	91.7	100.3
3.0	10.4	12.6	14.2	15.6	64.7	74.5	89.3	104.0	113.8
4.0	11.7	14.2	15.9	17.5	72.6	83.6	100.1	116.7	127.7
5.0	13.0	15.8	17.7	19.4	80.7	93.0	111.4	129.8	142.0
6.0	14.1	17.0	19.1	20.9	87.0	100.2	120.0	139.9	153.1
7.0	15.2	18.4	20.7	22.7	94.2	108.5	129.9	151.4	165.7
8.0	16.1	19.4	21.9	23.9	99.5	114.6	137.3	159.9	175.0
9.0	17.0	20.6	23.2	25.4	105.4	121.4	145.4	169.4	185.4
10.0	17.8	21.5	24.2	26.6	110.4	127.1	152.3	177.4	194.2
11.0	18.6	22.5	25.3	27.7	115.2	132.6	158.9	185.1	202.6
12.0	19.3	23.3	26.3	28.8	119.6	137.8	165.0	192.3	210.4
13.0	19.9	24.1	27.1	29.7	123.3	142.0	170.1	198.2	216.9
14.0	20.6	24.8	27.9	30.6	127.3	146.6	175.6	204.5	223.9
15.0	21.2	25.7	28.9	31.6	131.5	151.4	181.4	211.3	231.3
16.0	22.0	26.5	29.9	32.7	135.9	156.6	187.6	218.5	239.2
17.0	22.7	27.5	30.9	33.9	140.7	162.1	194.2	226.2	247.6
18.0	23.3	28.2	31.8	34.8	144.6	166.5	199.4	232.4	254.3
19.0	23.9	28.8	32.5	35.6	147.8	170.2	203.9	237.5	260.0
20.0	24.4	29.5	33.2	36.4	151.1	174.1	208.5	242.9	265.9
21.0	25.0	30.2	34.0	37.2	154.7	178.1	213.4	248.6	272.1
22.0	25.6	30.9	34.8	38.1	158.4	182.4	218.5	254.5	278.6
23.0	26.2	31.7	35.6	39.0	162.2	186.9	223.8	260.8	285.4
24.0	26.6	32.1	36.2	39.6	164.6	189.6	227.1	264.7	289.7
25.0	26.9	32.6	36.6	40.2	166.8	192.2	230.2	268.2	293.5
26.0	27.3	33.0	37.1	40.7	169.1	194.8	233.3	271.8	297.5
27.0	27.7	33.5	37.7	41.3	171.4	197.5	236.5	275.6	301.6
28.0	28.1	33.9	38.2	41.8	173.8	200.2	239.8	279.4	305.8
29.0	28.5	34.4	38.7	42.4	176.3	203.0	243.2	283.3	310.1
30.0	28.9	34.9	39.3	43.0	178.8	205.9	246.7	287.4	314.5
31.0	29.3	35.4	39.8	43.7	181.4	208.9	250.2	291.6	319.1
32.0	29.7	35.9	40.4	44.3	184.1	212.0	253.9	295.9	323.8
33.0	30.2	36.5	41.0	45.0	186.8	215.2	257.7	300.3	328.6
34.0	30.6	37.0	41.7	45.6	189.6	218.4	261.6	304.8	333.6
35.0	31.1	37.6	42.3	46.4	192.6	221.8	265.7	309.5	338.8
36.0	31.6	38.2	43.0	47.1	195.6	225.3	269.8	314.4	344.1
37.0	32.1	38.8	43.6	47.8	198.7	228.9	274.1	319.4	349.6
38.0	32.6	39.4	44.3	48.6	201.9	232.6	278.6	324.6	355.2
39.0	33.1	40.1	45.1	49.4	205.2	236.4	283.1	329.9	361.1
40.0	33.7	40.7	45.8	50.2	208.7	240.3	287.9	335.4	367.1
41.0	34.3	41.4	46.6	51.1	212.2	244.4	292.8	341.1	373.3
42.0	34.9	42.1	47.4	52.0	215.9	248.6	297.8	347.0	379.8
43.0	35.5	42.9	48.2	52.9	219.7	253.0	303.1	353.1	386.5
44.0	36.1	43.6	49.1	53.8	223.6	257.6	308.5	359.4	393.4
45.0	36.8	44.4	50.0	54.8	227.7	262.2	314.1	366.0	400.5
46.0	37.5	45.3	50.9	55.8	231.9	267.1	319.9	372.8	408.0
47.0	37.9	45.8	51.5	56.5	234.6	270.3	323.7	377.1	412.8
48.0	38.2	46.2	52.0	57.0	236.7	272.6	326.6	380.5	416.4
49.0	38.6	46.6	52.5	57.5	238.8	275.1	329.5	383.9	420.1
50.0	38.9	47.0	52.9	58.0	241.0	277.6	332.4	387.3	423.9
51.0	39.3	47.5	53.4	58.5	243.1	280.1	335.5	390.8	427.8
52.0	39.6	47.9	53.9	59.1	245.4	282.6	338.5	394.4	431.7
53.0	40.0	48.3	54.4	59.6	247.6	285.3	341.7	398.1	435.7
54.0	40.4	48.8	54.9	60.2	250.0	287.9	344.9	401.8	439.8
55.0	40.8	49.2	55.4	60.7	252.3	290.6	348.1	405.6	443.9
56.0	41.1	49.7	55.9	61.3	254.7	293.4	351.4	409.4	448.1

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	130	130	130	130	130	130	130	130	130
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
57.0	41.5	50.2	56.5	61.9	257.2	296.2	354.8	413.4	452.4
58.0	41.9	50.7	57.0	62.5	259.7	299.1	358.2	417.4	456.8
59.0	42.3	51.2	57.6	63.1	262.2	302.0	361.8	421.5	461.3
60.0	42.8	51.7	58.2	63.7	264.8	305.0	365.3	425.7	465.9
61.0	43.2	52.2	58.7	64.4	267.4	308.1	369.0	429.9	470.5
62.0	43.6	52.7	59.3	65.0	270.1	311.2	372.7	434.2	475.3
63.0	44.1	53.3	59.9	65.7	272.9	314.3	376.5	438.7	480.1
64.0	44.5	53.8	60.6	66.4	275.7	317.6	380.4	443.2	485.1
65.0	45.0	54.4	61.2	67.1	278.6	320.9	384.3	447.8	490.1
66.0	45.5	54.9	61.8	67.8	281.5	324.3	388.4	452.5	495.3
67.0	45.9	55.5	62.5	68.5	284.5	327.7	392.5	457.3	500.5
68.0	46.4	56.1	63.2	69.2	287.6	331.2	396.7	462.2	505.9
69.0	46.9	56.7	63.8	70.0	290.7	334.8	401.0	467.3	511.4
70.0	47.3	57.2	64.4	70.6	293.2	337.7	404.5	471.2	515.8
71.0	47.7	57.6	64.8	71.0	295.1	339.9	407.1	474.3	519.1
72.0	48.0	58.0	65.2	71.5	297.0	342.2	409.8	477.5	522.6
73.0	48.3	58.4	65.7	72.0	299.0	344.4	412.5	480.7	526.1
74.0	48.6	58.7	66.1	72.5	301.0	346.7	415.3	483.9	529.6
75.0	48.9	59.1	66.6	73.0	303.1	349.1	418.1	487.2	533.2
76.0	49.3	59.5	67.0	73.4	305.1	351.5	421.0	490.5	536.8
77.0	49.6	60.0	67.5	74.0	307.2	353.9	423.8	493.8	540.5
78.0	50.0	60.4	67.9	74.5	309.3	356.3	426.8	497.2	544.2
79.0	50.3	60.8	68.4	75.0	311.5	358.8	429.7	500.7	548.0
80.0	50.7	61.2	68.9	75.5	313.7	361.3	432.7	504.2	551.8
81.0	51.0	61.6	69.4	76.0	315.9	363.8	435.8	507.7	555.7
82.0	51.4	62.1	69.9	76.6	318.1	366.4	438.9	511.3	559.6
83.0	51.7	62.5	70.4	77.1	320.4	369.0	442.0	515.0	563.6
84.0	52.1	63.0	70.9	77.7	322.7	371.7	445.2	518.7	567.7
85.0	52.5	63.4	71.4	78.2	325.0	374.4	448.4	522.5	571.8
86.0	52.9	63.9	71.9	78.8	327.4	377.1	451.7	526.3	576.0
87.0	53.3	64.4	72.4	79.4	329.8	379.9	455.0	530.1	580.2
88.0	53.7	64.8	73.0	80.0	332.2	382.7	458.4	534.1	584.5
89.0	54.1	65.3	73.5	80.6	334.7	385.6	461.8	538.0	588.9
90.0	54.5	65.8	74.1	81.2	337.2	388.5	465.3	542.1	593.3
91.0	54.9	66.3	74.6	81.8	339.8	391.4	468.8	546.2	597.8
92.0	55.3	66.8	75.2	82.4	342.4	394.4	472.4	550.4	602.4
93.0	55.7	67.3	75.7	83.0	344.7	397.0	475.6	554.1	606.4
94.0	56.0	67.6	76.1	83.4	346.6	399.2	478.1	557.1	609.7
95.0	56.3	68.0	76.5	83.9	348.4	401.3	480.7	560.1	613.0
96.0	56.6	68.4	76.9	84.3	350.3	403.5	483.3	563.1	616.3
97.0	56.9	68.7	77.4	84.8	352.2	405.7	486.0	566.2	619.7
98.0	57.2	69.1	77.8	85.3	354.2	408.0	488.7	569.3	623.1
99.0	57.5	69.5	78.2	85.7	356.1	410.2	491.4	572.5	626.6
100.0	57.8	69.9	78.7	86.2	358.1	412.5	494.1	575.7	630.1

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	140	140	140	140	140	140	140	140	140
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
1.0	6.5	7.9	8.9	9.7	40.3	46.4	55.6	64.8	70.9
2.0	8.1	9.7	11.0	12.0	49.9	57.5	68.9	80.3	87.8
3.0	9.2	11.1	12.4	13.6	56.6	65.3	78.2	91.1	99.7
4.0	10.3	12.4	14.0	15.3	63.6	73.2	87.7	102.2	111.8
5.0	11.4	13.8	15.5	17.0	70.7	81.4	97.5	113.6	124.3
6.0	12.3	14.9	16.7	18.3	76.2	87.8	105.1	122.5	134.0
7.0	13.3	16.1	18.1	19.9	82.5	95.0	113.8	132.6	145.1
8.0	14.1	17.0	19.1	21.0	87.1	100.3	120.2	140.0	153.3
9.0	14.9	18.0	20.3	22.2	92.3	106.3	127.3	148.4	162.4
10.0	15.6	18.9	21.2	23.3	96.6	111.3	133.3	155.3	170.0
11.0	16.3	19.7	22.2	24.3	100.8	116.1	139.1	162.1	177.4
12.0	16.9	20.4	23.0	25.2	104.7	120.6	144.5	168.3	184.2
13.0	17.4	21.1	23.7	26.0	108.0	124.4	149.0	173.6	190.0
14.0	18.0	21.8	24.5	26.8	111.4	128.3	153.7	179.1	196.0
15.0	18.6	22.5	25.3	27.7	115.1	132.6	158.8	185.0	202.5
16.0	19.2	23.2	26.2	28.7	119.0	137.1	164.2	191.3	209.4
17.0	19.9	24.1	27.1	29.7	123.2	142.0	170.0	198.1	216.8
18.0	20.5	24.7	27.8	30.5	126.6	145.8	174.6	203.5	222.7
19.0	20.9	25.3	28.4	31.2	129.4	149.0	178.5	208.0	227.6
20.0	21.4	25.8	29.1	31.9	132.3	152.4	182.6	212.7	232.8
21.0	21.9	26.4	29.8	32.6	135.4	156.0	186.8	217.7	238.3
22.0	22.4	27.1	30.5	33.4	138.7	159.7	191.3	222.9	243.9
23.0	23.0	27.7	31.2	34.2	142.1	163.6	196.0	228.3	249.9
24.0	23.3	28.1	31.7	34.7	144.2	166.1	198.9	231.7	253.6
25.0	23.6	28.5	32.1	35.2	146.1	168.3	201.6	234.8	257.0
26.0	23.9	28.9	32.5	35.7	148.1	170.6	204.3	238.0	260.5
27.0	24.3	29.3	33.0	36.1	150.1	172.9	207.1	241.3	264.1
28.0	24.6	29.7	33.4	36.6	152.2	175.3	210.0	244.6	267.7
29.0	24.9	30.1	33.9	37.2	154.3	177.8	212.9	248.1	271.5
30.0	25.3	30.6	34.4	37.7	156.6	180.3	216.0	251.6	275.4
31.0	25.7	31.0	34.9	38.2	158.8	182.9	219.1	255.3	279.4
32.0	26.0	31.5	35.4	38.8	161.2	185.6	222.3	259.1	283.5
33.0	26.4	31.9	35.9	39.4	163.6	188.4	225.7	262.9	287.8
34.0	26.8	32.4	36.5	40.0	166.1	191.3	229.1	266.9	292.1
35.0	27.2	32.9	37.0	40.6	168.6	194.2	232.6	271.0	296.6
36.0	27.7	33.4	37.6	41.2	171.3	197.3	236.3	275.3	301.3
37.0	28.1	34.0	38.2	41.9	174.0	200.4	240.0	279.7	306.1
38.0	28.6	34.5	38.8	42.6	176.8	203.6	243.9	284.2	311.0
39.0	29.0	35.1	39.5	43.3	179.7	207.0	247.9	288.9	316.1
40.0	29.5	35.7	40.1	44.0	182.7	210.5	252.1	293.7	321.4
41.0	30.0	36.3	40.8	44.7	185.8	214.0	256.3	298.7	326.9
42.0	30.5	36.9	41.5	45.5	189.0	217.7	260.8	303.8	332.5
43.0	31.1	37.6	42.3	46.3	192.3	221.6	265.4	309.2	338.4
44.0	31.6	38.2	43.0	47.1	195.8	225.5	270.1	314.7	344.4
45.0	32.2	38.9	43.8	48.0	199.4	229.6	275.0	320.5	350.7
46.0	32.8	39.6	44.6	48.9	203.1	233.9	280.1	326.4	357.2
47.0	33.2	40.1	45.1	49.5	205.4	236.6	283.4	330.2	361.4
48.0	33.5	40.5	45.5	49.9	207.3	238.7	285.9	333.2	364.6
49.0	33.8	40.8	45.9	50.4	209.1	240.9	288.5	336.1	367.9
50.0	34.1	41.2	46.4	50.8	211.0	243.0	291.1	339.2	371.2
51.0	34.4	41.6	46.8	51.3	212.9	245.2	293.7	342.2	374.6
52.0	34.7	41.9	47.2	51.7	214.9	247.5	296.4	345.4	378.0
53.0	35.0	42.3	47.6	52.2	216.9	249.8	299.2	348.6	381.5
54.0	35.4	42.7	48.1	52.7	218.9	252.1	302.0	351.8	385.1
55.0	35.7	43.1	48.5	53.2	220.9	254.5	304.8	355.1	388.7
56.0	36.0	43.5	49.0	53.7	223.0	256.9	307.7	358.5	392.4

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	140	140	140	140	140	140	140	140	140
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
57.0	36.4	44.0	49.5	54.2	225.2	259.4	310.7	362.0	396.2
58.0	36.7	44.4	50.0	54.7	227.4	261.9	313.7	365.5	400.0
59.0	37.1	44.8	50.4	55.3	229.6	264.5	316.8	369.1	403.9
60.0	37.5	45.3	50.9	55.8	231.9	267.1	319.9	372.7	407.9
61.0	37.8	45.7	51.5	56.4	234.2	269.8	323.1	376.4	412.0
62.0	38.2	46.2	52.0	57.0	236.6	272.5	326.4	380.2	416.2
63.0	38.6	46.7	52.5	57.5	239.0	275.3	329.7	384.1	420.4
64.0	39.0	47.1	53.0	58.1	241.4	278.1	333.1	388.1	424.7
65.0	39.4	47.6	53.6	58.7	243.9	281.0	336.5	392.1	429.1
66.0	39.8	48.1	54.2	59.4	246.5	283.9	340.1	396.2	433.7
67.0	40.3	48.6	54.7	60.0	249.1	287.0	343.7	400.4	438.3
68.0	40.7	49.2	55.3	60.6	251.8	290.0	347.4	404.7	443.0
69.0	41.1	49.7	55.9	61.3	254.5	293.2	351.2	409.1	447.8
70.0	41.5	50.1	56.4	61.8	256.7	295.7	354.2	412.6	451.6
71.0	41.8	50.4	56.8	62.2	258.4	297.6	356.5	415.3	454.6
72.0	42.0	50.8	57.1	62.6	260.1	299.6	358.8	418.1	457.6
73.0	42.3	51.1	57.5	63.0	261.8	301.6	361.2	420.9	460.6
74.0	42.6	51.5	57.9	63.5	263.6	303.6	363.7	423.7	463.7
75.0	42.9	51.8	58.3	63.9	265.4	305.7	366.1	426.6	466.9
76.0	43.2	52.2	58.7	64.3	267.2	307.8	368.6	429.5	470.0
77.0	43.5	52.5	59.1	64.8	269.0	309.9	371.1	432.4	473.3
78.0	43.8	52.9	59.5	65.2	270.9	312.0	373.7	435.4	476.5
79.0	44.1	53.2	59.9	65.7	272.7	314.2	376.3	438.4	479.8
80.0	44.4	53.6	60.3	66.1	274.7	316.4	378.9	441.5	483.2
81.0	44.7	54.0	60.8	66.6	276.6	318.6	381.6	444.6	486.6
82.0	45.0	54.4	61.2	67.1	278.6	320.8	384.3	447.7	490.0
83.0	45.3	54.8	61.6	67.5	280.5	323.1	387.0	450.9	493.5
84.0	45.7	55.2	62.1	68.0	282.6	325.5	389.8	454.2	497.1
85.0	46.0	55.6	62.5	68.5	284.6	327.8	392.6	457.5	500.7
86.0	46.3	56.0	63.0	69.0	286.7	330.2	395.5	460.8	504.3
87.0	46.7	56.4	63.5	69.5	288.8	332.6	398.4	464.2	508.0
88.0	47.0	56.8	63.9	70.1	290.9	335.1	401.4	467.6	511.8
89.0	47.4	57.2	64.4	70.6	293.1	337.6	404.4	471.1	515.6
90.0	47.7	57.7	64.9	71.1	295.3	340.1	407.4	474.7	519.5
91.0	48.1	58.1	65.4	71.6	297.5	342.7	410.5	478.3	523.4
92.0	48.4	58.5	65.9	72.2	299.8	345.3	413.6	481.9	527.4
93.0	48.8	58.9	66.3	72.7	301.8	347.7	416.4	485.2	531.0
94.0	49.0	59.2	66.7	73.1	303.5	349.5	418.7	487.8	533.9
95.0	49.3	59.6	67.0	73.5	305.1	351.4	420.9	490.4	536.7
96.0	49.6	59.9	67.4	73.9	306.8	353.3	423.2	493.1	539.7
97.0	49.8	60.2	67.8	74.3	308.4	355.3	425.5	495.8	542.6
98.0	50.1	60.5	68.1	74.7	310.1	357.2	427.9	498.5	545.6
99.0	50.4	60.9	68.5	75.1	311.9	359.2	430.2	501.3	548.6
100.0	50.7	61.2	68.9	75.5	313.6	361.2	432.6	504.1	551.7

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000025 AK1.01	
	Importance Rating	3.9	

K/A Statement

AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation

Proposed Question: RO 7 Rev: 0

Given:

- The plant is in mode 6
- A loss of all Shutdown Cooling has occurred
- The crew has entered OP-901-131, Shutdown Cooling Malfunction
- RCS level is 15.13 feet
- CET temperature indicates 130 degrees F
- The plant has been shutdown for 50 days

The calculated RCS time to boil is _____________ hours.

A. 38.9

B. 58.0

C. 41.2

D. 47.0

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: Using OP-901-131 Attachment 2, 38.9 hours is the time to boil for the given conditions if the plant is at mid loop. The applicant would be required to know that 15.13 feet is the top of the hot leg and not mid-loop.
B. Incorrect: Using OP-901-131 Attachment 2, 58 hours is the time to boil for the given conditions if the plant is at the RCS flange. The applicant would be required to know that 15.13 feet is the top of the hot leg and not at the RCS flange.
C. Incorrect: Using OP-901-131 Attachment 2, 41.2 hours is the time to boil for the given conditions at the top of the hot leg if CET temperature is 140 degrees F.
D. **CORRECT:** Using OP-901-131 Attachment 2, 47.0 hours is the time to boil for the given conditions. The applicant would be required to know that 15.13 feet is the top of the hot leg and use the correct portion of Attachment 2.

Technical Reference(s): OP-901-131 revision 303
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: OP-901-131 attachment 2 (pages 1-12)

Learning Objective: WLP-OPS-REQ21 obj. 3 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

E1 SYSTEM LEAKAGE (CONT'D)

PLACEKEEPER
START DONE

d. Throttle the following valves as necessary to restore inventory:
- HPSI COLD LEG INJECTION 1A (SI 225B)
- HPSI COLD LEG INJECTION 1B (SI 226B)
- HPSI COLD LEG INJECTION 2A (SI 227B)
- HPSI COLD LEG INJECTION 2B (SI 228B)
- HOT LEG 2 INJECTION FLOW CONTROL (SI 506B).

<u>OR</u>

2.2 For Shutdown Cooling Train B:

a. Start Train A HPSI Pump.

b. Close HPSI HEADER ORIFICE BYPASS (SI 219A).

c. Open HOT LEG 1 INJECTION ISOLATION (SI 502A).

d. Throttle the following valves as necessary to restore inventory:
- HPSI COLD LEG INJECTION 1A (SI 225A)
- HPSI COLD LEG INJECTION 1B (SI 226A)
- HPSI COLD LEG INJECTION 2A (SI 227A)
- HPSI COLD LEG INJECTION 2B (SI 228A)
- HOT LEG 1 INJECTION FLOW CONTROL (SI 506A)

3. Restore <u>AND</u> maintain RCS level $\geq$15.13 feet, top of RCS Hot Leg.

4. Monitor RCS Hot Leg for saturation conditions <u>AND</u> determine RCS heatup rate using <u>EITHER</u>:

- CETs

<u>OR</u>

- <u>IF</u> CETs <u>NOT</u> available, <u>THEN</u> refer to Attachment 2: Calculated RCS Time to Boil.

ATTACHMENT 2: CALCULATED RCS TIME TO BOIL (CONTINUED)

Tem(F)	130	130	130	130	130	130	130	130	130
Level	Mid-Loop	Top-HL	18' MSL	RCS Flange	Pool-30' MSL	Pool-32' MSL	Pool-35' MSL	Pool-38' MSL	Pool-40' MSL
Time after Shutdown (days)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)	Time To Boil (min)
1.0	7.4	9.0	10.1	11.1	46.0	53.0	63.5	74.0	81.0
2.0	9.2	11.1	12.5	13.7	57.0	65.7	78.7	91.7	100.3
3.0	10.4	12.6	14.2	15.6	64.7	74.5	89.3	104.0	113.8
4.0	11.7	14.2	15.9	17.5	72.6	83.6	100.1	116.7	127.7
5.0	13.0	15.8	17.7	19.4	80.7	93.0	111.4	129.8	142.0
6.0	14.1	17.0	19.1	20.9	87.0	100.2	120.0	139.9	153.1
7.0	15.2	18.4	20.7	22.7	94.2	108.5	129.9	151.4	165.7
8.0	16.1	19.4	21.9	23.9	99.5	114.6	137.3	159.9	175.0
9.0	17.0	20.6	23.2	25.4	105.4	121.4	145.4	169.4	185.4
10.0	17.8	21.5	24.2	26.6	110.4	127.1	152.3	177.4	194.2
11.0	18.6	22.5	25.3	27.7	115.2	132.6	158.9	185.1	202.6
12.0	19.3	23.3	26.3	28.8	119.6	137.8	165.0	192.3	210.4
13.0	19.9	24.1	27.1	29.7	123.3	142.0	170.1	198.2	216.9
14.0	20.6	24.8	27.9	30.6	127.3	146.6	175.6	204.5	223.9
15.0	21.2	25.7	28.9	31.6	131.5	151.4	181.4	211.3	231.3
16.0	22.0	26.5	29.9	32.7	135.9	156.6	187.6	218.5	239.2
17.0	22.7	27.5	30.9	33.9	140.7	162.1	194.2	226.2	247.6
18.0	23.3	28.2	31.8	34.8	144.6	166.5	199.4	232.4	254.3
19.0	23.9	28.8	32.5	35.6	147.8	170.2	203.9	237.5	260.0
20.0	24.4	29.5	33.2	36.4	151.1	174.1	208.5	242.9	265.9
21.0	25.0	30.2	34.0	37.2	154.7	178.1	213.4	248.6	272.1
22.0	25.6	30.9	34.8	38.1	158.4	182.4	218.5	254.5	278.6
23.0	26.2	31.7	35.6	39.0	162.2	186.9	223.8	260.8	285.4
24.0	26.6	32.1	36.2	39.6	164.6	189.6	227.1	264.7	289.7
25.0	26.9	32.6	36.6	40.2	166.8	192.2	230.2	268.2	293.5
26.0	27.3	33.0	37.1	40.7	169.1	194.8	233.3	271.8	297.5
27.0	27.7	33.5	37.7	41.3	171.4	197.5	236.5	275.6	301.6
28.0	28.1	33.9	38.2	41.8	173.8	200.2	239.8	279.4	305.8
29.0	28.5	34.4	38.7	42.4	176.3	203.0	243.2	283.3	310.1
30.0	28.9	34.9	39.3	43.0	178.8	205.9	246.7	287.4	314.5
31.0	29.3	35.4	39.8	43.7	181.4	208.9	250.2	291.6	319.1
32.0	29.7	35.9	40.4	44.3	184.1	212.0	253.9	295.9	323.8
33.0	30.2	36.5	41.0	45.0	186.8	215.2	257.7	300.3	328.6
34.0	30.6	37.0	41.7	45.6	189.6	218.4	261.6	304.8	333.6
35.0	31.1	37.6	42.3	46.4	192.6	221.8	265.7	309.5	338.8
36.0	31.6	38.2	43.0	47.1	195.6	225.3	269.8	314.4	344.1
37.0	32.1	38.8	43.6	47.8	198.7	228.9	274.1	319.4	349.6
38.0	32.6	39.4	44.3	48.6	201.9	232.6	278.6	324.6	355.2
39.0	33.1	40.1	45.1	49.4	205.2	236.4	283.1	329.9	361.1
40.0	33.7	40.7	45.8	50.2	208.7	240.3	287.9	335.4	367.1
41.0	34.3	41.4	46.6	51.1	212.2	244.4	292.8	341.1	373.3
42.0	34.9	42.1	47.4	52.0	215.9	248.6	297.8	347.0	379.8
43.0	35.5	42.9	48.2	52.9	219.7	253.0	303.1	353.1	386.5
44.0	36.1	43.6	49.1	53.8	223.6	257.6	308.5	359.4	393.4
45.0	36.8	44.4	50.0	54.8	227.7	262.2	314.1	366.0	400.5
46.0	37.5	45.3	50.9	55.8	231.9	267.1	319.9	372.8	408.0
47.0	37.9	45.8	51.5	56.5	234.6	270.3	323.7	377.1	412.8
48.0	38.2	46.2	52.0	57.0	236.7	272.6	326.6	380.5	416.4
49.0	38.6	46.6	52.5	57.5	238.8	275.1	329.5	383.9	420.1
50.0	38.9	47.0	52.9	58.0	241.0	277.6	332.4	387.3	423.9
51.0	39.3	47.5	53.4	58.5	243.1	280.1	335.5	390.8	427.8
52.0	39.6	47.9	53.9	59.1	245.4	282.6	338.5	394.4	431.7
53.0	40.0	48.3	54.4	59.6	247.6	285.3	341.7	398.1	435.7
54.0	40.4	48.8	54.9	60.2	250.0	287.9	344.9	401.8	439.8
55.0	40.8	49.2	55.4	60.7	252.3	290.6	348.1	405.6	443.9
56.0	41.1	49.7	55.9	61.3	254.7	293.4	351.4	409.4	448.1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000027 AA2.15	
	Importance Rating	3.7	

K/A Statement

AA2.15 Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Actions to be taken if PZR pressure instrument fails high

Proposed Question: RO 8 Rev: 0

Given:

- Plant is at 100% power
- Pressurizer Pressure Channel X/Y recorder (RC-IPR-0100) indicates that the in-service Pressurizer Pressure Control Channel instrument has failed high

In accordance with OP-901-120, Pressurizer Pressure Control Malfunction, the CRS will direct the ATC to place the ____(1)____ controller in manual to restore pressurizer pressure. The ATC will _____(2)_____ the output of the controller that was placed in manual to restore pressurizer pressure to setpoint.

	(1)	(2)
A.	Pressurizer Pressure	raise
B.	Pressurizer Pressure	lower
C.	Pressurizer Spray	raise
D.	Pressurizer Spray	lower

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: The pressurizer pressure controller output is the input to the spray controller and would close the spray valve if its ouput was lowered. This is not the controller that OP-901-120 directs using in manual. Raising the output of the pressure controller would open the spray valve.
B. Incorrect. The pressurizer pressure controller output is the input to the spray controller and would close the spray valve if its ouput was lowered. This is not the controller that OP-901-120 directs using in manual.
C. Incorrect. The first part is correct. OP-901-120 directs the ATC to lower spray controller output to 0%, which means the ATC would have to lower output.
D. **CORRECT:** OP-901-120 directs the crew to take manual control of the pressurizer spray controller and adjust output to 0% if the in-service Pressurizer Pressure Control Channel instrument has failed high.

Technical Reference(s): OP-901-120 revision 302
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PP010 obj. 3 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7, 10
55.43 ______

Comments:

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

CAUTION

STEAM GENERATOR PRESSURES DROPPING CONCURRENTLY WITH DROPPING PRESSURIZER LEVEL MAY BE INDICATIVE OF AN EXCESS STEAM DEMAND.

	PLACEKEEPER START	PLACEKEEPER DONE	PLACEKEEPER N/A
1. If Pressurizer Pressure and Level are dropping concurrently, or RCS leakage is otherwise indicated, then go to OP-901-111, Reactor Coolant System Leak.	☐	☐	☐
2. If Pressurizer Pressure is dropping and any of the following have occurred, then place Pressurizer Spray Controller (RC-IHIC-0100) to MAN and adjust output to 0%: • Pressurizer Pressure Channel X/Y recorder (RC-IPR-0100) indicates in-service Pressurizer Pressure Control Channel instrument has failed high • Pressurizer Pressure controller (RC-IPIC-0100) output has failed high • Any Pressurizer Spray Valve (RC-301A or RC-301B) has failed open • Pressurizer Spray Controller (RC-IHIC-0100) output has failed high.	☐	☐	☐
2.1 If any Pressurizer spray valve remains failed open, then Place Pressurizer Spray Valves selector switch to select operable Spray Valve.	☐	☐	☐
3. If Pressurizer Pressure Channel X/Y recorder (RC-IPR-0100) indicates a Pressurizer Pressure Control Channel instrument has failed, then go to Subsection E_1, Pressurizer Pressure Control Channel Instrument Failure.	☐	☐	☐

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000029 EA1.12	
	Importance Rating	4.1	

K/A Statement

EA1.12 Ability to operate and monitor the following as they apply to a ATWS: M/G set power supply and Reactor Trip Breakers

Proposed Question: RO 9 Rev: 0

Given:

- Plant operating at 90% power
- DRTS is out of service
- A loss of the 1A bus occurs and all 4 RCS low flow trip signals are generated, but NO Reactor Trip Circuit Breakers open. The BOP operator pushes the Reactor Trip push buttons on **CP-8**

Based on these actions, Reactor Trip Circuit Breakers __(1)__ should directly OPEN. If not, the BOP will manually open the __(2)__ feeder breakers.

	(1)	(2)
A.	1, 4, 5, and 8	31A and 31B
B.	2, 3, 6, and 7	31A and 31B
C.	1, 4, 5, and 8	32A and 32B
D.	2, 3, 6, and 7	32A and 32B

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: Pushbuttons B and C are located on CP-8 which will directly open Reactor Trip Breakers 2, 3, 6 and 7. Pushbuttons A and D are located on CP-2 which will directly open Reactor Trip Breakers 1, 4, 5 and 8. If DRTS is out of service, the ATC will open the 32 feeder breakers. The 31 feeder breakers can be manually opened from the Control Room.
B. Incorrect: Pushbuttons B and C are located on CP-8 which will directly open Reactor Trip Breakers 2, 3, 6 and 7. If DRTS is out of service, the ATC will open the 32 feeder breakers. The 31 feeder breakers can be manually opened from the Control Room.
C. Incorrect: Pushbuttons B and C are located on CP-8 which will directly open Reactor Trip Breakers 2, 3, 6 and 7. Pushbuttons A and D are located on CP-2 which will directly open Reactor Trip Breakers 1, 4, 5 and 8. If DRTS is out of service, the ATC will open the 32 feeder breakers.
D. **CORRECT**: Pushbuttons B and C are located on CP-8 which will directly open Reactor Trip Breakers 2, 3, 6 and 7. If DRTS is out of service, the ATC will open the 32 feeder breakers.

Technical Reference(s): OP-902-000 revision 15
(Attach if not previously provided) SD-PPS Revision 15
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CED00 obj. 2 (As available)
WLP-OPS-PPE01 obj. 9

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Closing the Reactor Trip Breakers – Refer to Figure 07. Reactor trip breakers close whenever the closing coil (CC) is energized. Energizing the closing coil causes an upward movement of the coil armature, which initiates the mechanical closing action. A closing signal energizes the X relay, which closes contacts to pick up CC. When the breaker closes, bb opens and aa closes to pick up relay Y. Relay Y is an anti-pumping relay and will lock out the X relay until the close signal goes away. Notice that relay X and the closing coil (CC) are de-energized once the breaker closes. Reactor trip breakers 1 through 8 are normally operated from the Control Room. The tie breaker (TCB-9) cannot be operated from the Control Room and it is closed by pushing the close pushbutton mounted on the breaker. Reactor trip breakers 1-8 also have this pushbutton but they are covered to prevent use. Additionally, TCB-1 through 9 may be closed by pushing the Close button on the left side of the breaker cubicle (inside) only when the breaker is in the Test position.

To close breakers 1 through 8 from the Control Room, take the Trip Reset Path key switch on CP-10 to the UNLK position and push the RPS button. CP-10 is divided into 4 sections, channels A through D. The close buttons are arranged on CP-10 as follows:

Channel A	RPS-1	TCB-1
	RPS-5	TCB-5
Channel B	RPS-2	TCB-2
	RPS-6	TCB-6
Channel C	RPS-3	TCB-3
	RPS-7	TCB-7
Channel D	RPS-4	TCB-4
	RPS-8	TCB-8

Opening the Reactor Trip Breakers – Refer to Figure 07. The operating springs, which provide the motive force for tripping the breaker, are charged by the closing action of the breaker. When the breaker is closed, a 52 contact closes in the path for 52/TC. Completing the rest of the trip circuit will energize 52/TC which will cause the breaker to trip.

TCB-1 through 8 may be opened from the Control Room. There are four red buttons labeled Reactor Trip A, B, C, or D. B and C are on CP-8. A and D are on CP-2. The buttons trip the breakers as follows:

Channel A	TCB-1	TCB-5
Channel B	TCB-2	TCB-6
Channel C	TCB-3	TCB-7
Channel D	TCB-4	TCB-8

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
Verify Reactivity Control	
___1. Determine Reactivity Control acceptance criteria are met:	
___a. Check reactor power is dropping.	a.1 Perform the following as necessary to insert CEAs: 1) Manually trip the reactor. 2) Manually initiate DIVERSE REACTOR TRIP. 3) Open **BOTH** the following breakers for 5 seconds and close: • SST A32 FEEDER • SST B32 FEEDER
___b. Check startup rate is negative.	
___c. Check less than **TWO** CEAs are **NOT** fully inserted.	c.1 Commence emergency boration.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000038 EK 1.02	
	Importance Rating	3.2	

K/A Statement

EK1.02 Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop

Proposed Question: RO 10 Rev: 0

With a Steam Generator Tube Rupture in progress it is ultimately desired to depressurize the RCS to within ___(1)___ psid of the ruptured Steam Generator to ___(2)___ and minimize the potential for Steam Generator overfill.

	(1)	(2)
A.	50	minimize the potential release to the environment
B.	50	prevent a loss of subcooled margin
C.	100	minimize the potential release to the environment
D.	100	prevent a loss of subcooled margin

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** Per step 12 of OP-902-007 and TG-OP-902-007, 50 psid is the correct pressure differential. Per TG-OP-902-007 this minimizes pri-to-sec leakage which minimizes release magnitude and helps to prevent S/G overfill.
B. Incorrect. Right D/P, wrong bases. The substep within step 12 that provides instructions for maintaining RCS pressure within the P/T curve limits prevents loss of subcooled margin and allows continued operation of RCPs.
C. Incorrect. Per step 12 of OP-902-007 and TG-OP-902-007, 50 psid is the correct pressure differential; right bases.
D. Incorrect. Per step 12 of OP-902-007 and TG-OP-902-007, 50 psid is the correct pressure differential. . The substep within step 12 that provides instructions for maintaining RCS pressure within the P/T curve limits prevents loss of subcooled margin and allows continued operation of RCPs.

Technical Reference(s): OP-902-007 revision 15
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE07 obj. 8 (As available)

Question Source: Bank # X Question #19
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS
Cooldown RCS to Less Than 520°F T_H	
11. Commence a rapid RCS cooldown to less than 520°F T_H using the steam bypass valves.	11.1 Commence a rapid RCS cooldown to less than 520°F T_H using **BOTH** atmospheric dump valves.
Maintain RCS Pressure	
* 12. Depressurize the RCS:	
a. Maintain pressurizer pressure within **ALL** of the following criteria: • Within Appendix 2A-D, "RCS Pressure and Temperature Limits" • Less than 930 psia • Within 50 psi of the most affected steam generator pressure • **IF** RCPs are operating, greater than the minimum RCP NPSH of Appendix 2A-D, "RCS Pressure and Temperature Limits"	
b. Operate main or auxiliary pressurizer spray.	
c. **IF** HPSI throttle criteria are met, **THEN** perform **ANY** of the following: • Control charging and letdown flow • Throttle HPSI flow	

Step Number 12 Maintain RCS Pressure

Objective

The intent of this step is to establish control of RCS pressure. The general goals associated with RCS pressure control are:

- Providing subcooling to support the core heat removal process
- Avoiding overpressure situations for PTS and RT NDT considerations
- Minimizing the pressure differential between the steam generator and the RCS to minimize the leakage
- Deliberately creating a primary to secondary differential pressure to establish backflow to control SG level rise or reduce SG pressure/temperature
- Controlling RCS pressure below the atmospheric dump valve or main steam safety valve lift pressure to prevent uncontrolled release of radioactivity to the environment

Instructions

Maintaining the RCS pressure below the lift setpoint of the ADV, within the PT limits and approximately equal to the isolated steam generator pressure [±50 psi] will minimize the loss of primary fluid to the secondary side and the possibility of overfilling the isolated SG. This action will minimize the potential for release of radiation to the environment by minimizing RCS to steam generator leakage.

The RCS pressure limit is reduced to 930 psia from 945 psia. The 930 psia is based on instrument uncertainty for a containment temperature of 200°F while the 945 psia was based on instrument uncertainty for a containment temperature of 120°F. This change is consistent in applicable EOPs.

Maintaining RCS pressure approximately equal to or less than the affected SG pressure allows for the backflow of secondary water into the RCS which provides several operational benefits. These benefits include:

- SG level can be maintained within the indicating range by controlling SG pressure
- The probability of filling the main steam piping with water is greatly reduced
- Use of the blowdown system for SG level control can be minimized, thus minimizing contamination of the secondary
- Depressurization of the isolated SG can be performed without steaming to the condenser or to the atmosphere
- Less secondary makeup water is required for the RCS cooldown

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000040 CE/E05 2.1.27	
	Importance Rating	3.9	

K/A Statement

2.1.27 Knowledge of system purpose and/or function.

Proposed Question: RO 11 Rev: 0

Given:

- An Excess Steam Demand has occurred for Steam Generator #1
- Crew has entered OP-902-004, Excess Steam Demand Recovery Procedure
- Steam Generator # 1 level is 20% WR and lowering
- The CRS has directed the BOP to perform actions to stabilize RCS temperature when conditions permit

The BOP will take actions to stabilize RCS temperature once ____(1)_____ .
The stabilization of RCS temperature is required to prevent _______(2)______ .

	(1)	(2)
A.	CET temperature **and** pressurizer pressure rise	RCS pressure rising above HPSI shutoff head
B.	CET temperature **and** pressurizer pressure rise	Pressurized Thermal Shock (PTS)
C.	CET temperature **or** pressurizer pressure rise	Pressurized Thermal Shock (PTS)
D.	CET temperature **or** pressurizer pressure rise	RCS pressure rising above HPSI shutoff head

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: The note prior to OP-902-004 step 16 states that actions to stabilize RCS temperature should be initiated when both CET and pressurizer pressure rise. These actions are taken to prevent PTS. RCS pressure will be stabilized above HPSI shutoff head to prevent going solid in the pressurizer.
B. **CORRECT:** The note prior to OP-902-004 step 16 states that actions to stabilize RCS temperature should be initiated when both CET and pressurizer pressure rise. These actions are taken to prevent PTS.
C. Incorrect: The note prior to OP-902-004 step 16 states that actions to stabilize RCS temperature should be initiated when both CET and pressurizer pressure rise. These actions are taken to prevent PTS.
D. Incorrect: The note prior to OP-902-004 step 16 states that actions to stabilize RCS temperature should be initiated when both CET and pressurizer pressure rise. These actions are taken to prevent PTS. RCS pressure will be stabilized above HPSI shutoff head to prevent going solid in the pressurizer.

Technical Reference(s): OP-902-004 revision 14
(Attach if not previously provided) TGOP-902-004 revision 305
(including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE04 obj. 3 and 4 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 5, 10
55.43 ______

Comments:

INSTRUCTIONS | CONTINGENCY ACTIONS

----- **NOTE** -----

Actions to stabilize RCS temperature following an excess steam demand event should be initiated when **BOTH** of the following parameters are met:

- CET temperatures rise
- Pressurizer pressure rise

Stabilize RCS Temperature

* 16. Verify RCS temperature is stabilized by performing the following:

a. Place the ADV for the least affected steam generator to manual and fully open the ADV.

b. Manually initiate EFAS for the least affected steam generator.

c. Place the EFW Flow Control Valve to manual and commence feeding the least affected steam generator.

d. **IF** RCS pressure is greater than or equal to 1500 psia, **THEN** stabilize RCS pressure at a value not to exceed 1600 psid between the RCS and the lowest SG pressure.

d.1 **IF** RCS pressure is less than 1500 psia, **THEN** stabilize RCS pressure at greater than HPSI shutoff head (1500-1600 psia).

Step Number 16 Stabilize RCS Temperature

Objective

The intent of this step is to direct the operator to take control of, and stabilize, RCS temperature following an ESDE.

Instructions

Stabilize RCS temperature by steaming the least affected SG using the atmospheric dump valves.

The ADV on the least affected steam generator is opened fully when the affected steam generator is empty. The operator also needs to be aware of the PTS concerns following steam generator dryout and the factors or mitigating trends that tend to lessen PTS concerns. These include: 1) limiting RCS repressurization as much as possible while maintaining RCS subcooling requirements, and 2) maintaining control of RCS temperature within the limits of the Post Accident PT curve.

In case the atmospheric dump valves are not available, then SG safeties on the unaffected SG will serve as a heat removal method. Use of SG safeties are not preferred. Therefore, every effort should be made to regain use of the atmospheric dump valves to eliminate the need for RCS repressurization.

RCS temperature will rise after the affected steam generator dries out unless a means of controlling RCS heat removal is established. The rise in RCS temperature may result in a water solid condition due to the inventory added from safety injection and charging operation during the blowdown phase of the event. The post dryout heatup and repressurization also presents a PTS concern.

In order to mitigate RCS heatup, a controllable heat removal method should be established before the dryout condition occurs. The intent is to regain RCS temperature control and stabilize Tc, thus preventing uncontrolled RCS heatup and repressurization.

Feedwater temperature and flow rate can have a significant affect on RCS temperatures. Therefore, the affect of feedwater should be taken into account when attempting to stabilize RCS temperatures. A rapid refill of the unaffected steam generator with cold feedwater may: 1) further the cooldown, or 2) provide a substantial RCS heat sink and thus mitigate the RCS heatup.

Step Number 16 Stabilize RCS Temperature (cont)

In order to provide additional cooling to limit the RCS heatup the least affected steam generator is fed using emergency feedwater.

RCS pressure should be allowed to rise above HPSI shutoff head and then stabilize. The operator should at all times maintain the RCS within the RCS Pressure Temperature Limits. Allowing RCS pressure to rise above HPSI shutoff head (1500-1600 psia) minimizes the probability of the RCS becoming water solid due to continued flow from the HPSI pumps.

IF RCS pressure does not fall below HPSI shutoff head, the operator should stabilize RCS pressure at a value not to exceed 1600 psid between RCS and SG pressure. The maximum differential pressure of 1600 psid between the RCS and S/G pressures ensures us that the design differential pressure across the S/G tubes does not exceed 1600 psi. This value has been adjusted for harsh environmental conditions.

The following provides some methods of establishing RCS temperature control:

1. A break downstream of the MSIVs can be controlled by closing the MSIVs and using the atmospheric dump valves to control RCS temperature.

2. A blowdown may result in the unaffected steam generator being isolated at a temperature much higher than the minimum RCS temperature. This may result in a significant delay between the time the RCS begins to heatup and the time when the unaffected steam generator becomes an effective heat sink.

Establishing RCS temperature control following an excess steam demand may require selecting an optimal time to begin steaming the least affected steam generator. The following are methods that can be used to establish an optimal time to steam the unaffected steam generator:

- As the CET temperatures begin to rise, start steaming the least affected steam generator.

- As pressurizer pressure starts to rise, start steaming the least affected steam generator.

Step Number 16 Stabilize RCS Temperature (cont)

Contingency Actions

Guidance is provided to stabilize RCS pressure at greater than HPSI shutoff head (1500-1600 psia) if RCS pressure had fallen below HPSI shutoff head.

Justification for Deviations

Waterford includes additional guidance to limit the RCS heatup and repressurization. The small size of the Waterford ADVs does not provide enough steaming capability to limit the RCS heatup, therefore EFW is required to minimize the heatup. Guidance is also provided to stabilize the RCS repressurization which is not found in the EPG. This is necessary to ensure the RCS is maintained within the RCS Pressure Temperature Limits and the design basis of the plant.

References

None

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000054 CE/E06 EK2.1	
	Importance Rating	3.3	

K/A Statement

EK2.01 Knowledge of the interrelations between the (Loss of Feedwater) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO 12 Rev: 0

Given:

- Plant is at 100 percent power
- FWCS 1 and 2 in AUTOMATIC

Which of the following conditions would result in entry to OP-902-006, Loss of Main Feedwater Recovery?

A. Loss of Offsite Power

B. S/G #1 Wide Range Level 96 percent

C. Both Steam Generators 75% NR

D Condenser vacuum 13 in Hg

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: Loss of Offsite Power will cause a loss of both Main Feed Pumps but the diagnostic flow chart will direct the crew to OP-902-003, Loss of Offsite power even though an actual loss of Main Feedwater has occurred.

B. Incorrect: 96% WR level will close the MFIV for only one Steam Generator. The other steam generator is still available. The crew would diagnose to OP-902-000, Standard Post Trip Actions.

C. Incorrect: SG NR 74% is the High Level Override setpoint. This would not result in a loss of main feedwater, but will close the Main Feed Reg Valves and Startup Feed Reg valves. The HLO will reset itself at 69.4% NR.

D. **CORRECT:** 13 in Hg is below the Feedwater Pump low Vacuum trip. Both Main Feed pumps will trip at 14 in Hg. The crew will diagnose into OP-902-006.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)
- OP-902-009 Appendix 1 Revision 309
- OP-902-006 Revision 15
- OP-901-201 Revision 6
- OP-003-033 Revision 313

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE06 obj. 3 (As available)

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
- 55.41 10
- 55.43

Comments:

9.0 AUTOMATIC FUNCTIONS

9.1 Main Feedwater Pump Turbine Trip

9.1.1	FWPT A RECIRC FAILURE	SGFP A(B) discharge pressure <900 PSIG and either of the following: • Flow demand vs. actual suction flow differential ≥562 GPM or • either recirc isolation valve not fully open
9.1.2	FWPT A LUBE OIL PRESS LO (LOF-IPS-3006A1, A2, A3)	10 PSIG
9.1.3	FWPT A OVERSPEED (FW-IST-2010A)	5720 RPM
9.1.4	FWPT A VACUUM LO (FW-IPS-3007A5)	14” Hg vac.
9.1.5	FWPT A SUCTION PRESS LO (CD-IPS-2203A, 2204A, 2205A) (2/3) (Time Delay 10 Seconds)	250 PSIG (R: 280 PSIG)
9.1.6	FWPT A FLOW LO (CD-IDPIS-2207A, 2208A and CD-IFIS-2202A) (2/3)	2700 GPM and SGFP A(B) discharge pressure >900 PSIG (R: 760 PSIG)
9.1.7	FWPT A CD PUMP LOST	A and B FWPT HP governor valves open and A and C Condensate Pumps tripped.
9.1.8	FWPT A THRUST BEARING WEAR (FW-IZD-3016A) (2/2)	Channel A Normal 25 mils Channel A Reverse -10 mils Channel B Normal 25 mils Channel B Reverse -10 mils

9.1.9	FWPT B RECIRC FAILURE	SGFP A(B) discharge pressure <900 PSIG and either of the following: • Flow demand vs. actual suction flow differential ≥562 GPM or • either recirc isolation valve not fully open
9.1.10	FWPT B LUBE OIL PRESS LO (LOF-IPS-3006B1, B2, B3)	10 PSIG
9.1.11	FWPT B OVERSPEED (FW-IST-2010B)	5720 RPM
9.1.12	FWPT B VACUUM LO (FW-IPS-3007B5)	14” Hg vac.
9.1.13	FWPT B SUCTION PRESS LO (CD-IPS-2203B, 2204B, 2205B) (2/3) (Time Delay 30 Seconds)	250 PSIG (R: 280 PSIG)
9.1.14	FWPT B FLOW LO (CD-IDPIS-2207B, 2208B and CD-IPIS-2202B) (2/3)	2700 GPM and SGFP A(B) discharge pressure >900 PSIG (R: 760 PSIG)
9.1.15	FWPT B CD PUMP LOST	A and B FWPT HP governor valves open and B and A, or C Condensate Pumps tripped.
9.1.16	FWPT B THRUST BEARING WEAR (FW-IZD-3016B) (2/2)	Channel A Normal 25 mils Channel A Reverse -10 mils Channel B Normal 25 mils Channel B Reverse -10 mils

9.2 Main Feedwater Pump Turbine Speed lowers to “0”

9.2.1	SSPU A FAILURE (TGB +40 MUX TB4005 ONLY)	Both speed sensors fail, or loss of power to cabinet (4 second time delay)
9.2.2	SSPU B FAILURE (TGB +40 MUX TB4005 ONLY)	Both speed sensors fail, or loss of power to cabinet (4 second time delay)

C AUTOMATIC ACTIONS

1. Steam Generator Control Channel level deviation of 7% will shift following controllers to MAN for the affected Steam Generator:

   a. For Steam Generator 1:

      - FW IHIC1107, Main Feedwater Pump A Speed Controller
      - FW IHIC1111, Main Feedwater Regulating Valve A Controller
      - FW IHIC1105, Startup Feedwater Regulating Valve A Controller

   b. For Steam Generator 2:

      - FW IHIC1108, Main Feedwater Pump B Speed Controller
      - FW IHIC1121, Main Feedwater Regulating Valve B Controller
      - FW IHIC1106, Startup Feedwater Regulating Valve B Controller

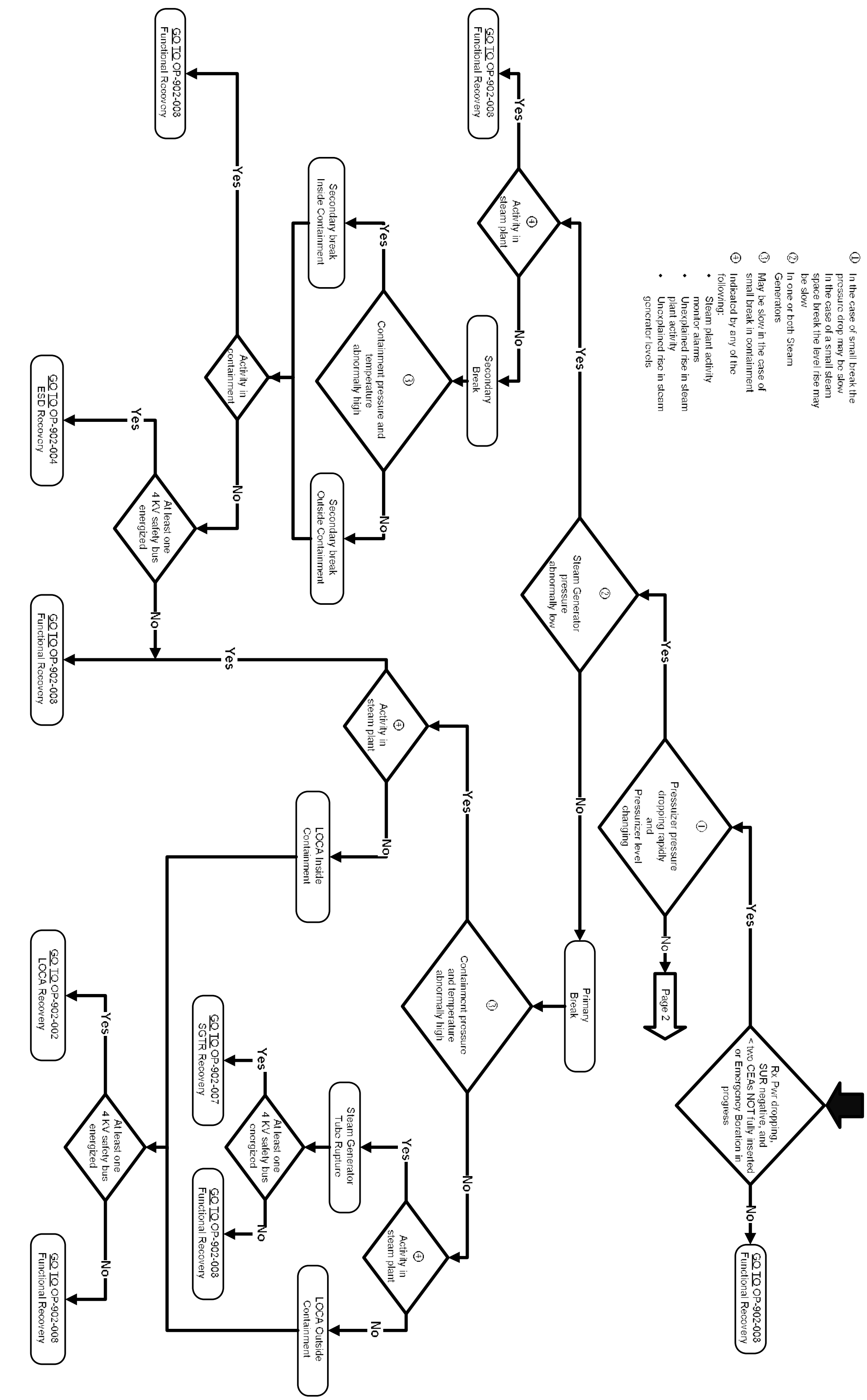
2. A Reactor trip will be generated for either of the following:
   - Steam Generator level at less than or equal to 27.4% Narrow Range on two out of four Reactor Protection System Channels
   - Steam Generator level at greater than or equal to 87.7% Narrow Range on two out of four Reactor Protection System Channels

3. At greater than or equal to 74% Narrow Range the following valves for the affected Steam Generator will close due to a Steam Generator High Level Override:
   - Main Feedwater Regulating Valve
   - Startup Feedwater Regulating Valve

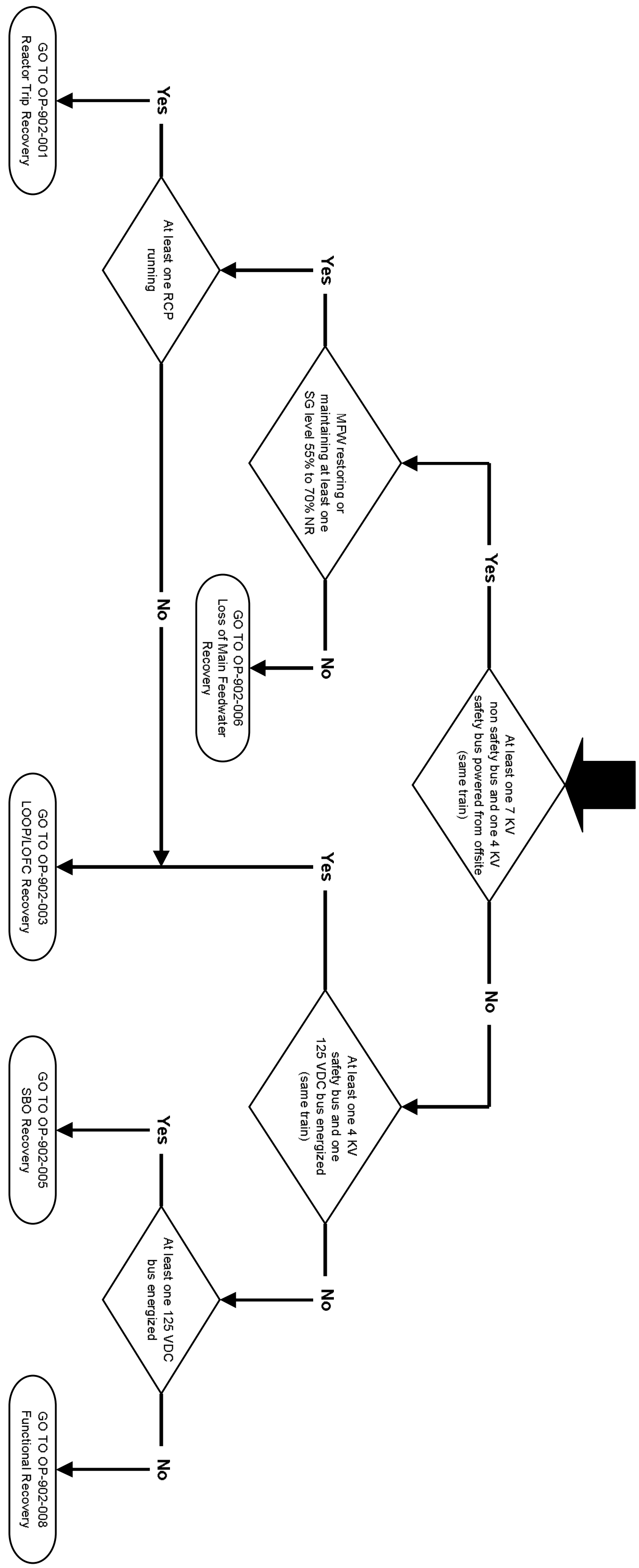
4. At greater than or equal to 96 % Wide Range Steam Generator level <u>and</u> greater than or equal to 750 PSIA Steam Generator pressure a Steam Generator High Level Isolation will close the associated Main Feedwater Isolation Valve.

5. SGFP controllers will automatically go to minimum speed on a Reactor Trip with the controller in Automatic or Manual Mode of Operation.

Diagnostic Flow Chart



Diagnostic Flow Chart (Cont'd)



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000055 EK1.01	
	Importance Rating	3.3	

K/A Statement

EK1.01 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Effect of battery discharge rates on capacity.

Proposed Question: RO 13 Rev: 0

Given:

- At 1800, a reactor trip occurred due to a Station Blackout
- At 1810, the crew entered OP-902-005, Station Blackout Recovery
- At 1815, the CRS directs the ATC operator to reduce battery loads

Which of the following describes the clock time limit to accomplish these actions and the reason for those actions?

A. 1845; meet two hour coping time for 125 VDC buses to carry vital loads.

B. 1830; meet two hour coping time for125 VDC buses to carry vital loads.

C. 1845; meet four hour coping time for 125 VDC buses to carry vital loads.

D. 1830; meet four hour coping time for 125 VDC buses to carry vital loads.

Proposed Answer: D

Explanation: (Optional)

A. Incorrect. Time is incorrect. 30 minutes is from the time of the SBO occurrence. Coping time is four hours. Two hours is TS 3.8.2.1 time limit for restoration of an inoperable battery bank.
B. Incorrect. Time is correct. Coping time is four hours. Two hours is TS 3.8.2.1 time limit for restoration of an inoperable battery bank.
C. Incorrect. Time is incorrect. 30 minutes is from the time of the SBO occurrence. Coping time is correct. The station blackout coping time analysis is performed in EC-E89-016, which is 4 hours.
D. **CORRECT:** Time is correct. 30 minutes is from the time of the SBO occurrence. The station blackout coping time analysis is performed in EC-E89-016, which is 4 hours.

Technical Reference(s): OP-902-005 revision 17
TG-OP-902-005 step 15 revision 306 and in the introduction (page 1)
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE05 obj. 2 (As available)

Question Source: Bank # 08313
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2008 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Portable emergency lighting is available in Appendix R lockers located at LCP-43, Remote Shutdown Panel and +35 RAB Relay Room.

Reduce Battery Loads

* 15. **IF** AC power is **NOT** expected to be restored within 30 minutes, **THEN** perform the following to reduce unnecessary station loads:

- REFER TO Appendix 7, "Station Blackout Actions", and deenergize plant equipment.
- Open the following breakers on LTN-EPNL-322: (behind CP-8)
  - LTN-EBKR-322-1
  - LTN-EBKR-322-2
  - LTN-EBKR-322-3
- Open the following breakers on LTN-EPNL-323: (near main Control Room entrance)
  - LTN-EBKR-323-2
  - LTN-EBKR-323-3
  - LTN-EBKR-323-4

Introduction

The bases section of the Station Blackout (SBO) Recovery procedure describes the SBO transient in relation to the actions the operator takes during a SBO. The purpose of the bases section is to provide the operators with information which will enable them to understand the reasons for, and the consequences of, the actions they take during a SBO. This Technical Guide should be used in conjunction with other EOP support documents to provide a better understanding of the EOPs.

The Station Blackout analysis is performed in calculation EC-E89-016, Station Blackout Response. The coping period for a Station Blackout at Waterford 3 is 4 hours.

Characterization of a Station Blackout

A Station Blackout event is an interruption of electrical power to the plant's electrical distribution system which results in a reactor trip and a concurrent loss of all vital AC power. The Station Blackout has also been referred to as a loss of all AC power, or a loss of off-site power with a concurrent loss of on-site power.

Natural circulation and heat transfer from primary to secondary via the steam generators is the preferred method of RCS and core heat removal for this event. The natural circulation capability is the primary means of core cooling since the RCPs are unavailable to provide forced circulation and the shutdown cooling system is unavailable to provide RCS heat removal.

Step Number 15 Reduce Battery Loads

Objective

The intent of this step is to strip unnecessary loads off the DC buses to conserve battery capacity.

Instructions

During a station blackout event, the DC loads will be supplied by the station batteries. The battery chargers will not be available to maintain the batteries charged to capacity. Therefore, until power is restored, unnecessary loads should be stripped off the DC buses to conserve DC power.

The operator is referred to a Standard Appendix to strip unnecessary DC loads outside the Control Room.

The breakers located in the Control Room are listed in the step.

Contingency Actions

None

Justification for Deviations

Waterford reverses the EPG steps 10 and 11. EPG step 10 is to strip unnecessary DC loads to conserve battery power and EPG step 11 is to provide supplemental cooling to the appropriate equipment. Both steps are performed within the first 30 minutes of the SBO. The electrical breakers opened in EOP step 14 provide lighting to areas behind the control panels. After the control room cabinets doors are opened there are no further actions to take in the back of the control room. The EPG steps 10 and 11 are reversed to allow the control panel doors to be opened while lighting is still available to the back of the control room.

References

1. PEIR 10682, Supplement
2. W3F192-0015, Response to NRC SER on Station Blackout
3. W3F192-0079, Additional Information Regarding Station Blackout (SBO)

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery Bank No. 3A-S and one associated full capacity charger (3A1-S or 3A2-S).

b. 125-volt Battery Bank No. 3B-S and one associated full capacity charger (3B1-S or 3B2-S).

c. 125-volt battery Bank No. 3AB-S and one associated full capacity charger (3AB1-S or 3AB2-S).

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

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and at least one associated charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 125 volts on float charge.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000056 AA2.40	
	Importance Rating	3.3	

K/A Statement

AA2.40 Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Service water pump ammeter and flowmeter

Proposed Question: RO 14 Rev: 0

Given:

- A Loss of Offsite power has occurred
- Crew has entered OP-902-003, Loss of Offsite Power/Loss of Forced Circulation Recovery
- SUPS MA was lost during the event

Auxiliary Component Cooling Water (ACCW) pumps will ____(1)_____ . CC HX A ACC Outlet TCV (ACC-126A) must be manually throttled because ACCW flow is reading ____(2)______ gpm at CP-33.

	(1)	(2)
A.	start	0
B.	start	> 4000
C.	require a manual start	0
D.	require a manual start	> 4000

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: ACCW Pumps start on the sequencer and is unaffected by the loss of SUPS MA, however ACC-126A fails open on the loss of SUPS MA and requires manual control.
B. **CORRECT**: ACCW Pumps start on the sequencer and is unaffected by the loss of SUPS MA, however ACC-126A fails open on the loss of SUPS MA and requires manual control.
C. Incorrect: ACCW Pumps start on the sequencer and is unaffected by the loss of SUPS MA, however ACC-126A fails open on the loss of SUPS MA and requires manual control.
D. Incorrect: ACCW Pumps start on the sequencer and is unaffected by the loss of SUPS MA, however ACC-126A fails open on the loss of SUPS MA and requires manual control.

Technical Reference(s): OP-901-312 revision 306
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO30 obj. 3 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

C. AUTOMATIC ACTIONS

1. Emergency Diesel Generator A:

    a. Starts in Emergency Mode.

    b. Emergency Diesel Generator A Output breaker closes.

    c. Sequencer begins loading A Safety Bus.

2. Sequencer Loading:

    a. The SST 32A Feeder is closed by the 0.5 second load block.

    b. The SST 315A Feeder is closed by the 1.0 second load block.

    c. Component Cooling Water Pump(s) is started on the 7.0 second load block.

    d. Auxiliary Component Cooling Water Pump(s) is started on the 17.0 second load block.

    e. Operation of Dry Cooling Fans will be restored by the 41.0 second load block.

    f. The following breakers do not receive an undervoltage trip and will energize their associated busses when Emergency Diesel Generator A Output Breaker closes:

        - A3 TIE TO AB3
        - SST 31A FEEDER
        - A31 TIE TO AB31
        - MCC 311A SUPPLY (SSD-EBKR31A-7C)
        - MCC 312A SUPPLY (SSD-EBKR31A-6C)
        - MCC 313A SUPPLY (SSD-EBKR31A-6A)
        - MCC 314A SUPPLY (SSD-EBKR31A-5A)
        - MCC 317A SUPPLY (SSD-EBKR31A-7A)

E_1 LOSS OF SUPS MA

1. Monitor Primary Plant parameters on Safety Channels B, C, and D.

2. Perform OP-901-501, PMC or Core Operating Limit Supervisory System Inoperable, concurrently with this procedure.

3. Select ALTERNATE on Startup Channel 2, if Startup Channel 2 energized on the loss of SUPS MA.

4. To monitor Component Cooling Water Train A temperature, have PMI Technician connect test instrument capable of reading a Type E thermocouple to VLL Card 422 (CC-ITE-7075A) or VLL Card 423 (CC-ITE-7076A) in CP-48.

NOTE

CC HX A ACC Outlet TCV (ACC-126A) fails open on loss of power.

5. If Auxiliary Component Cooling Water Pump A is running, then perform the following:
   - Manually throttle CC HX A ACC Outlet TCV (ACC-126A) to obtain desired flow rate and to control Component Cooling Water temperature
   - Manually throttle Wet Cooling Tower Basin Manual M/U valve (CMU- 410A) as required to maintain Wet Cooling Tower basin level.

6. Maintain Component Cooling Water Train A temperature by using manual control of the following:
   - Dry Cooling Tower Train A Fans
   - Wet Cooling Tower Train A Fans

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000057 AK3.01	
	Importance Rating	4.1	

K/A Statement

AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus

Proposed Question: RO 15 Rev: 0

Given:

- Reactor has tripped due to a Grid Disturbance
- SUPS MA is de-energized
- The BOP is verifying Station Loads are energized per OP-902-000, Standard Post Trip Actions

The minimum Vital AC instrument channel(s) required for electrical train A is ___(1)_____. The vital instrument channel(s) ensures ________(2)_______ is available.

	(1)	(2)
A.	not met	control power for remote operation of breakers
B.	not met	monitoring and limited control of other safety functions
C.	met	control power for remote operation of breakers
D.	met	monitoring and limited control of other safety functions

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: OP-902-000 requires one channel of vital AC instrument bus to be to be energized per train. 125 VDC is used for control power for remote operation of breakers.
B. Incorrect: OP-902-000 requires one channel of vital AC instrument bus to be to be energized per train. 125 VDC is used for control power for remote operation of breakers. Vital 120 VAC is used for monitoring and limited control when addressing other safety functions.
C. Incorrect: OP-902-000 requires one channel of vital AC instrument bus to be to be energized per train. 125 VDC is used for control power for remote operation of breakers.
D. **CORRECT:** OP-902-000 requires one channel of vital AC instrument bus to be energized per train. 125 VDC is used for control power for remote operation of breakers. 120 VAC is used for monitoring and limited control when addressing other safety functions.

Technical Reference(s): TG-OP-902-008 page 46 Revision 308
(Attach if not previously provided) TG-OP-902-008 page 82 Revision 308
(including version/revision number) OP-902-000 Revision 15

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 9 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS
2. (continued)	
___ c. Check station loads are energized from offsite electrical power as follows: **Train A** • A1, 6.9 KV non safety bus • A2, 4.16 KV non safety bus • A3, 4.16 KV safety bus • A-DC electrical bus • A or C vital AC Instrument Channel **Train B** • B1, 6.9 KV non safety bus • B2, 4.16 KV non safety bus • B3, 4.16 KV safety bus • B-DC electrical bus • B or D vital AC Instrument Channel	c.1 **IF ANY** 4.16 KV safety bus is **NOT** powered from offsite, **THEN:** 1) Verify associated EDG has started **AND** EDG output breaker closed. 2) **IF** EDG output breaker is **NOT** closed **THEN**: • Verify stable EDG Voltage 3920 - 4350 AC Volts. • Verify 3-2 Breaker open. • Check Sequencer LOCKOUT **NOT** illuminated. 3) **IF** EDG output breaker is **NOT** closed **AND** Step c.1.2) is met **THEN** locally close EDG output breaker. 4) Verify CCW cooling available to EDG.

SFSC 2 - Maintenance of Vital Auxiliaries (AC and DC Power) (cont)

MVA-AC-1: Startup Transformer

Objective

The intent of the Maintenance of Vital Auxiliaries safety function status check is to check that electrical power is available to support and maintain control of all other safety functions. This safety function checks that at least one 4.16 safety KV bus is powered from the startup transformers (offsite power).

Criteria

Condition 1 & 2

Electrical power is essential to the fulfillment of succeeding safety functions. In general, the normal post-trip electrical power requirements needed to maintain all other safety functions are:

1) at least one 4.16 safety KV bus
2) at least one 125 VDC safety bus
3) at least one 120 VAC safety instrument bus

The above electrical busses must all be powered from the same side of the electrical distribution system to ensure the minimum electrical requirements are met. In the case of the 4.16 KV AC bus, in addition to having the proper voltage on the bus there must also be control power available to ensure auto/remote operation of the breakers. This requires that the DC bus and the 4.16 KV bus be part of the same safety train.

Items 2 and 3 are checked in the safety function MVA-DC-1, Battery Chargers/Station Batteries.

Justification for Deviations

There are no deviations

References

None

Safety Function: Maintenance of Vital Auxiliaries (AC and DC Power)
Success Path: MVA-DC-1: Battery Chargers/Station Batteries

Step Number 2 Check ONE AC Instrument Bus Energized

Objective

The intent of this step is to ensure that at least one 120 VAC instrument bus is energized from the same train as the energized 125 VDC bus. This is the minimum power required to provide monitoring and limited control of the other safety functions.

Instructions

The operator checks that at least one 120 VAC instrument bus is energized from the same train as the energized 125 VDC bus using control room indications.

Contingency Actions

The Contingency Action directs the operator to energize a 120 VAC instrument bus in the same train as the energized 125 VDC bus.

Justification for Deviations

There are no deviations

References

None

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000026 AK3.03	
	Importance Rating	4.0	

K/A Statement

AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Guidance actions contained in the EOP for Loss of CCW

Proposed Question: RO 16 Rev: 0

Given:

- An Excess Steam Demand Event has occurred
- At time 1630, a Containment Spray Actuation Signal has been initiated.

Reactor Coolant Pumps must be secured no later than time ___(1)_____.
Reactor Coolant Pumps (RCPs) are secured to prevent _______(2)______ .

	(1)	(2)
A.	1640	RCP seal failure
B.	1640	damage to the RCP bearings
C.	1633	RCP seal failure
D.	1633	damage to the RCP bearings

Proposed Answer: C

Explanation: (Optional)

A. Incorrect. Per TGOP-902-004 step 10, RCPs must be secured within 3 minutes of a loss of CCW to prevent RCP seal failure. 10 minutes, as mentioned in OP-901-510 Section E1, CAUTION 1 is the time that if exceeded could cause damage to RCP Seals if CCW flow is then restored to an affected RCP

B. Incorrect. Per TGOP-902-004 step 10, RCPs must be secured within 3 minutes of a loss of CCW to prevent RCP seal failure. 10 minutes, as mentioned in OP-901-510 Section E1, CAUTION 1 is the time that if exceeded could cause damage to RCP Seals if CCW flow is then restored to an affected RCP. RCP lower and upper bearings are cooled by CCW.

C. **CORRECT:** Per TGOP-902-004 step 10, RCPs must be secured within 3 minutes of a loss of CCW to prevent RCP seal failure

D. Incorrect. Per TGOP-902-004 step 10, RCPs must be secured within 3 minutes of a loss of CCW to prevent RCP seal failure. RCP lower and upper bearings are cooled by CCW.

Technical Reference(s): OP-902-004 step 10 Revision 14
(Attach if not previously provided) TGOP-902-004 page 24 revision 305
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE04 obj. 8 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS
Verify MSIS Actuation	
8. Verify MSIS Actuation.	
RCP Trip Strategy	
* 9. **IF** pressurizer pressure is less than 1621 psia, **AND** SIAS is actuated, **THEN**:	
a. Verify no more than two RCPs are operating.	
b. **IF** pressurizer pressure is less than the minimum RCP NPSH of Appendix 2A-D, "RCS Pressure and Temperature Limits", **THEN** stop **ALL** RCPs.	
Verify RCP Operating Limits	
* 10. **IF** RCPs are operating, **THEN**:	
a. Verify CCW available to RCPs.	a.1 **IF** CCW is lost to RCPs **AND** is **NOT** restored within 3 minutes, **THEN** stop the affected pumps.
b. **IF** a CSAS is initiated, **THEN** stop **ALL** RCPs.	
c. **IF** RCS T_C is less than 382ºF **[384ºF]**, **THEN** verify no more than two RCPs are operating.	

Step Number 10 Verify RCP Operating Limits

Objective

The intent of this step is to verify that RCPs are operating within their acceptable operating conditions.

It is important to verify RCP operating limits in the EOPs given the off-normal conditions of the RCS so that any RCP that is not operating within their operating limits is stopped prior to being damaged.

Instructions

The operator is expected to verify CCW flow to the RCPs. The CCW system provides cooling for the RCPs. Without proper cooling the likelihood of a RCP seal failure is increased.

A substep is included to secure all RCPs if a containment pressure has reached the CSAS setpoint. This step is performed for protection of the RCPs, since CCW, which provides cooling to the RCPs, is isolated upon CSAS actuation.

A plant specific step has been developed to verify no more than two RCPs are operating when Tcold is less than 382°F [384°F]. The RCS PT Limit Curve specifies that one RCP should be tripped when Tcold is less than 382°F [384°F] to prevent a core uplift problem.

Contingency Actions

The RCPs can be operated for approximately 3 minutes without CCW. The operator should secure the RCPs if CCW cannot be restored within 3 minutes.

Justification for Deviations

Waterford divides the EPG step into substeps. The substeps are concerned with RCP operating limits. Waterford meets the intent of the EPG, but since a greater level of detail is required the additional information is included in substeps.

References

1. ER-W3-97-0492-00-00 - RCP Operation in EOPs
2. EC-S98-001 T.24

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 Two independent trains of ultimate heat sink (UHS) cooling towers shall be OPERABLE with each train consisting of a dry cooling tower (DCT) and a wet mechanical draft cooling tower (WCT) and its associated water basin with:

a. A minimum water level in each wet tower basin of 97% (-9.86 ft MSL)

b. An average basin water temperature of less than or equal to 89°F.

c. Fans as required by Table 3.7-3.

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

ACTION:

a. With 1 UHS train inoperable, restore the inoperable train to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both UHS trains inoperable, restore at least one UHS train to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With a Tornado Watch in effect, all 9 DCT fans under the missile protected portion of the DCT shall be OPERABLE. If the number of fans OPERABLE is less than required, restore the inoperable fan(s) to OPERABLE status within 1 hour, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With any UHS fan inoperable, determine the outside ambient temperature at least once every 2 hours and verify that the minimum fan requirements of Table 3.7-3 are satisfied (required only if the associated UHS is OPERABLE).

SURVEILLANCE REQUIREMENTS

4.7.4. Each train of UHS shall be determined OPERABLE:

a. At least once per 24 hours by verifying the average water temperature and water level to be within specified limits.

b. At least once per 31 days, by verifying that each wet tower and dry tower fan that is not already running, starts and operates for at least 15 minutes.

TABLE 3.7-3

ULTIMATE HEAT SINK MINIMUM FAN REQUIREMENTS PER TRAIN

DRY COOLING TOWER

AMBIENT CONDITION	DRY BULB $\geq$ 97°F	< 97°F DRY BULB $\geq$ 91°F	< 91°F DRY BULB
Fan Requirements[(1)]	15	14*	12*

WET COOLING TOWER

Fan Requirements - 8

[(1)] With any of the above required DCT Fans inoperable comply with ACTION d.

[redacted]

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000062 2.2.36	
	Importance Rating	3.1	

K/A Statement

2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question: RO 17 Rev: 0

Given:

- Plant is at 100% power
- DCT Fan 8A is inoperable due to maintenance
- Outside air temperature is 94 °F
- A tornado watch has gone into effect for St. Charles Parish.

What is the most limiting time to restore Dry Cooling Tower fan 8A to comply with Tech Spec 3.7.4 (if any)?

A. 72 hours

B. 2 hours

C. 1 hour

D. no time requirement

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: DCT Fan 8A is under the missile shield. With a tornado watch in effect all fans under the missile shield must be operable. If not, the fans must be restored within 1 hour IAW 3.7.4 action c. The applicant could select 72 hours if the determination is made that one UHS is inoperable and TS 3.7.4 action a is entered.
B. Incorrect: DCT Fan 8A is under the missile shield. With a tornado watch in effect all fans under the missile shield must be operable. If not, the fans must be restored within 1 hour IAW 3.7.4 action c. The 2 hour limitation per TS 3.7.4 action d is performed but is not a time limitation for restoring a DCT fan.
C. **CORRECT:** DCT Fan 8A is under the missile shield. With a tornado watch in effect all fans under the missile shield must be operable. If not, the fans must be restored within 1 hour IAW 3.7.4 action c.
D. Incorrect: No time limitation is plausible if the applicant determines that DCT Fan 8A is not under the missile shield.

Technical Reference(s): TS 3.7.4
(Attach if not previously provided) SD-CC page 19
(including version/revision number)

Proposed references to be provided to applicants during examination: TS 3.7.4

Learning Objective: WLP-OPS-CC00 obj. 9 (As available)

Question Source:
Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 8
55.43 ____

Comments:

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 Two independent trains of ultimate heat sink (UHS) cooling towers shall be OPERABLE with each train consisting of a dry cooling tower (DCT) and a wet mechanical draft cooling tower (WCT) and its associated water basin with:

a. A minimum water level in each wet tower basin of 97% (-9.86 ft MSL)

b. An average basin water temperature of less than or equal to 89°F.

c. Fans as required by Table 3.7-3.

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

ACTION:

a. With 1 UHS train inoperable, restore the inoperable train to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both UHS trains inoperable, restore at least one UHS train to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With a Tornado Watch in effect, all 9 DCT fans under the missile protected portion of the DCT shall be OPERABLE. If the number of fans OPERABLE is less than required, restore the inoperable fan(s) to OPERABLE status within 1 hour, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With any UHS fan inoperable, determine the outside ambient temperature at least once every 2 hours and verify that the minimum fan requirements of Table 3.7-3 are satisfied (required only if the associated UHS is OPERABLE).

SURVEILLANCE REQUIREMENTS

4.7.4. Each train of UHS shall be determined OPERABLE:

a. At least once per 24 hours by verifying the average water temperature and water level to be within specified limits.

b. At least once per 31 days, by verifying that each wet tower and dry tower fan that is not already running, starts and operates for at least 15 minutes.

TABLE 3.7-3

ULTIMATE HEAT SINK MINIMUM FAN REQUIREMENTS PER TRAIN

DRY COOLING TOWER

AMBIENT CONDITION	DRY BULB ≥ 97°F	< 97°F DRY BULB ≥ 91°F	< 91°F DRY BULB
Fan Requirements[(1)]	15	14*	12*

WET COOLING TOWER

Fan Requirements - 8

[(1)] With any of the above required DCT Fans inoperable comply with ACTION d.

[redacted]

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000065 AA1.01	
	Importance Rating	2.7	

K/A Statement

AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Remote manual loaders

Proposed Question: RO 18 Rev: 0

Given:

- An extended loss of Offsite power has occurred
- The crew is performing actions in OP-902-003, Loss of Offsite Power/Loss of Forced Circulation Recovery
- Instrument and Station Air Compressors are unavailable
- The crew is performing actions in OP-901-511, Instrument Air Malfunction

To maintain ____(1)____ , Essential Air is aligned in accordance with the appropriate _____(2)_____.

	(1)	(2)
A.	containment integrity	Emergency Operating procedure
B.	desired EFW flow	Offnormal Operating procedure
C.	desired EFW flow	Emergency Operating procedure
D.	containment integrity	Offnormal Operating procedure

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: Essential Air is aligned per OP-901-511 after 10 hours of a Loss of Instrument Air. Essential Air maintains containment isolation valves that fail open in the closed position. The EOPs do not mention Essential air.
B. Incorrect: OP-901-511 does require the crew to take actions on EFW valves to establish desired flow within 10 hours, but this is not part of the Essential Air system. The EOPs do not mention Essential air.
C. Incorrect: OP-901-511 does require the crew to take actions on EFW valves to establish desired flow within 10 hours, but this is not part of the Essential Air system. The EOPs do not mention Essential air.
D. **CORRECT:** Essential Air is aligned per OP-901-511 after 10 hours of a Loss of Instrument Air. Essential Air maintains containment isolation valves that fail open in the closed position. The EOPs do not mention Essential air.

Technical Reference(s): OP-901-511 revision 14
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO50 obj. 3 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 10
55.43

E_0 GENERAL (CONT'D)

NOTE

(1) Normal FHB Ventilation will secure due to low air pressure to the associated dampers. Upon restoration of Instrument Air, FHB Ventilation will automatically restart.

(2) Essential Chiller Coolant Select Valves, ACC-112A(B), ACC-139A(B), CC-301A(B), and CC-322A(B) fail *as is* after Nitrogen Accumulator I (II) is exhausted approximately 10 hours following a loss of Instrument Air. The timing and sequence for manipulating these valves is important for protecting Essential Chiller cooling, ACCW WCT Basins A(B), and preventing water hammer. Nitrogen Accumulators I and II are equipped with Backup Air Supplies to recharge the accumulators several days into an event. If Nitrogen Accumulators I and/or II are recharged with Backup Air, then repositioning Essential Chiller Select Valves from Wet Tower to Dry Tower Mode to conserve WCT Basin Inventory must be completed within 30 minutes. The valve swap should be performed when CCW temperature can be maintained <110 degrees F using only the Dry Cooling Tower. **[EC-41095, EC-41355]**

(3) The valves EFW-223A(B), EFW-224A(B), EFW-228A(B), EFW-229A(B), MS-116A(B), CC-114A(B), CC-115A(B), CC-126A(B), CC-127A(B), CC-200A(B), CC-727, CC-563 and ACC-126A(B) are evaluated to have a mission time of 10 hours on a loss of Instrument Air with no additional charging of their associated accumulator. If the loss of Instrument Air is expected to last longer than 10 hours, actions should be taken prior to valves repositioning. **[EC-41355]**

Step	PLACEKEEPER START	DONE	N/A
18. If the loss of Instrument Air is expected to last longer than 10 hours, then perform the following actions within 10 hours: **[EC-41355]**	☐	☐	☐
18.1 If EFW is in service, then establish desired flow as follows:	☐	☐	☐
18.1.1 For the EFW Flow Isolation Valves, EFW-228A(B) and EFW-229A(B), either gag closed or fail open by isolating motive air.	☐	☐	
18.1.2 For the EFW Flow Isolation Valves, EFW-223A(B) and EFW-224A(B), gag to the appropriate position and isolate motive air.	☐	☐	

ATTACHMENT 3: AIR OPERATED VALVES THAT FAIL OPPOSITE THEIR ESF POSITION (CONT'D)

COMP #	VALVE	FAIL POSITION AIR/POWER	NOTES	SIGNAL	ACTUATED POSITION
CC-115B	CCW PUMP AB TO B SUCT CROSSCONNECT	OPEN	1, 6	SIAS	CLOSED
CC-126A	CCW PUMP A TO AB DISCH CROSSCONNECT	OPEN	1, 6	SIAS	CLOSED
CC-126B	CCW PUMP B TO AB DISCH CROSSCONNECT	OPEN	1, 6	SIAS	CLOSED
CC-127A	CCW PUMP AB TO A DISCH CROSSCONNECT	OPEN	1, 6	SIAS	CLOSED
CC-127B	CCW PUMP AB TO B DISCH CROSSCONNECT	OPEN	1, 6	SIAS	CLOSED
CC-641	CCW TO CONTAINMENT OUTSIDE CNTMT ISOL	OPEN	2, 3	CSAS	CLOSED
CC-710	CNTMT CCW RETURN HDR INSIDE CNTMT ISOL	OPEN	2, 3	CSAS	CLOSED
CC-713	CNTMT CCW RETURN HDR OUTSIDE CNTMT ISOL	OPEN	2, 3	CSAS	CLOSED
EFW-223A	EFW HDR A TO SG 1 BACKUP FLOW CONTRL	OPEN	1, 4, 7	MSIS	CLOSED
				EFAS	VARIES
EFW-223B	EFW HDR B TO SG 2 BACKUP FLOW CONTRL	OPEN	1, 4, 7	MSIS	CLOSED
				EFAS	VARIES
EFW-224A	EFW HDR A TO SG 1 PRIMARY FLOW CONTRL	OPEN	1, 4, 7	MSIS	CLOSED
				EFAS	VARIES
EFW-224B	EFW HDR B TO SG 2 PRIMARY FLOW CONTRL	OPEN	1, 4, 7	MSIS	CLOSED
				EFAS	VARIES

ATTACHMENT 3: AIR OPERATED VALVES THAT FAIL OPPOSITE THEIR ESF POSITION (CONT'D)

COMP #	VALVE	FAIL POSITION AIR/POWER	NOTES	SIGNAL	ACTUATED POSITION
FW-173A	SG1 MAIN FEEDWATER REG	AS IS	2	MSIS	CLOSED
FW-173B	SG2 MAIN FEEDWATER REG	AS IS	2	MSIS	CLOSED
FW-184A	MAIN FEEDWATER ISOLATION 1	AS IS	2	MSIS	CLOSED
FW-184B	MAIN FEEDWATER ISOLATION 2	AS IS	2	MSIS	CLOSED
MS-116A	STEAM GENERATOR 1 ATMOSPHERE DUMP VLV	CLOSED	1, 4, 7	>992.7°F MS	VARIES
MS-116B	STEAM GENERATOR 2 ATMOSPHERE DUMP VLV	CLOSED	1, 4, 7	>992.7°F MS	VARIES
SI-106A	RWSP OUTL HDR A ISOL	AS IS	1	SIAS	OPEN
SI-106B	RWSP OUTL HDR B ISOL	AS IS	1	SIAS	OPEN

NOTE 1 Valve is equipped with a Nitrogen Accumulator for extended operation following a loss of Instrument Air pressure.

NOTE 2 Valve is equipped with an Air Accumulator for extended operation following a loss of Instrument Air pressure.

NOTE 3 Containment isolation valve equipped with backup 30 day essential air supply.

NOTE 4 Manual Local Handwheel Operation or Accumulator Recharge Required after 10 hours following loss of Instrument Air.

NOTE 5 Recharging Nitrogen Accumulators I and II with installed Backup Air Supplies in accordance with Attachment 11, Recharging N2 Accumulators I and II with Backup Air, allows 30 minutes for repositioning Essential Chiller Coolant Select Valves from Wet Tower to Dry Tower Mode to conserve WCT Basin Inventory.

NOTE 6 If this Component Cooling Water Cross-Connect Valve is required to be closed for train separation, then manual handwheel jackscrew must be positioned to gag valve in the current position prior to exhausting accumulator (within ~10 hours following loss of Instrument Air).

NOTE 7 Valve position will vary as required to maintain setpoint temperature.

ATTACHMENT 9: CONTAINMENT ISOLATION VALVES SUPPLIED WITH ESSENTIAL AIR

Valve #	Valve Description	Accumulator Station
CC-823A,	Containment Fan Cooler C CCW Outlet Isolation	I
CC-807A,	Containment Fan Cooler C CCW Inlet Isolation	
CS-125A,	Containment Spray Hdr A Isolation	
CC-641,	CCW To Containment Outside Containment Isol	II
CC-808B,	Containment Fan Cooler D CCW Inlet Isolation	
CC-822B,	Containment Fan Cooler D CCW Outlet Isolation	
CC-713,	Cntmt CCW Return Header Outside Cntmt Isolation	
CC-710 *,	Cntmt CCW Return Header Inside Cntmt Isolation	III
CC-808A,	Containment Fan Cooler A CCW Inlet Isolation	
CC-822A,	Containment Fan Cooler A CCW Outlet Isolation	
CVC-209,	Charging Header Isolation	
SI-405A *,	RC Loop 2 SDC Suction Inside Containment Isolation	
SI-405B *,	RC Loop 1 SDC Suction Inside Containment Isolation	
CC-807B,	Containment Fan Cooler B CCW Inlet Isolation	IV
CC-823B,	Containment Fan Cooler B CCW Outlet Isolation	
CS-125B,	Containment Spray Hdr B Isolation	

* Note: Key operated essential air supply switch on CP-8

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000003 AA1.06	
	Importance Rating	4.0	

K/A Statement

AA1.06 Ability to operate and / or monitor the following as they apply to the Dropped Control Rod: RCS pressure and temperature

Proposed Question: RO 19 Rev: 0

Given:

- Plant is at 100% power
- All CEAs are fully withdrawn
- ONE (1) Reg Group 4 CEA inserts 25 inches into the core
- The crew enters OP-901-102, CEA or CEDMCS Malfunctions

The crew will match Tave and Tref using turbine load and ____(1)_____. A power reduction must be started within ____(2)_____ minutes of the CEA misalignment.

	(1)	(2)
A.	CEA manipulations	30
B.	RCS boron changes	30
C.	RCS boron changes	15
D.	CEA manipulations	15

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: OP-901-102 directs the crew to match Tave and Tref using turbine load or RCS boron changes (not control rods). 30 minutes is plausible because the applicant may determine that he has an additional 15 minutes to restore the rod within 7 inches of its group. There is no option if the rod is misaligned by greater than 19 inches
B. Incorrect: OP-901-102 directs the crew to match Tave and Tref using turbine load or RCS boron changes. 30 minutes is plausible because the applicant may determine that he has an additional 15 minutes to restore the rod within 7 inches of its group. There is no option if the rod is misaligned by greater than 19 inches.
C. **CORRECT:** OP-901-102 directs the crew to match Tave and Tref using turbine load or RCS boron changes. The crew must start a power reduction within 15 minutes if a CEA is misaligned from its group by greater than 7 inches.
D. Incorrect: Match Tave and Tref using turbine load or RCS boron changes (not control rods).

Technical Reference(s): OP-901-102, CEA or CEDMCS Malfunction, Revision 301
(Attach if not previously provided) TS 3.1.3.1
(including version/revision number) COLR Figure 3

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PP01 obj. 3 (As available)

Question Source: Bank # X
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2007 RO makeup exam

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

	PLACEKEEPER		
	START	DONE	N/A
1. Place CEDMCS Mode Select switch to OFF.		☐	
2. If any of the following occur, then manually trip the Reactor and go to OP-902-000, Standard Post Trip Actions: • Mode 1, ≤10 % Power, and one or more Control Element Assemblies drop • Mode 1, ≤10 % Power, and any Control Element Assemblies are misaligned by >19 inches • Mode 2 and one or more Control Element Assemblies drop • Mode 2 and any Control Element Assemblies are misaligned by >19 inches	☐	☐	☐
3. If Control Element Assembly is misaligned >7 inches, then go to section E_1, CEA Misalignment Greater Than 7 Inches.			
4. If unable to move Control Element Assembly, then go to section E_2, Immovable CEA.	☐	☐	
5. If continuous motion of Control Element Assembly group occurs, then go to section E_3, Continuous Movement of CEA Group.	☐	☐	
6. If failure of controlling group to move in automatic occurs, then go to section E_4, Failure of Controlling Group to Move in Automatic.	☐	☐	
7. If Control Element Assembly position indication malfunctions, then go to section E_5, CEA Position Indication Malfunction.	☐	☐	

E_1 CEA MISALIGNMENT GREATER THAN 7 INCHES

	PLACEKEEPER START	DONE	N/A
1. Match T_{avg} and T_{ref} by performing the following:	☐	☐	
• Adjust Turbine load in accordance with OP-010-004, Power Operations,	☐	☐	
• Adjust RCS boron concentration in accordance with OP-002-005, Chemical and Volume Control.	☐	☐	
2. Notify Duty Plant Manager <u>and</u> Duty Engineering. **[P-16411]**	☐	☐	
3. Record time of CEA misalignment >7 inches in Station Log. **[P-16411]**		☐	
4. <u>If</u> CEA misalignment >19 inches, <u>then</u> <u>go</u> <u>to</u> step 8.	☐	☐	☐

CAUTION

A POWER REDUCTION MUST BE STARTED WITHIN 15 MINUTES OF CEA MISALIGNMENT >7 INCHES TO COMPLY WITH TECH SPEC 3.1.3.1.

	START	DONE	N/A
5. Restore CEA alignment to within 7 inches of all other CEAs in group as follows:	☐	☐	
• Position misaligned CEA at a rate <u>not</u> to exceed 15 inches per minute using Manual Individual mode.	☐	☐	
• Compensate for reactivity changes during CEA alignment by boration, dilution, <u>or</u> CEA movement during CEA alignment.	☐	Continuous ■	
6. <u>If</u> PMC is Operable, <u>then</u> verify CEA Pulse Counter indication is correct <u>or</u> enter the correct CEA position in the PMC database.	☐	☐	☐
7. <u>If</u> CEA misalignment corrected within 15 minutes, <u>then</u> <u>go</u> <u>to</u> step 16.	☐	☐	☐

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000005 AA2.03	
	Importance Rating	3.5	

K/A Statement

AA2.03 Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable

Proposed Question: RO 20 Rev: 0

Given:

- A Reactor trip and Loss of Offsite Power has occurred
- Three CEA's are stuck out on the Reactor trip
- Emergency boration is in progress using BAM pump A

After five minutes, the RAB watch reports a large lube oil leak on EDG A and the CRS directs securing EDG A by pulling the overspeed trip mechanism. Which of the following actions should the crew take as a result of the changing plant conditions?

A. Restore Emergency Boration using BAM Pump B.

B. Restore Emergency Boration using the Gravity Feed valves.

C. Commence a plant cooldown to 400°F using ADVs and EFW.

D. Depressurize the RCS to less than 1400 psia and initiate HPSI flow into one cold leg.

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: BAM Pump B is powered from the A train of electrical distribution. This pump will not be available.
B. **CORRECT:** Gravity feed valves are powered from B train of electrical distribution and are available to perform Emergency Boration.
C. Incorrect: This action is for a Station Blackout. EDG B is still available.
D. Incorrect: This action is performed in the functional recovery procedure (RC-3) if boration is not greater than 40 gpm. The applicant would choose this distractor if he determines boration through the charging pumps is not available.

Technical Reference(s): OP-902-008 revision 22
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE08 obj. 4 (As available)

Question Source: Bank # 07688
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43

D. IMMEDIATE OPERATOR ACTIONS

1. Place Makeup Mode selector switch to MANUAL.

2. Align borated water source by performing one of the following (a. or b.):

   a. Initiate Emergency Boration using Boric Acid Pump as follows:

      - Open Emergency Boration Valve, BAM-133.
      - Start one Boric Acid Pump.
      - Close recirc valve for Boric Acid Pump started:
        - BAM-126A Boric Acid Makeup Pump Recirc Valve A

          or

        - BAM-126B Boric Acid Makeup Pump Recirc Valve B

      or

   b. Initiate Emergency Boration using Gravity Feed as follows:

      - Open the following Boric Acid Makeup Gravity Feed valves:
        - BAM-113A Boric Acid Makeup Gravity Feed Valve A
        - BAM-113B Boric Acid Makeup Gravity Feed Valve B

3. Close VCT Disch Valve, CVC-183.

4. Verify at least one Charging Pump operating and Charging Header flow $\geq$40 GPM.

E. SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

	PLACEKEEPER START	PLACEKEEPER DONE	PLACEKEEPER N/A
1. Verify Emergency Boration flow as follows:	☐	☐	
1.1 Lowering level in Boric Acid Makeup Tank A (BAM-ILI-0206) or Boric Acid Makeup Tank B (BAM-ILI-0208).	☐	☐	☐
1.2 Charging Header flow (CVC-IFI-0212) >40 GPM.		☐	☐
2. Direct Chemistry Department to take RCS boron samples and verify rising boron concentration.		☐	
3. Advise Shift Manager to implement EP-001-001, Recognition and Classification of Emergency Condition.		☐	
4. If Boric Acid Make-up Tank A Level LO-LO (Cabinet G, H-6) or Boric Acid Make-up Tank B Level LO-LO (Cabinet G, H-7) alarms for Boric Acid Makeup Tank used as Emergency Boration source, then transfer to other tank as follows:	☐	☐	☐
4.1 Start Boric Acid Pump for Boric Acid Makeup Tank to be used as Emergency Boration source.		☐	
4.2 Close the recirc valve for operating Boric Acid Pump:		☐	
• BAM-126A Boric Acid Makeup Pump A Recirc Valve		☐	☐
or			
• BAM-126B Boric Acid Makeup Pump B Recirc Valve		☐	☐

E_0 GENERAL (CONT'D)

PLACEKEEPER: START | DONE | N/A

4.3 If Boric Acid Pump cannot be started, then Open the associated Gravity Feed valve: ☐ DONE ☐ N/A

- BAM-113A Boric Acid Makeup Gravity Feed Valve Tank A ☐ DONE ☐ N/A

or

- BAM-113B Boric Acid Makeup Gravity Feed Valve Tank B ☐ DONE ☐ N/A

4.4 Stop Boric Acid Pump operating on Boric Acid Makeup Tank with LO-LO level annunciator. ☐ DONE

5. If Boric Acid Makeup Tank A and Boric Acid Makeup Tank B levels are ≤15%, then align RWSP as Emergency Boration source as follows: ☐ START ☐ DONE ☐ N/A

5.1 Open RWSP to Charging Pumps, CVC-507. ☐ DONE

5.2 Stop Boric Acid Pumps. ☐ DONE

5.3 Close Emergency Boration Valve, BAM-133. ☐ DONE

5.4 Close Gravity Feed Valves: ☐ START ☐ DONE

- BAM-113A Boric Acid Makeup Gravity Feed Valve Tank A ☐ DONE
- BAM-113B Boric Acid Makeup Gravity Feed Valve Tank B ☐ DONE

SAFETY FUNCTION: **Reactivity Control**
SUCCESS PATH: **RC-3: Boration using Safety Injection**
RESOURCE TREE: **Tree A**

INSTRUCTIONS | CONTINGENCY ACTIONS

Depressurize the RCS

* 4. **IF** safety injection flow is less than 40 gpm due to high RCS pressure, **THEN** depressurize the RCS:

a. REFER TO the HR success path in use and control RCS heat removal.

b. Control pressurizer heaters and main or auxiliary pressurizer spray.

c. **IF** HPSI throttle criteria are met, **THEN** perform the following:

- Control charging and letdown
- Throttle HPSI flow

11.3 BORIC ACID MAKEUP SYSTEM STANDBY BREAKER LINEUP

COMPONENT NUMBER	COMPONENT DESCRIPTION	LOCATION	REQUIRED POSITION	PERFORMED BY (INITIAL/DATE)	IV BY (INITIAL/DATE)
BAM-EBKR-014AB-8	[redacted]	RAB+21 11A&K	ON		
BAM-EBKR-1AB-17	BORIC ACID BATCH TANK TEMPERATURE CONTROLLER	RAB+21 10A&J (SUPS AB)	ON		
[redacted]	BORIC ACID MAKEUP TANK A HEATERS 1-1 (BAMEHTR311A)	RAB+21 10A&H	ON		
BAM-EBKR-311A-12D	BORIC ACID MAKEUP TANK B HEATERS 1-2 (BAMEHTR311B)	RAB+21 10A&H	ON		
BAM-EBKR-311A-7C	EMERGENCY BORATION VALVE (BAM-133)	RAB+21 10A&H	ON		
[redacted]	BORIC ACID MAKEUP TANK A HEATERS 1-2 (BAMEHTR311B)	RAB+21 10A&K	ON		
BAM-EBKR-311B-12D	BORIC ACID MAKEUP TANK B HEATERS 2-2 (BAMEHTR311B)	RAB+21 10A&K	ON		
BAM-EBKR-311B-7C	BORIC ACID MAKEUP TANK B GRAVITY FEED (BAM-113B)	RAB+21 10A&K	ON		
BAM-EBKR-312A-2D	BORIC ACID MAKEUP PUMP B (BAMMPMP0001-B)	RAB+21 11A&H	ON		
BAM-EBKR-312AB-2M	BORIC ACID BATCH TANK HEATERS (BAMEHTR312AB)	RAB+21 11A&J	OFF		
BAM-EBKR-312B-2J	[redacted]	RAB+21 9A&K	ON		
BAM-EBKR-313A-3D	BORIC ACID MAKEUP PUMP A (BAMMPMP0001-A)	RAB+21 10A&G	ON		
BAM-EBKR-64AB-9	BORIC ACID BATCH TANK MIXER (BAMEMTR64AB)	RAB+21 11A&J	ON		
BAM-EBKR-90A-10	BORIC ACID MAKEUP PUMPS A & B CONTROL POWER	RAB+35 11A&G	ON		
BAM-EBKR-90A-5	BORIC ACID MAKEUP PUMP A RECIRC VALVE (BAM-126A)	RAB+35 11A&G	ON		

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.2.9 Boron concentration shall be verified consistent with SHUTDOWN MARGIN requirements of Specifications 3.1.1.1, 3.1.1.2, and 3.9.1. Boron dilution events shall be precluded by either "a" or "b" below.

a. 1. Two boron dilution alarms (startup channel high neutron flux) shall be OPERABLE with the alarms set in accordance with Specification 4.1.2.9.5

and

2. i. If the plant is in MODE 4, then remove power to at least one charging pump.

ii. If the plant is in MODE 5 with $k_{eff} \leq 0.97$, then remove power to at least one charging pump.

iii. If the plant is in MODE 5 with $k_{eff} > 0.97$, then remove power to at least two charging pumps.

iv. If the plant is in MODE 6, then remove power to at least two charging pumps.

OR

b. 1. The primary makeup water flow path to the reactor coolant system shall be isolated

and

2. Do not operate the plant in the configurations prohibited by the COLR for the current MODE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

*While any shutdown CEA is less than 145 inches withdrawn.

ACTION:

a. With the boron concentration not consistent with required SHUTDOWN MARGIN, initiate emergency boration.

b. With one boron dilution alarm inoperable and the primary makeup water flow path to the reactor coolant system not isolated, determine reactor coolant system boron concentration within one hour and at least at the monitoring frequency specified in the COLR.

c. With both boron dilution alarms inoperable and the primary makeup water flow path to the reactor coolant system not isolated, determine the reactor coolant system boron concentration by two independent means within one hour and at least at the monitoring frequency specified in the COLR; otherwise, immediately suspend all operations involving positive reactivity changes or CORE ALTERATIONS (if applicable).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.9.1 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 from MODE 2.

4.1.2.9.2 Each required boron dilution alarm shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

4.1.2.9.3 If the primary makeup water flow path to the Reactor Coolant System is isolated to fulfill 3.1.2.9.b, the required primary makeup water flow path to the Reactor Coolant System shall be verified to be isolated by either locked closed manual valves, deactivated automatic valves secured in the isolation position, or by power being removed from all charging pumps, at least once per 24 hours.

4.1.2.9.4 The requirements of Specification 3.1.2.9.a.2 or 3.1.2.9.b.2 shall be verified at least once per 24 hours.

4.1.2.9.5 Each required boron dilution alarm setpoint shall be adjusted to less than or equal to the existing neutron flux (cps) multiplied by the value specified in the COLR, at the frequencies specified in the COLR.

THIS PAGE IS NOT USED

COLR TABLE 1

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.98

$K_{eff} > 0.98$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	0.75 hours	Operation not allowed **	
4	12 hours	Operation not allowed **		
5 RCS filled	8 hours	Operation not allowed **		
5 RCS partially drained	8 hours	Operation not allowed **		
6	Operation not allowed **			

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 2

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.97 AND LESS THAN OR EQUAL TO 0.98

$0.98 \geq K_{eff} > 0.97$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	2.0 hours	0.5 hours	Operation not allowed**
4	12 hours	0.75 hours	Operation not allowed**	
5 RCS filled	8 hours	0.75 hours	Operation not allowed**	
5 RCS partially drained	8 hours	0.5 hours	Operation not allowed**	
6		Operation not allowed**		

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 3

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.96 AND LESS THAN OR EQUAL TO 0.97

$0.97 \geq K_{eff} > 0.96$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	3.0 hours	1.25 hours	0.5 hours
4	12 hours	1.5 hours	Operation not allowed**	
5 RCS filled	8 hours	1.5 hours	Operation not allowed**	
5 RCS partially drained	8 hours	0.75 hours	Operation not allowed**	
6		Operation not allowed**		

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.95 AND LESS THAN OR EQUAL TO 0.96

$0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	4.0 hours	2.0 hours	1.0 hours
4	12 hours	2.25 hours	0.75 hours	Operation not allowed**
5 RCS filled	8 hours	2.0 hours	0.75 hours	Operation not allowed**
5 RCS partially drained	8 hours	2.0 hours	0.5 hours	Operation not allowed**
6		Operation not allowed**		

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 5

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} LESS THAN OR EQUAL TO 0.95

$K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	5.0 hours	2.0 hours	1.0 hours
4	12 hours	2.75 hours	1.0 hours	Operation not allowed**
5 RCS filled	8 hours	3.0 hours	1.0 hours	0.5 hours
5 RCS partially drained	8 hours	2.5 hours	0.75 hours	Operation not allowed**
6	24 hours	2.25 hours	0.5 hours	Operation not allowed**

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000032 2.1.25	
	Importance Rating	3.9	

K/A Statement

2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO 21 Rev: 0

Given:

- The plant is in Mode 5
- RCS level is 19.0 feet
- The last OP-903-090 Shutdown Margin calculated a Keff of 0.965
- Charging pump A and AB are in operation to support RCS CRUD cleanup
- Charging pump B breaker is racked out
- Startup Channel #2 has just failed high

Which describes **all** required actions to comply with Tech Specs 3.1.2.9?

A. Isolate all Primary Water flow paths to the RCS.

B. Determine RCS boron concentration within 1 hour and every 90 minutes thereafter.

C. Determine RCS boron concentration within 1 hour, isolate all Primary Makeup Water flow paths to the RCS. Secure an additional charging pump and rack out the pump breaker.

D. Determine RCS boron concentration within 1 hour, Isolate all Primary Makeup Water flow paths to the RCS and secure the remaining charging pumps.

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: This selection is plausible because the candidate could determine that charging pump status is adequate and no additional requirements other than isolating PMU are needed.
B. Incorrect: This selection is plausible because the candidate could misinterpret the table which directs the crew to remove two charging pumps from service. In this scenario, one charging pump would be required to be removed from service and breaker racked out.
C. CORRECT: These actions comply with the requirements of TS 3.1.2.9
D. Incorrect: This distractor is plausible because it would be a correct answer if all charging pumps are removed from service, but this distractor does not rack out the charging pump breakers. The charging pump breakers must be racked out to meet the table.

Technical Reference(s): TS 3.1.2.9 and the COLR
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: TS 3.1.2.9 and COLR Tables 1, 2, 3, 4, 5

Learning Objective: WLP-OPS-REQ04 obj. 1 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 6
55.43 ______

Comments:

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.2.9 Boron concentration shall be verified consistent with SHUTDOWN MARGIN requirements of Specifications 3.1.1.1, 3.1.1.2, and 3.9.1. Boron dilution events shall be precluded by either "a" or "b" below.

a. 1. Two boron dilution alarms (startup channel high neutron flux) shall be OPERABLE with the alarms set in accordance with Specification 4.1.2.9.5

and

2. i. If the plant is in MODE 4, then remove power to at least one charging pump.

ii. If the plant is in MODE 5 with $k_{eff} \leq 0.97$, then remove power to at least one charging pump.

iii. If the plant is in MODE 5 with $k_{eff} > 0.97$, then remove power to at least two charging pumps.

iv. If the plant is in MODE 6, then remove power to at least two charging pumps.

OR

b. 1. The primary makeup water flow path to the reactor coolant system shall be isolated

and

2. Do not operate the plant in the configurations prohibited by the COLR for the current MODE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

*While any shutdown CEA is less than 145 inches withdrawn.

ACTION:

a. With the boron concentration not consistent with required SHUTDOWN MARGIN, initiate emergency boration.

b. With one boron dilution alarm inoperable and the primary makeup water flow path to the reactor coolant system not isolated, determine reactor coolant system boron concentration within one hour and at least at the monitoring frequency specified in the COLR.

c. With both boron dilution alarms inoperable and the primary makeup water flow path to the reactor coolant system not isolated, determine the reactor coolant system boron concentration by two independent means within one hour and at least at the monitoring frequency specified in the COLR; otherwise, immediately suspend all operations involving positive reactivity changes or CORE ALTERATIONS (if applicable).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.9.1 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 from MODE 2.

4.1.2.9.2 Each required boron dilution alarm shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

4.1.2.9.3 If the primary makeup water flow path to the Reactor Coolant System is isolated to fulfill 3.1.2.9.b, the required primary makeup water flow path to the Reactor Coolant System shall be verified to be isolated by either locked closed manual valves, deactivated automatic valves secured in the isolation position, or by power being removed from all charging pumps, at least once per 24 hours.

4.1.2.9.4 The requirements of Specification 3.1.2.9.a.2 or 3.1.2.9.b.2 shall be verified at least once per 24 hours.

4.1.2.9.5 Each required boron dilution alarm setpoint shall be adjusted to less than or equal to the existing neutron flux (cps) multiplied by the value specified in the COLR, at the frequencies specified in the COLR.

COLR TABLE 1

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.98

$K_{eff} > 0.98$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	0.75 hours	Operation not allowed **	
4	12 hours	Operation not allowed **		
5 RCS filled	8 hours	Operation not allowed **		
5 RCS partially drained	8 hours	Operation not allowed **		
6	Operation not allowed **			

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 2

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.97 AND LESS THAN OR EQUAL TO 0.98

$0.98 \geq K_{eff} > 0.97$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	2.0 hours	0.5 hours	Operation not allowed**
4	12 hours	0.75 hours	Operation not allowed**	
5 RCS filled	8 hours	0.75 hours	Operation not allowed**	
5 RCS partially drained	8 hours	0.5 hours	Operation not allowed**	
6		Operation not allowed**		

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 3

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.96 AND LESS THAN OR EQUAL TO 0.97

$0.97 \geq K_{eff} > 0.96$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	3.0 hours	1.25 hours	0.5 hours
4	12 hours	1.5 hours	Operation not allowed**	
5 RCS filled	8 hours	1.5 hours	Operation not allowed**	
5 RCS partially drained	8 hours	0.75 hours	Operation not allowed**	
6		Operation not allowed**		

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} GREATER THAN 0.95 AND LESS THAN OR EQUAL TO 0.96

$0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
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4	12 hours	2.25 hours	0.75 hours	Operation not allowed**
5 RCS filled	8 hours	2.0 hours	0.75 hours	Operation not allowed**
5 RCS partially drained	8 hours	2.0 hours	0.5 hours	Operation not allowed**
6		Operation not allowed**		

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

COLR TABLE 5

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR K_{eff} LESS THAN OR EQUAL TO 0.95

$K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps* 0	1	2	3
3	12 hours	5.0 hours	2.0 hours	1.0 hours
4	12 hours	2.75 hours	1.0 hours	Operation not allowed**
5 RCS filled	8 hours	3.0 hours	1.0 hours	0.5 hours
5 RCS partially drained	8 hours	2.5 hours	0.75 hours	Operation not allowed**
6	24 hours	2.25 hours	0.5 hours	Operation not allowed**

* Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

** The precluded number of charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000037 AA2.15	
	Importance Rating	3.4	

K/A Statement

AA2.15 Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Magnitude of atmospheric radioactive release if cool-down must be completed using steam dump or atmospheric reliefs

Proposed Question: RO 22 Rev: 0

Given:

- A Steam Generator Tube leak on #1 Steam Generator is in progress
- The crew is performing the actions in OP-901-201, Steam Generator Tube Leakage or High Activity

The crew will perform a plant cooldown using the ____(1)_____ . The release path will be ____(2)_____.

	(1)	(2)
A.	Atmospheric Dump Valve	monitored
B.	Atmospheric Dump Valve	unmonitored
C.	Steam Bypass Control System	monitored
D.	Steam Bypass Control System	unmonitored

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: OP-901-202 directs the crew to use the Steam Bypass control system if the main condenser is available. The applicant will recognize the Main Condenser is available since the crew is not performing actions IAW OP-902-007. The main condenser is a monitored release path.
B. Incorrect: OP-901-202 directs the crew to use the Steam Bypass control system if the main condenser is available. The applicant will recognize the Main Condenser is available since the crew is not performing actions IAW OP-902-007. The atmospheric dump valves are an unmonitored release path.
C. **CORRECT:** OP-901-202 directs the crew to use the Steam Bypass control system if the main condenser is available. The applicant will recognize the Main Condenser is available since the crew is not performing actions IAW OP-902-007. The main condenser is a monitored release path.
D. Incorrect: OP-901-202 directs the crew to use the Steam Bypass control system if the main condenser is available. The applicant will recognize the Main Condenser is available since the crew is not performing actions IAW OP-902-007. The atmospheric dump valves are an unmonitored release path, but the main condenser is monitored.

Technical Reference(s): OP-901-202 revision 11
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO20 obj. 3 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

E0 GENERAL (CONT'D)

	PLACEKEEPER	
START	DONE	N/A

NOTE

(1) The calculated Primary to Secondary Leak Rate values must be considered valid, unless the reading can be quickly diagnosed as incorrect due to an obvious malfunction of the PMC or AE Discharge Rad Monitor.

(2) The AE Discharge Rad Monitor is considered the primary Rad Monitor which has the sensitivity to measure small Primary to Secondary Leakage. The AE Discharge Rad Monitor reading inputs into the Primary to Secondary Leak Calculation on PMC Group PLSR. The MS Line N16 Rad Monitors may be used as verification of AE Discharge Rad Monitor or as primary indication if the AE Discharge Rad Monitor is OOS.

(3) If Primary to Secondary leakage will result in a backup Charging Pump cycling to maintain Pressurizer level, starting and continuously running an additional Charging Pump will allow for a more accurate leakrate determination.

Step	START	DONE	N/A
2. Determine RCS leak rate using ANY of the following:	☐	☐	
• Calculated Steam Generator leakage displayed on PMC Group PSLR (PMC PID C48304)	☐	☐	☐
• Calculated Steam Generator leakage displayed on PMC PID C48251 (RE5501 CH1 LEAK RATE) and C48252 (RE5501 CH2 LEAK RATE)	☐	☐	☐
• Approximate RCS leak rate by subtracting total of Letdown flow AND RCP CBO flow from Charging Flow	☐	☐	☐
• Calculated Charging / Letdown Mismatch displayed on PMC Group Leakrate (PMC PID S13001)	☐	☐	☐
• RCS Leak Rate calculation in accordance with OP-903-024, REACTOR COOLANT SYSTEM WATER INVENTORY BALANCE	☐	☐	☐
• Calculated steam generator leakage based upon chemistry sample, per CE-003-705, DETERMINATION OF PRIMARY-TO-SECONDARY LEAK RATE	☐	☐	☐
2.1 If necessary, then start an additional Charging Pump.	☐	☐	☐

E0 GENERAL (CONT'D)

	PLACEKEEPER		
	START	DONE	N/A

NOTE

Steps 21 through 28 are applicable only after the Plant has entered Mode 3.

21. Maintain level in BOTH Steam Generators 50% to 70% Narrow Range.

CAUTION

(1) MAINTAIN RCS TEMPERATURE AND PRESSURE WITHIN THE LIMITS OF TECHNICAL SPECIFICATION 3.4.8.1, FIGURE 3.4-3.

(2) RCS TEMPERATURE SHALL BE LIMITED TO A MAXIMUM COOLDOWN RATE OF 100▪F PER HOUR (T.S. 3.4.8.1).

(3) RCS PARAMETERS SHALL NOT EXCEED RG 1.121 BASED PRESSURE AND TEMPERATURE LIMITS SHOWN ON THE RCS PRESSURE AND TEMPERATURE LIMITS GRAPH (ATTACHMENT 11.1) IN OP-001-002, REACTOR COOLANT PUMP OPERATION.

22. Commence a normal plant cooldown to obtain Hot Leg temperature of ■520°F as follows:

22.1 WHEN the following Steam Generator Low Pressure Pretrip alarms occur, THEN reset the Steam Generator Low Pressure Trip setpoint on ALL FOUR channels:

(A) For Steam Generator 1:

- SG 1 PRESSURE LO PRETRIP A/C (F-15, Cabinet K)
- SG 1 PRESSURE LO PRETRIP B/D (G 15, Cabinet K)

(B) For Steam Generator 2:

- SG 2 PRESSURE LO PRETRIP A/C (F-16, Cabinet K)
- SG 2 PRESSURE LO PRETRIP B/D (G-16, Cabinet K)

E0 GENERAL (CONT'D)

Step		PLACEKEEPER START	PLACEKEEPER DONE	PLACEKEEPER N/A
22.2	During cooldown, record Reactor Coolant System (RCS) and Pressurizer cooldown rates every 15 minutes on Attachment 1, Pressurizer / RCS Cooldown Log.	☐	Continuous	☐
22.3	PRIOR TO reducing RCS temperature below 380°F Cold Leg Temperature, secure two RCPs, preferably RCP 1A and RCP 2A, in accordance with OP-001-002, REACTOR COOLANT PUMP OPERATION.	☐	☐	
22.3.1	Align Pressurizer Spray Valves Select Switch to select the spray valve for the Operating RCP (LOOP 1B <u>OR</u> LOOP 1A).		☐	
22.4	<u>IF</u> Condenser is available, <u>THEN</u> perform the following:	☐	☐	☐
22.4.1	Place SBCS Master Controller, MS-IPIC-1010, to MANUAL.		☐	☐
22.4.2	Place M/A station for one SBCS Valve to MANUAL.		☐	☐
22.4.3	Place its associated permissive switch to MANUAL.		☐	☐
22.4.4	Place remaining permissive SBCV switches to OFF.		☐	☐

E0 GENERAL (CONT'D)

	PLACEKEEPER START	PLACEKEEPER DONE	PLACEKEEPER N/A
22.5 IF Condenser is NOT available, THEN perform the following:	☐	☐	☐
22.5.1 Throttle Open the Atmospheric Dump valve on the Steam Generator with the lowest indicated activity.	☐	☐	
22.5.2 Notify Chemistry for calculating of exposure doses.		☐	
22.5.3 IF an Atmospheric Dump valve for a Steam Generator with activity is Opened, THEN notify the Shift Manager an unmonitored release is occurring through the Atmosphere Dump valve.		☐	☐

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000061 AK1.01	
	Importance Rating	2.5	

K/A Statement

AK1.01 Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring (ARM) System Alarms: Detector limitations

Proposed Question: RO 23 Rev: 0

During a LOCA or Steam Line Break in Containment, the _____(1)______ Radiation Monitor(s) will read erroneously _____(2)______ while Containment temperature is rising.

	(1)	(2)
A.	Containment PIG	low
B.	Containment PIG	high
C.	Containment High Range	low
D.	Containment High Range	high

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: The Containment High Range Radiation Monitors have been determined to be susceptible to TIC post accident which will cause erroneously high readings for at least 15 minutes from the time containment temperature stabilizes.
B. Incorrect: The Containment High Range Radiation Monitors have been determined to be susceptible to TIC post accident which will cause erroneously high readings for at least 15 minutes from the time containment temperature stabilizes.
C. Incorrect: The correct Rad monitor but the wrong effect.
D. **CORRECT:** The Containment High Range Radiation Monitors have been determined to be susceptible to TIC post accident which will cause erroneously high readings for at least 15 minutes from the time containment temperature stabilizes

Technical Reference(s): OI-038-000 step 5.5.1 Revision 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 4 (As available)
WLP-OPS-MCD06 obj. 3

Question Source: Bank # X Question # 71
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

5.5 STANDARD APPENDICES

5.5.1 Appendix 1 Diagnostic Flowchart

- All available indications should be used to aid in evaluating plant conditions since the accident may cause irregularities in a particular instrument reading.
- High containment temperature conditions may adversely impact the accuracy of instruments whose transmitters are located inside containment and may impact the continued availability of equipment located inside containment.
- Post Accident Thermally induced currents will cause the Containment High Range Radiation Monitors to read erroneously high and it will be approximately 15 minutes post event until the Containment High Range Radiation Monitors read actual Radiation levels in Containment.
- Actions taken as an immediate response to an event may reduce or eliminate the symptoms contained in the Diagnostic Flowchart. The initial indications should not be ignored in establishing a diagnosis.

5.5.2 Appendix 2 Figures

- T cold should be used when available to evaluate the RCP NPSH curve.
- The HPSI flow curve is based on individual injection line flow rates. All injection paths must be available to meet acceptable limits.
- The LPSI flow curve is based on either Train A or B total flow.

5.5.3 Appendix 3 Logs and Tables

- None

5.5.4 Appendix 4 ESFAS Auto Actions

- The crew should utilize the third NPO if available to perform the ESFAS Auto Actions Checklist.
- Any auto actions that do not occur should be reported to the CRS.

5.5.5 Appendix 5 ESFAS Reset

- None

5.5.6 Appendix 6 ESFAS Equipment Restoration

- None

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000074 EK2.09	
	Importance Rating	2.6	

K/A Statement

EK2.09 Knowledge of the interrelations between the and the following Inadequate Core Cooling: Controllers and positioners

Proposed Question: RO 24 Rev: 0

Per TS 3.7.1.7, Atmospheric Dump Valves, the Atmospheric Dump Valve controller setpoints are required to be operable prior to exceeding ____(1)____ percent power. The basis for this requirement is to mitigate the effects of a ____(2)_____.

	(1)	(2)
A.	70	Small Break LOCA
B.	70	Station Blackout
C.	85	Small Break LOCA
D.	85	Station Blackout

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** Per TS 3.7.1.7, the ADV automatic actuation channels shall be operable greater than 70% Rated Thermal Power. Per TS 3.7.1.7 basis, above 70% power, an ADV is required along with a HPSI Train to mitigate the effects of a SBLOCA to ensure adequate core cooling.
B. Incorrect: Per TS 3.7.1.7 basis, above 70% power, an ADV is required along with a HPSI Train to mitigate the effects of a SBLOCA. The station blackout is credible due to the importance of the ADVs during a loss of all power.
C. Incorrect: 85% power is the power level that the plant is required to reduce to in the event of one MSSV inoperable.
D. Incorrect: 85% power is the power level that the plant is required to reduce to in the event of one MSSV inoperable. Per TS 3.7.1.7 basis, above 70% power, an ADV is required along with a HPSI Train to mitigate the effects of a SBLOCA to ensure adequate core cooling.

Technical Reference(s): TS 3.7.1.7
(Attach if not previously provided) Basis for TS 3.7.1.7
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-MS00 obj. 6 (As available)

Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 4
55.43

Comments:

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.7 Each Atmospheric Dump Valve (ADV) shall be OPERABLE*.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With the automatic actuation channel for one ADV inoperable, restore the inoperable ADV to OPERABLE status within 72 hours or reduce power to less than or equal to 70% RATED THERMAL POWER within the next 6 hours.

b. With the automatic actuation channels for both ADVs inoperable, restore one ADV to OPERABLE status within 1 hour or reduce power to less than or equal to 70% RATED THERMAL POWER within the next 6 hours.

c. With one ADV inoperable, for reasons other than above, restore the ADV to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The provisions of Specification 3.0.4 are not applicable provided one ADV is OPERABLE.

SURVEILLANCE REQUIREMENTS

4.7.1.7 The ADVs shall be demonstrated OPERABLE:

a. By performing a CHANNEL CHECK at least once per 12 hours when the automatic actuation channels are required to be OPERABLE.

b. By verifiying each ADV automatic actuation channel is in automatic with a setpoint of less than or equal to 1040 psia at least once per 92 days when the automatic actuation channels are required to be OPERABLE.

c. By verifying one complete cycle of each ADV when tested pursuant to the Inservice Testing Program.

d. By performing a CHANNEL CALIBRATION of each ADV automatic actuation channel at least once per 18 months.

e. By verifying actuation of each ADV to the open position on an actual or simulated automatic actuation signal at least once per 18 months.

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* ADV automatic actuation channels (one per ADV, in automatic with a setpoint of less than or equal to 1040 psia) are not required to be OPERABLE when less than or equal to 70% RATED THERMAL POWER for greater than 6 hours.

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)

• (DRN 02-1684, Ch. 15; 03-1807, Ch. 30)

the trip circuitry, is less than the setpoint specified in the Component Database plus the specified tolerance at least once per 18 months. The AFW pump trip shall be verified through the performance of Operations surveillance test procedure, "AFW High Discharge Pressure Trip Test." The relay delay time shall be verified through the performance of an Electrical Maintenance task document for relay AFWEREL 1419. The 18 month frequency is based on the refueling cycle, similar to testing performed per the Inservice Testing Program. This frequency is acceptable from a reliability standpoint to detect degradation.

• (DRN 02-1684, ch. 15; 03-1807, Ch. 30)

• (DRN 04-1243, Ch. 38)

3/4.7.1.7 ATMOSPHERIC DUMP VALVES (ADVs)

Two ADVs are provided, one per steam generator. The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ADVs are provided with a pressurized nitrogen gas supply that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The ADVs can also be operated manually once the nitrogen gas supply is depleted.

The ADVs provide a safety grade method for cooling the unit to Shutdown Cooling (SDC) System entry conditions, should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section 10.3. This is done in conjunction with the Emergency Feedwater System providing cooling water from the condensate storage pool (CSP) to meet Branch Technical Position (BTP) RSB 5-1.

The automatic operation of the ADVs to open is assumed in the Small Break LOCA (SBLOCA) analysis at power levels above 70% RATED THERMAL POWER. ADVs are credited for SBLOCA analysis to lower steam generator secondary side pressures, compared to crediting only MSSVs, and thus provide increased cooling of the RCS. This results in a lower calculated peak cladding temperature (PCT) for SBLOCA ECCS analysis.

Analysis has shown that automatic operation of the ADV is not required when the unit is at or below 70% RATED THERMAL POWER for greater than six hours because, based on decay heat load, one high-pressure safety injection train is capable of mitigating the SBLOCA event. At greater than 70% RATED THERMAL POWER, one high-pressure safety injection train and one ADV, in automatic, are capable of mitigating the SBLOCA event. Therefore, the ADVs, in automatic, are required at greater than 70% RATED THERMAL POWER and for six hours after reducing power to less than or equal to 70% RATED THERMAL POWER.

Limiting Condition for Operation

The LCO requires that each ADV be OPERABLE.

The ADV manual controls must be OPERABLE in MODES 1, 2, 3, and 4 to allow operator action needed for decay heat removal and safe shutdown in accordance with BTP RSB 5-1.

• (DRN 04-1243, Ch. 38)

3.4.7.1.7 ATMOSPHERIC DUMP VALVES (ADVs)(Continued)

The LCO is modified by a footnote requiring that ADV automatic actuation controls be OPERABLE (i.e., ADVs in automatic and capable of automatic actuation at less than or equal to 1040 psia (992 psig indicated)) when operating at greater than 70% RATED THERMAL POWER and for six hours after reducing power to less than or equal to 70% RATED THERMAL POWER for mitigation of the SBLOCA.

The ADVs are containment isolation valves and must be capable of manual isolation of the ADV lines in MODE 1, 2, 3, and 4 in order to be considered OPERABLE. Because the OPERABILITY of the ADVs is controlled by this Technical Specification, Technical Specification 3.6.3, "Containment Isolation Valves," does not apply to the ADVs.

ACTIONs

The ACTIONs are modified by a note indicating that the provisions of Specification 3.0.4 are not applicable provided one ADV is OPERABLE. This allows for MODE changes with one ADV inoperable provided the appropriate ACTION is entered upon entry into the applicability MODEs.

ACTIONs (a) and (b) would be applicable only when the automatic actuation channels are required to be OPERABLE per the LCO footnote.

a. This ACTION addresses the condition when one ADV is incapable of automatic actuation. This condition includes:
   - A malfunctioning automatic actuation channel, or
   - When the automatic actuation controls for one ADV have been placed in manual.

   A 72 hour allowed outage time is provided to restore the ADV to an OPERABLE status. The 72 hour allowed outage time takes into account the capability afforded by the remaining OPERABLE ADV and is consistent with the allowed outage time of an inoperable high-pressure safety injection train.

   If the ADV can not be restored to an OPERABLE status within the allowed outage time, the unit must be placed in a status in which the LCO does not apply. To achieve this status, power must be reduced to less than or equal to 70% RATED THERMAL POWER within the next 6 hours. The LCO will no longer apply once the unit has been at less than or equal to 70% RATED THERMAL POWER for greater than six hours.

b. This ACTION addresses the condition when both ADVs are incapable of automatic actuation. This condition includes:
   - Malfunctioning of both ADV's automatic actuation channel,
   - When the automatic actuation controls for both ADVs have been placed in manual, or
   - A combination of the above such that both ADVs are incapable of automatic operation.

• (DRN 04-1243, Ch. 38)

3.4.7.1.7 ATMOSPHERIC DUMP VALVES (Continued)

In this condition, the SBLOCA can not be mitigated by one high-pressure safety injection train alone. Therefore, one of the ADVs must be restored to OPERABLE status within 1 hour or power must be reduced to less than or equal to 70% RATED THERMAL POWER within the next six hours. The LCO will no longer apply once the unit has been at less than or equal to 70% RATED THERMAL POWER for greater than six hours.

c. This ACTION address the condition when one ADV is inoperable for reasons other than those addressed in ACTIONs (a) and (b) above. This condition includes:
   - The inability to operate the ADV manually via the handwheel, or
   - The inability to operate the ADV manually via the controller in the control room, or
   - An inoperable nitrogen accumulator.

   A 72 hour allowed outage time is provided to restore the ADV to an OPERABLE status. The 72 hour allowed outage time takes into account the capability afforded by the remaining OPERABLE ADV, a nonsafety grade backup in the Steam Bypass System and MSSVs, the closed system inside containment, and the backup isolation capability of the block valve.

   If the ADV can not be restored to an OPERABLE status within the allowed outage time, the unit must be placed in a status in which the LCO does not apply. To achieve this status, the unit must be placed in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The following conditions are not addressed by the ACTION statements:
- The automatic actuation channel for one ADV is inoperable and the other ADV is inoperable for other reasons.
- Both ADVs are inoperable for reasons other than the automatic actuation channels.

For these conditions, Specification 3.0.3 is entered.

Surveillance Requirements

a. To mitigate the SBLOCA event, the ADVs must automatically open at a pressure of less than or equal to 1040 psia (992 psig indicated). This CHANNEL CHECK provides assurance that the behavior of the steam line pressure input to the automatic actuation channel is reasonable for the existing plant conditions. This steam line pressure input is available on the plant monitoring computer or from appropriate maintenance and test equipment. This Surveillance Requirement (SR) need not be performed when the ADV automatic actuation channels are not required to be OPERABLE per the LCO footnote.

• (DRN 04-1243, Ch. 38)

3.4.7.1.7 ATMOSPHERIC DUMP VALVES (ADVs) (Continued)

b. To mitigate the SBLOCA event, the ADVs must automatically open at a pressure of less than or equal to 1040 psia (992 psig indicated). This Surveillance Requirement (SR) ensures that the ADV controllers are in automatic and set at an appropriate setpoint that is bounded by the SBLOCA safety analysis. The setpoint must be verified using the plant monitoring computer or appropriate maintenance and test equipment. This SR need not be performed when the ADV automatic actuation channels are not required to be OPERABLE per the LCO footnote.

c. To perform a controlled cooldown of the reactor coolant system, the ADVs must be able to be opened and throttled through their full range. Additionally, the ADV must be capable of being closed to fulfill its secondary function of containment isolation. This SR ensures the ADVs are tested through a full control cycle. The test interval is in accordance with the Inservice Testing Program.

d. The SR to calibrate the ADV automatic actuation channels ensures that the system will generate an actuation signal at 1040 psia (992 psig indicated) as assumed for the SBLOCA. The calibration should include the plant monitoring computer points used to set the setpoint.

e. The SR for actuation testing ensures that the ADV will automatically open on a high steam pressure signal, with a response time of less than or equal to 60 seconds, as assumed for the SBLOCA. Credit may be taken for an actual or simulated actuation signal.

• (DRN 04-1243, Ch. 38)

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator secondary pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitation to 115°F and 210 psig is based on a steam generator RTNDT of 40°F and is sufficient to prevent brittle fracture. Below this temperature of 115°F the system pressure must be limited to a maximum of 20% of the secondary hydrostatic test pressure of 1375 psia (corrected for instrument error). Should steam generator temperature drop below 115°F an engineering evaluation of the effects of the overpressurization is required. However, to reduce the potential for brittle failure the steam generator temperature may be increased to a limit of 200°F while performing the evaluation. The limitations on the primary side of the steam generator are bounded by the restrictions on the reactor coolant system in Specification 3.4.8.1.

3/4.7.3 COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS

The OPERABILITY of the component cooling water system and its corresponding auxiliary component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the safety analyses.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000076 AK3.05	
	Importance Rating	2.9	

K/A Statement

AK3.05 Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Corrective actions as a result of high-fission product radioactivity level in the RCS

Proposed Question: RO 25 Rev: 0

Given:

- Chemistry reports a high RCS Activity in the RCS
- The crew has entered OP-901-410, High Activity in the Reactor Coolant System

Per OP-901-410, the crew will _____(1)______ . The reason for performing this action is to _____(2)______ letdown flow through the Purification Ion Exchangers.

	(1)	(2)
A.	start backup charging pumps	maximize
B.	place an additional Purification Ion Exchanger in service	maximize
C.	bypass Purification Ion Exchangers	minimize
D.	secure backup charging pumps	minimize

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: OP-901-410 step 7 directs the crew to start available backup charging pumps to maximize letdown flow.

B. Incorrect: OP-901-410 does not provide direction for placing a second Purification Ion Exchanger in service. It does provide direction for placing the standby Purification Ion Exchanger in service only if the in-service IX must be removed from service because the DF < 1.0.

C. Incorrect: OP-901-410 does not provide direction for bypassing Purification IXs. Letdown flow is not minimized. This distractor is plausible because during a SGTL, the crew will remove polisher vessels from service as to not contaminate the resin.

D. Incorrect: OP-901-410 does not provide direction for securing Charging pumps. Letdown flow is not minimized. This distractor is plausible because during a SGTL, the crew will remove polisher vessels from service as to not contaminate the resin.

Technical Reference(s): OP-901-410 revision 4
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO40 obj. 3 (As available)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 10
55.43 ______

Comments:

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

	PLACEKEEPER		
	START	DONE	N/A
1. Request Chemistry Department to sample and analyze the following:	☐	☐	
• RCS for Dose Equivalent I-131 concentration and isotopics		☐	
• RCS for gross activity		☐	
• Purification Ion Exchanger influent and effluent to determine the Decontamination Factor (D.F.)		☐	
2. Advise the Shift Manager to perform the following:	☐	☐	
• Implement EP-001-001, Recognition and Classification of Emergency Conditions		☐	
• Refer to NF-102, Corporate Fuel Reliability		☐	
3. Using the Plant Paging System, make the following announcement twice:	☐	☐	
3.1 "ATTENTION ALL PERSONNEL, ATTENTION ALL PERSONNEL. RISING RADIATION LEVELS ARE IN THE REACTOR AUXILIARY BUILDING (state affected areas). ALL PERSONNEL EXIT THESE AREAS AND PROCEED TO A RADIATION PROTECTION CONTROL POINT."		☐	
4. Advise Radiation Protection of the possibility of higher dose levels in the Reactor Auxiliary Building and to consider additional controls.		☐	

E0 GENERAL (CONT'D)

	PLACEKEEPER		
	START	DONE	N/A
5. If the specific activity of the primary coolant is > 1.0 microcuries/gram Dose Equivalent I-131 or > 100/E microcuries/gram, then perform the following:	☐	☐	☐
• Comply with the actions of Technical Specification 3.4.7, Reactor Coolant System Specific Activity.	☐	Continuous ↓	
• Comply with the sampling frequency of Table 4.4-4.	☐	Continuous ↓	
5.1 If a Plant Shutdown is required, then perform OP-010-005, Plant Shutdown, concurrently with this Procedure.	☐	Continuous ↓	
6. If sample analysis indicates exhaustion of Purification Ion Exchanger (D.F. less than 1.0), then perform the following:	☐	☐	☐
6.1 Isolate the in service Purification Ion Exchanger in accordance with OP-002-005, Chemical and Volume Control.	☐	☐	☐
6.2 Place standby Purification Ion Exchanger in service in accordance with OP-002-005, Chemical and Volume Control.	☐	☐	☐
6.3 If standby Purification Ion Exchanger is not available, then contact Chemistry and Radiation Protection to expedite flushing and replacement of resin in Purification Ion Exchanger.	☐	☐	☐
7. If purification system is operating properly, then maximum Letdown flow should be established as follows:	☐	☐	☐
7.1 Start one available Backup Charging Pump by placing Control Switch to ON.		☐	
7.2 Verify that COLSS Charging and Letdown constants are correct for Charging Pump configuration in accordance with OP-004-005, Core Operating Limits Supervisory System Operation.		☐	

E$_0$ GENERAL (CONT'D)

	PLACEKEEPER		
	START	DONE	N/A
7.3 Monitor Pressurizer Level and Pressure, allowing time for stabilization.	☐	Continuous ⬇	
7.4 Verify Letdown Backpressure Controller, CVC-IPIC-0201, maintains approximately 460 PSIG.	☐	Continuous ⬇	
7.5 Verify Letdown Temperature Controller, CVC-ITIC-0223, maintains approximately 120°F.	☐	Continuous ⬇	
7.6 <u>If</u> available, <u>then</u> start second Backup Charging Pump by placing Control Switch to ON.		☐	☐
7.6.1 Verify that COLSS Charging and Letdown constants are correct for Charging Pump configuration in accordance with OP-004-005, Core Operating Limits Supervisory System Operation.		☐	☐
7.7 Monitor Pressurizer Level <u>and</u> Pressure, allowing time for stabilization.	☐	Continuous ⬇	
7.8 Verify Letdown Backpressure Controller, CVC-IPIC-0201, maintains approximately 460 PSIG.	☐	Continuous ⬇	
7.9 Verify Letdown Temperature Controller, CVC-ITIC-0223, maintains approximately 120°F.	☐	Continuous ⬇	
8. Continue to comply with the sampling frequency of Technical Specification Table 4.4-4.	☐	Continuous ⬇	☐
9. <u>When</u> maximum Letdown flow is no longer required, <u>then</u> return Charging system to desired number of pumps operating.	☐	☐	

END

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	CE/A 16 AK2.2	
	Importance Rating	3.0	

K/A Statement

AK2.2 Knowledge of the interrelations between the (Excess RCS Leakage) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: RO 26 Rev: 0

Given:

- Plant is in MODE 5
- The RCS was drained to 14.5 ft. MSL
- LPSI pumps have been secured due to RCS leakage
- HPSI Pump B has been started in accordance with OP-901-131, Shutdown Cooling Malfunction
- RCS level has been raised to and is being maintained at 16 ft MSL

To restore a Shutdown Cooling train to service in accordance with OP-901-131, the crew will vent and start LPSI Pump ____(1)____ because HPSI Pump B is injecting to _____(2)_____ .

	(1)	(2)
A.	A	Hot Leg 1
B.	A	Hot Leg 2
C.	B	Hot Leg 1
D.	B	Hot Leg 2

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: The crew will start LPSI pump A because HPSI Pump B is injecting to the suction of LPSI pump A. HPSI Pump B injects to Hot leg #2 (not hot leg #1). LPSI pump A takes a suction from hot leg 2.
B. **CORRECT:** The crew will start LPSI pump A because HPSI Pump B is injecting to the suction of LPSI pump A. HPSI Pump B injects to Hot leg #2. LPSI pump A takes a suction from hot leg 2.
C. Incorrect: The crew will start LPSI pump A because HPSI Pump B is injecting to the suction of LPSI pump A. HPSI Pump B injects to Hot leg #2 (not hot leg #1). LPSI pump A takes a suction from hot leg 2.
D. Incorrect: The crew will start LPSI pump A because HPSI Pump B is injecting to the suction of LPSI pump A. HPSI Pump A (not running) injects to Hot leg #1. LPSI pump A takes a suction from hot leg 2.

Technical Reference(s): OP-901-131 revision 303
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-REQ21 obj. 6 (As available)

Question Source: Bank # 08281
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

E1. SYSTEM LEAKAGE

PLACEKEEPER
START DONE

1. IF ANY of the following LPSI Pump cavitation/Air binding indications occur, THEN stop affected LPSI Pump: ☐ ☐
   - Dropping OR erratic ammeter indication
   - Dropping OR erratic Shutdown Cooling flow
   - Steady low flow AND amperage less than expected for system configuration
   - Local observation.

CAUTION

(1) IF RCS IS OPEN FOR MAINTENANCE, THEN FILLING MAY RESULT IN DISCHARGE FROM OPENING. THIS COULD ENDANGER PERSONNEL IN OR AROUND OPENINGS. CLOSED RCS IMPLIES NO MAJOR OPENINGS THAT WOULD PREVENT FILLING RCS HOT LEG. RCS IS CONSIDERED CLOSED WITH RCP SEAL REMOVED.

(2) IF STEAM VOIDS ARE PRESENT IN THE RCS OR SDC PIPING, THEN MAKEUP SHOULD BE PERFORMED SLOWLY TO MINIMIZE WATER HAMMER CONCERNS.

2. IF RCS makeup is required, THEN using Shutdown Cooling Train A or B perform the following: ☐ ☐

   2.1 For Shutdown Cooling Train A: ☐ ☐

   a. Start Train B HPSI Pump. ☐

   b. Close HPSI HEADER ORIFICE BYPASS VALVE (SI 219B). ☐

   c. Open HOT LEG 2 INJECTION ISOLATION VALVE (SI 502B). ☐

E1 SYSTEM LEAKAGE (CONT'D)

PLACEKEEPER
START DONE

d. Throttle the following valves as necessary to restore inventory: ☐
- HPSI COLD LEG INJECTION 1A (SI 225B)
- HPSI COLD LEG INJECTION 1B (SI 226B)
- HPSI COLD LEG INJECTION 2A (SI 227B)
- HPSI COLD LEG INJECTION 2B (SI 228B)
- HOT LEG 2 INJECTION FLOW CONTROL (SI 506B).

<u>OR</u>

2.2 For Shutdown Cooling Train B: ☐ ☐

a. Start Train A HPSI Pump. ☐

b. Close HPSI HEADER ORIFICE BYPASS (SI 219A). ☐

c. Open HOT LEG 1 INJECTION ISOLATION (SI 502A). ☐

d. Throttle the following valves as necessary to restore inventory: ☐
- HPSI COLD LEG INJECTION 1A (SI 225A)
- HPSI COLD LEG INJECTION 1B (SI 226A)
- HPSI COLD LEG INJECTION 2A (SI 227A)
- HPSI COLD LEG INJECTION 2B (SI 228A)
- HOT LEG 1 INJECTION FLOW CONTROL (SI 506A)

3. Restore <u>AND</u> maintain RCS level $\geq$15.13 feet, top of RCS Hot Leg. ☐ ☐

4. Monitor RCS Hot Leg for saturation conditions <u>AND</u> determine RCS heatup rate using <u>EITHER</u>: ☐ ☐

- CETs

<u>OR</u>

- <u>IF</u> CETs <u>NOT</u> available, <u>THEN</u> refer to Attachment 2: Calculated RCS Time to Boil.

E1 SYSTEM LEAKAGE (CONT'D)

PLACEKEEPER
START DONE

5. IF NO LPSI Pump is operating, THEN perform the following: ☐ ☐

   a. Vent suction piping of LPSI Pump that will take suction on Hot Leg with operating HPSI Pump. ☐

   b. Place Shutdown Cooling Train in service in accordance with OP-009-005, SHUTDOWN COOLING SYSTEM. ☐

   c. Continue venting until all air is removed. ☐

6. IF source of system leakage is known AND isolated, THEN go to step 17.

7. IF additional RCS makeup is required, THEN start second available HPSI Pump in accordance with step 2. ☐ ☐

   7.1 IF additional RCS makeup is required, THEN start third available HPSI Pump. ☐

8. Available Charging Pumps may be started to provide additional RCS makeup. ☐ ☐

CAUTION

IF RCS IS OPEN FOR MAINTENANCE, THEN FILLING MAY RESULT IN DISCHARGE FROM OPENING. THIS COULD ENDANGER PERSONNEL IN OR AROUND OPENINGS. CLOSED RCS IMPLIES NO MAJOR OPENINGS THAT WOULD PREVENT FILLING RCS HOT LEG. RCS IS CONSIDERED CLOSED WITH RCP SEAL REMOVED.

9. IF RCS level is NOT being restored by available HPSI AND Charging Pumps, THEN align LPSI Pump A or B for injection as follows: ☐ ☐

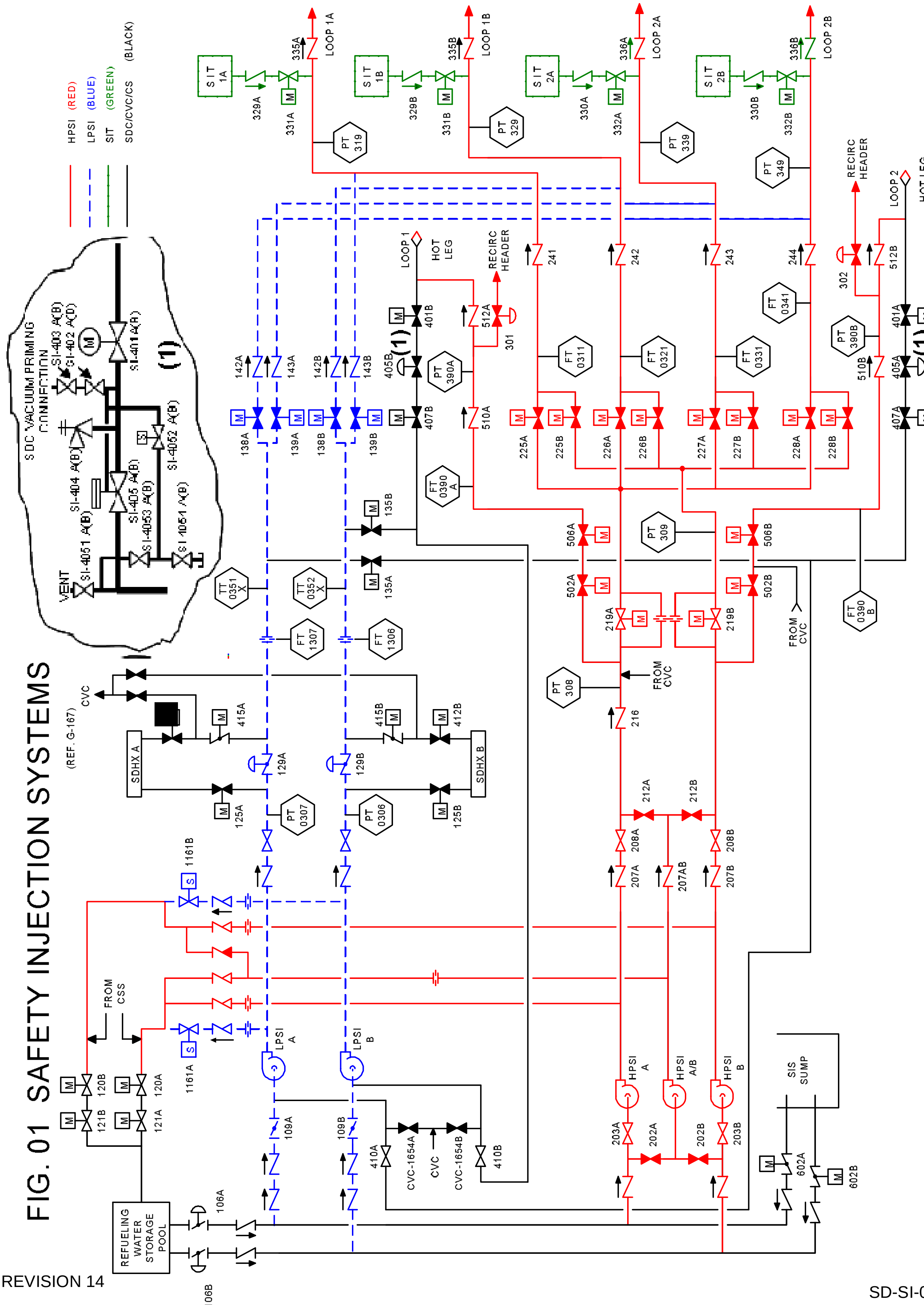
   9.1 For Train A: ☐ ☐

      a. Verify LPSI Pump A secured. ☐

      b. Locally open LPSI PUMP A SUCTION ISOLATION VALVE FROM RWSP (SI 109A). ☐

FIG. 01 SAFETY INJECTION SYSTEMS

(REF. G-167)



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	CE/E09 EA1.3	
	Importance Rating	3.6	

K/A Statement

EA1.3 Ability to operate and / or monitor the following as they apply to the (Functional Recovery): Desired operating results during abnormal and emergency situations.

Proposed Question: RO 27 Rev: 0

Given:

- Emergency Feedwater Pump ‘AB’ is tagged for maintenance
- REACTOR TRIP occurred due to a loss of main feedwater
- On the trip, all offsite and onsite AC power was lost
- Both steam generator levels indicate 0% WR
- Both steam generator pressures indicate 15 psia

Emergency Feedwater Pump B is now available. Which of the following describes the appropriate method for restoring feedwater?

A. Slowly restore feed to ONE SG.

B. Slowly restore feed to BOTH SGs.

C. Rapidly restore feed to ONE SG.

D. Rapidly restore feed to BOTH SGs.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: OI-038-000 step 5.4.50 states, If both Steam Generators become dry, slowly refill only one Steam Generator to reinitiate core cooling.
B. Incorrect: OI-038-000 step 5.4.50 states, If both Steam Generators become dry, slowly refill only one Steam Generator to reinitiate core cooling.
C. Incorrect: OI-038-000 step 5.4.50 states, If both Steam Generators become dry, slowly refill only one Steam Generator to reinitiate core cooling. This distractor is plausible because the applicant may assume that feeding a steam generator rapidly will more quickly restore an available steam generator.
D. Incorrect: OI-038-000 step 5.4.50 states, If both Steam Generators become dry, slowly refill only one Steam Generator to reinitiate core cooling. This distractor is plausible because the applicant may assume that feeding a steam generator rapidly will more quickly restore an available steam generator.

Technical Reference(s): OI-038-000 Revision 7
(Attach if not previously provided) OP-902-008 Revision 22
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE08 obj. 4 (As available)

Question Source: Bank # 1186-A
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2007 RO Make-up exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

5.4.50 Maintain SG Level

- Overfeeding of Steam Generators may cause excessive RCS cooldown. The operator should not add feedwater to a dry steam generator if another steam generator still contains water. Re-establish feedwater only to the Steam Generator that is not dry. If both Steam Generators become dry, slowly refill only one Steam Generator to reinitiate core cooling.

5.4.51 Maintain Shutdown Margin

- None

5.4.52 Maintain Shutdown Margin During the Cooldown

- Shutdown Margin is required to be maintained during the use of the EOPs with the following exception:
  - During the SBO and FRP procedure the reactor is only required to be maintained shutdown. This requirement is a 1%ΔK/K.
  - If unable to obtain an RCS boron sample or time does not permit performing a Shutdown Margin calculation, then Emergency Boration may be terminated under the following conditions:
    - Pressurizer level is approaching the upper end of the control band
    - HPSI Throttle Criteria is met
    - Reactor power is stable or dropping

    A Shutdown Margin calculation should be performed as soon plant conditions permit.
- Shutdown Margin is maintained with all CEAs fully inserted with Tc greater than 400°F.

5.4.53 Maintain Success Paths

- None

5.4.54 Makeup to the CSP

- ACCW flow to EFW should not be established until the CSP contents have depleted. The lineup should commence at 25% CSP level, however, ACCW flow to the suction of EFW should be completed before a CSP level of 11% is reached to prevent cavitation of the EFW Pumps.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K6.02	
	Importance Rating	2.7	

K/A Statement

K6.02 Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply

Proposed Question: RO 28 Rev: 0

OP-902-005, Station Blackout Recovery, directs the operator to verify closed the CCW non safety header containment isolation valves, CC-641, CC-710, ad CC-713. What is the reason for this?

A. To prevent running out the CCW pumps

B. To prevent thermal shocking the RCP seals

C. To prevent thermal shocking the CEDM coolers

D. To prevent water hammer in the Containment Building piping

Proposed Answer: B

Explanation: (Optional)

A. Incorrect. CCW valves to the RCPs are closed to ensure upon restoration of power the RCP seals are not thermally shocked when CCW is started. The CEDM coolers are get a supply from CCW inside containment.
B. **CORRECT**: CCW valves to the RCPs are closed to ensure upon restoration of power the RCP seals are not thermally shocked when CCW is started.
C. Incorrect. CCW valves to the RCPs are closed to ensure upon restoration of power the RCP seals are not thermally shocked when CCW is started.
D. Incorrect. CCW valves to the RCPs are closed to ensure upon restoration of power the RCP seals are not thermally shocked when CCW is started.

Technical Reference(s): OP-902-005 revision 17
(Attach if not previously provided) TG-OP-902-005 revision 307
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE05 obj. 7 (As available)

Question Source: Bank # 08389
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2007 NRC RO Makeup

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

INSTRUCTIONS | CONTINGENCY ACTIONS

Minimize RCS Leakage

* 7. Minimize RCS leakage:

a. Verify the following letdown containment isolation valves are closed:

- CVC 101, LETDOWN STOP VALVE
- CVC 103, LETDOWN ISOL VALVE
- CVC 109, LETDOWN ISOL VALVE

b. Verify the following RCP controlled bleedoff valves are closed:

- CVC 401, RCP BLEEDOFF
- RC 606, RCP BLEEDOFF

c. Verify the following CCW to RCP isolation valves are closed:

- CC 641, COMPONENT COOLING WATER RCP INLET OUTSIDE ISOL
- CC 710, COMPONENT COOLING WATER RCP OUTLET INSIDE ISOL
- CC 713, COMPONENT COOLING WATER RCP OUTLET OUTSIDE ISOL

Step Number 7 Minimize RCS Leakage

Objective

The intent of this step is to minimize loss of RCS inventory and prevent core uncovery, since there is no RCS make up capability during a SBO. Ensuring a minimal RCS leak rate reduces the rate of depressurization attributed to RCS leakage.

Instructions

Minimal RCS leakage can be met by ensuring the following are isolated:

- Letdown isolation valves
- RCP controlled bleedoff line isolation valves
- RCS sample lines isolation valves

All of the containment isolation valves listed above will fail closed on loss of power or air.

Letdown is assumed to isolate on high temperature after a loss of charging flow. The analysis assumes that letdown will not isolate for two minutes, but the actual isolation is approximately 15 seconds, after the loss of charging flow.

CCW valves to the RCPs are closed to ensure upon restoration of power the RCP seals are not thermally shocked when CCW is started.

Contingency Actions

None

Justification for Deviations

Waterford includes a plant specific deviation to close the CCW valves to the RCPs. When power is restored the CCW pump will start and CCW cooling may be restored to the RCPs. The valves are closed to ensure the operator has control of the evolution and to prevent thermally shocking the RCPs.

References

1. EC-M88-020, Station Blackout (SBO) Reactor Coolant System Inventory

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 A2.18	
	Importance Rating	3.1	

K/A Statement

A2.18 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High VCT level

Proposed Question: RO 29 Rev: 0

Given:

- A Plant startup is in progress
- The crew is purging the VCT to establish a Hydrogen blanket in accordance with section 8.19 of OP-002-005, Chemical and Volume Control Procedure
- The ATC is directed to raise VCT level to 90%

The crew will be required to align CVC-169 control switch to the ____(1)______ position to prevent CVC-169 from diverting to the ____(2)____ on a high level.

	(1)	(2)
A.	VCT	Boric Acid Condensate Tanks
B.	VCT	Hold-Up tanks
C.	Mid	Hold-up tanks
D.	Mid	Boric Acid Condensate Tanks

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: CVC-169 must be placed in the VCT position to prevent it from diverting to the Hold Up tanks at a high level of 76%. The Boric Acid Condensate Tanks are a part of the BM system, but are not the discharge path of CVC-169.
B. **CORRECT**: CVC-169 must be placed in the VCT position to prevent it from diverting to the Hold Up tanks at a high level of 76%.
C. Incorrect: CVC-169 must be placed in the VCT position to prevent it from diverting to the Hold Up tanks at a high level of 76%.
D. Incorrect: CVC-169 must be placed in the VCT position to prevent it from diverting to the Hold Up tanks at a high level of 76%. The Boric Acid Condensate Tanks are a part of the BM system, but are not the discharge path of CVC-169.

Technical Reference(s): OP-002-005 revision 43
(Attach if not previously provided) OP-500-007 attachment 4.3 revision 12
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CVC00 obj. 1 and 3 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

8.19 PURGING THE VCT WITH HYDROGEN TO ESTABLISH HYDROGEN BLANKET ON VCT (C)

NOTE

This section should only be performed if Volume Control Tank already has a nitrogen blanket.

8.19.1 Notify Chemistry and Radiation Protection Departments that this section is being performed.

8.19.2 Verify Gaseous Waste Management System in service in accordance with OP-007-003, Gaseous Waste Management.

8.19.3 Verify following valves Closed:

- NG-222 Volume Control Tank Nitrogen Supply Isolation Valve
- NG-224 Volume Control Tank Nitrogen Pressure Regulator Outlet

8.19.4 Verify VCT makeup is not in the Auto Makeup mode.

8.19.5 Open Excess Flow Control Valve Bypass, HG-124.

CAUTION

OPERATION WITH RCP CONTROL BLEEDOFF PRESSURE GREATER THAN 65 PSIG MAY INCREASE RCP SEAL FACE WEAR WITHOUT SUFFICIENT RCS PRESSURE. REFER TO RCS PRESSURE AND TEMPERATURE LIMITS GRAPH (ATTACHMENT 11.1) IN OP-001-002, REACTOR COOLANT PUMP OPERATION.

8.19.6 Open Volume Control Tank Hydrogen Regulator Inlet Isolation, HG-202.

8.19.7 Ensure RCP Control Bleedoff Pressure is maintained throughout the remainder of this Section as follows:

8.19.7.1 Verify RCP Control Bleedoff Pressure is maintained 30 PSIG to 120 PSIG (RC-IPI-0215 or PID A39402).

8.19.7.2 If RCP Control Bleedoff Pressure BPCV, CVC-4063, is not automatically maintaining Control Bleedoff backpressure 30 PSIG to 120 PSIG, then perform Section 8.24, Manual Control of Reactor Coolant Pump Control Bleedoff.

8.19.8 Monitor Plant Stack activity and Gas Surge Tank pressure while venting the VCT in Steps 8.19.9 and 8.19.10.

8.19.9 Cycle VCT Vent To Gas Surge Header, GWM-112, as necessary to establish VCT pressure as low as possible within the band of 15 PSIG to 65 PSIG.

8.19.10 Cycle VCT Vent To Gas Surge Header, GWM-112, as necessary to maintain VCT pressure as low as possible within the band of 15 PSIG to 65 PSIG while performing the following:

8.19.10.1 Place VCT Inlet/Bypass to Holdup Tanks, CVC-169, to VCT.

8.19.10.2 Makeup to VCT at desired boron concentration to raise VCT level to 90% - 95%.

8.19.11 Lower VCT level to desired level as follows:

8.19.11.1 Place VCT Inlet/Bypass to Holdup Tanks, CVC-169, to BMS.

8.19.11.2 Throttle Volume Control Tank Hydrogen Regulator Outlet Isolation, HG-204, as necessary to maintain VCT pressure within the band of 15 PSIG to 65 PSIG.

8.19.11.3 When desired VCT level is reached, then align VCT Inlet/Bypass to Holdup Tanks, CVC-169, to the VCT by placing control switch to AUTO.

8.19.12 When VCT pressure is restored to the desired pressure, then Close Volume Control Tank Hydrogen Regulator Outlet Isolation, HG-204.

8.19.13 Notify Chemistry Department to sample VCT gas space for hydrogen concentration.

8.19.14 If VCT gas space sample does not meet hydrogen concentration requirements, then repeat Steps 8.19.8 through 8.19.13 as necessary to achieve required hydrogen concentration.

8.19.15 Close Volume control Tank Hydrogen Regulator Inlet Isolation, HG-202.

8.19.16 Close Excess Flow Control Valve Bypass, HG-124.

9.0 AUTOMATIC FUNCTIONS

9.1	Charging Pump A(AB)(B) Suction Press Low Trip (CVC-IPS-0224X(CVC-IPS-0224Y)(CVC-IPS-0224Z)).....	10 PSIA (R: 14 PSIA)
9.2	Ion Exchanger Bypass, CVC-140, diverts to bypass Purification Ion Exchangers (CVC-ITAC-0224)	140°F (R: 134.4°F)
9.3	VCT Makeup (CVC-ILAL-0226)	36.7% (47") (R: 50% (64"))
9.4	VCT Lo-Lo Level Divert to RWSP Suction (CVC-ILAL-0227)	5.5% (7") (R: 7.8% (10"))
9.5	VCT Divert to BMS (CVC-ILAL-0227)	75.8% (97") (R: 73.4% (94"))
9.6	Letdown Isolation on Regenerative Heat Exchanger Tube Outlet Temperature High (CVC-ITAC-0221)	470°F (R: Manual)
9.7	CC-636 Closes to Isolate CC Flow to Letdown Heat Exchanger...	Whenever CVC-103 <u>or</u> CVC-109 is Closed
9.8	Boric Acid Makeup Tank A and B Heater Control (BAM-ITIC-0206, 0207, 0208, 0209)	
	On:	100°F (R: 103°F)
	Off:	110°F (R: 106°F)
9.9	Charging Pump A(AB)(B) Low Oil Press Trip (CVC-IPS-0229X(CVC-IPS-0229Y)(CVC-IPS-0229Z))	10 PSIG (30 sec delay) (R: 17 PSIG)

VCT LEVEL HI/LO

INITIATING DEVICE	SETPOINT
CVC-ILAC-226	100" (R: 97") [(78.1%) (75.8%)] 44" (R: 47") [(34.4%) (36.7%)]

POSSIBLE EFFECTS

1. Charging pump suction auto shifted to RWSP (5.5%).
2. Letdown auto diverts to BM System (76%).
3. Malfunction of Auto Make up to VCT.

CONTROL ROOM INDICATIONS	LOCAL INDICATIONS
CVC-ILI-226, VCT Level	NONE
VCT Divert Valve, CVC-169, position VCT Makeup Valve, CVC-510, position VCT Outlet, CVC-183, position	
Charging Pump Suct From RWSP, CVC-507, position	
A39400 CVCS VCT WTR LVL 1 A39401 CVCS VOL CONT TK LVL 2	

VCT LEVEL HI/LO

POSSIBLE CAUSES	RECOMMENDED ACTIONS
1. Failure of PZR LVL Control System.	1.1 Go to OP-901-110, Pressurizer Level Control Malfunction.
2. Excessive plant heatup or cooldown.	2.1 Reduce plant heatup or cooldown rate.
3. Charging or Letdown Malfunction.	3.1 Go to OP-901-112, Charging/Letdown Malfunction.
4. Instrument Failure	4.1 Go to OP-901-113, Volume Control Tank Makeup Control Malfunction
	4.2 If gas binding of any Charging Pump is suspected, then refer to OP-901-112, Charging and Letdown Malfunction.

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

K/A Statement

K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger

Proposed Question: RO 30 Rev: 0

Given:

- Shutdown Cooling (SDC)Train B is in service
- Component Cooling Water (CCW) to SDC Heat Exchanger B was lost 10 minutes ago
- Reactor Coolant System temperature is 220 degrees
- The crew has entered OP-901-131, Shutdown Cooling Malfunction
- Low Pressure Safety Injection Pump B has been secured

Per OP-901-131, Shutdown Cooling Malfunction, the crew must reinitiate ____(1)____ flow to SDC Heat Exchanger B slowly to prevent damage to _____(2)_____ .

	(1)	(2)
A.	Component Cooling Water	Safety Injection piping
B.	Component Cooling Water	Component Cooling Water Piping
C.	Shutdown Cooling	Safety Injection piping
D.	Shutdown Cooling	Component Cooling Water Piping

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: The Caution in OP-901-131 section E3 directs the crew to restore CCW flow slowly to the SDC heat exchanger to prevent damage to CCW piping.
B. **CORRECT:** The Caution in OP-901-131 section E3 directs the crew to restore CCW flow slowly to the SDC heat exchanger to prevent damage to CCW piping.
C. Incorrect: The Caution in OP-901-131 section E3 directs the crew to restore CCW flow slowly to the SDC heat exchanger to prevent damage to CCW piping. There are no restrictions on restoring SDC flow in OP-901-131.
D. Incorrect: The Caution in OP-901-131 section E3 directs the crew to restore CCW flow slowly to the SDC heat exchanger to prevent damage to CCW piping. There are no restrictions on restoring SDC flow in OP-901-131.

Technical Reference(s): OP-901-131 revision 303
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-REQ21 obj. 6 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

E3. Loss of Shutdown Cooling Heat Removal Capability

	PLACEKEEPER START	DONE
1. IF CCW Pump has tripped, THEN start standby CCW pump.	☐	☐
2. Verify shutdown cooling train with operating CCW pump is in service in accordance with OP-009-005, SHUTDOWN COOLING SYSTEM.	☐	☐
3. Implement OP-901-510, COMPONENT COOLING WATER SYSTEM MALFUNCTION, concurrently with this procedure.	☐	☐

CAUTION

RAPID RESTORATION OF CCW FLOW WHEN SHUTDOWN COOLING HEAT EXCHANGER TEMPERATURE IS >200°F MAY CAUSE DAMAGE TO CCW PIPING.

4. IF CCW flow is NOT restored to affected Shutdown Cooling Heat Exchanger within 5 minutes AND RCS temperature is >200°F, THEN perform the following:	☐	☐
4.1 Verify associated Shutdown Cooling Heat Exchanger Outlet valve in SETPNT:		☐

- SHDN HX A OUTLET valve (CC 963A)

OR

- SHDN HX B OUTLET valve (CC 963B)

4.2 Close Temperature Control valve for affected Train:		☐

- SDCS LOOP 2 TEMPERATURE CONTROL (SI 415A)

OR

- SDCS LOOP 1 TEMPERATURE CONTROL (SI 415B)

4.3 Place associated LPSI Header Flow controller to MAN AND adjust output to 10%.		☐

- LPSI B DISCH HDR FLOW controller (SI IFIC 0306)

OR

- LPSI A DISCH HDR FLOW controller (SI IFIC 0307)

E3. LOSS OF SHUTDOWN COOLING HEAT REMOVAL CAPABILITY (CONT'D)

PLACEKEEPER
START DONE

4.4 Open associated Warmup valve: ☐

- SDCS LOOP 2 WARMUP VALVE (SI 135A)

OR

- SDCS LOOP 1 WARMUP VALVE (SI 135B)

4.5 Close associated LPSI Flow Control valves: ☐

- LPSI FLOW CONTROL COLD LEG 2A (SI 139A)
- LPSI FLOW CONTROL COLD LEG 2B (SI 138A)

OR

- LPSI FLOW CONTROL COLD LEG 1A (SI 139B)
- LPSI FLOW CONTROL COLD LEG 1B (SI 138B)

4.6 Stop associated LPSI Pump. ☐

4.7 Locally isolate CCW flow to Shutdown Cooling Heat Exchanger by closing: ☐

- SHUTDOWN HEAT EXCHANGER A CCW INLET ISOLATION (CC 949A)
- SHUTDOWN HEAT EXCHANGER A CCW FCV DOWNSTREAM ISOL (CC 965A)

OR

- SHUTDOWN HEAT EXCHANGER B CCW INLET ISOLATION (CC 949B)
- SHUTDOWN HEAT EXCHANGER B CCW FCV DOWNSTREAM ISOL (CC 965B)

E3. LOSS OF SHUTDOWN COOLING HEAT REMOVAL CAPABILITY (CONT'D)

PLACEKEEPER
START DONE

4.8 WHEN CCW flow is restored to affected CCW header, THEN restore CCW flow to Shutdown Cooling Heat Exchanger by the following: ☐ ☐

- For Shutdown Cooling Heat Exchanger A, slowly open:
  1) SHUTDOWN HEAT EXCHANGER A CCW INLET ISOLATION (CC 949A) ☐
  2) SHUTDOWN HEAT EXCHANGER A CCW FCV DOWNSTREAM ISOL (CC 965A)

OR

- For Shutdown Cooling Heat Exchanger B, slowly open: ☐
  1) SHUTDOWN HEAT EXCHANGER B CCW INLET ISOLATION (CC 949B)
  2) SHUTDOWN HEAT EXCHANGER B CCW FCV DOWNSTREAM ISOL (CC 965B)

5. IF it is suspected that a Shutdown Cooling Heat Exchanger has become air bound, THEN locally vent the Heat Exchanger with the following valves: ☐ ☐
   - SHUTDOWN COOLING HEAT EXCHANGER A SHELL VENT (CC 951A)
   - SHUTDOWN COOLING HEAT EXCHANGER A SHELL VENT (CC 9511A)

   OR

   - SHUTDOWN COOLING HEAT EXCHANGER B SHELL VENT (CC 951B)
   - SHUTDOWN COOLING HEAT EXCHANGER B SHELL VENT (CC 9511B)

6. WHEN Shutdown Cooling Heat Removal Capability is restored, THEN align Shutdown Cooling in accordance with OP-009-005, SHUTDOWN COOLING SYSTEM. ☐ ☐

7. WHEN Shutdown Cooling Heat Removal Capability is restored, THEN align Containment Cooling Fans in accordance with OP-008-003, CONTAINMENT COOLING SYSTEM. ☐ ☐

END

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K3.02	
	Importance Rating	4.3	

K/A Statement

K3.02 Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: Fuel

Proposed Question: RO 31 Rev: 0

Following a large break LOCA with a failure of ESFAS to automatically initiate a SIAS, the Technical Specification Safety Limit for the ____(1)_____ temperature could exceed its melting threshold of _______(2)______ degrees without operator intervention.

	(1)	(2)
A.	cladding	5080
B.	cladding	3350
C.	fuel centerline	5080
D.	fuel centerline	3350

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: Technical Specification safety limit 2.1.1.2 states that peak fuel centerline temperature shall be maintained less than 5080°F. Cladding melt is mentioned in the basis for this safety limit but the question is specific to the knowledge of what the safety limit is (fuel centerline temperature)
B. Incorrect**:** Technical Specification safety limit 2.1.1.2 states that peak fuel centerline temperature shall be maintained less than 5080°F. 3350 °F is the melting point for the cladding. Cladding melt is mentioned in the basis for this safety limit but the question is specific to the knowledge of what the safety limit is (fuel centerline temperature)
C. **CORRECT**: Technical Specification safety limit 2.1.1.2 states that peak fuel centerline temperature shall be maintained less than 5080°F.
D. Incorrect: Technical Specification safety limit 2.1.1.2 states that peak fuel centerline temperature shall be maintained less than 5080°F. 3350 °F is the melting point for the cladding.

Technical Reference(s): TS 2.1
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-TS00 obj. 3 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 2
55.43 ____

Comments:

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.24.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.24, be in HOT STANDBY within 1 hour.

PEAK FUEL CENTERLINE TEMPERATURE

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A.)

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak fuel centerline temperature has equaled or exceeded 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A), be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4, and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

BASES

2.1.1 REACTOR CORE

→(DRN 02-458)

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the peak fuel centerline temperature below the melting point.

←(DRN 02-458)

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

→(EC-18510, Ch. 64)

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to 1.24 for the WSSV-T and ABB-NV correlations and is established as a Safety Limit. This value is based on a statistical combination of uncertainties. It includes uncertainties in the Critical Heat Flux (CHF) correlation, allowances for rod bow and hot channel factors (related to fuel manufacturing variations) and allowances for other hot channel calculative uncertainties (CEN-356(V)-P-A, “Modified Statistical Combination of Uncertainties,” Revision 01-P-A, May 1988).

←(EC-18510, Ch. 64)

→(DRN 02-458)

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, fuel centerline melting is established as a Safety Limit. The design melting point of new fuel with no burnable poison is 5080°F. The melting point is adjusted downward from this temperature depending on the amount of burnup and amount and type of burnable poison in the

←(DRN 02-458)

2.1.1 REACTOR CORE (Continued)

• **(DRN 02-458)**

fuel. The 58°F per 10,000 MWD/MTU adjustment for burnup was accepted by the NRC in Topical Report CEN-386-P-A, “Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel,” August 1992. Adjustments for burnable poisons are established based on NRC approved Topical Report CENPD-382-P-A, “Methodology for Core Designs Containing Erbium Burnable Absorbers,” August 1993.

A steady state peak linear heat rate of 21 kW/ft has been established as the Limiting Safety System Setting to prevent fuel centerline melting during normal steady state operation. Following design basis anticipated operational occurrences, the transient linear heat rate may exceed 21 kW/ft provided the fuel centerline melt temperature is not exceeded.

• **(DRN 02-458)**

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1974 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. RPS Trip Setpoints values are determined by means of an explicit setpoint calculation analysis.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K4.16	
	Importance Rating	3.2	

K/A Statement

K4.25 Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Interlocks between RHR valves and RCS

Proposed Question: RO 32 Rev: 0

Which of the following describes the interlocks associated with SI-401A, RC Loop 2 SHDN Cooling Upstream Isolation?

A. Can **only** be opened if RCS pressure is $\leq$ 358 psia and RCS temperature is less than 350° F.

B. Can **only** be opened if RCS pressure is $\leq$ 358 psia and can only be closed if LPSI pump A is off.

C. Can **only** be opened if RCS pressure is $\leq$ 386 psia and can only be closed if LPSI pump A is off.

D. Can **only** be opened if RCS pressure is $\leq$ 386 psia and RCS temperature is less than 350° F.

Proposed Answer: C

Explanation: (Optional)

A: Incorrect: If a harsh environment exists in containment, OP-009-005 provides an administrative requirement that SI-401A should not be opened until RCS pressure is $\leq$ 358 psia. An RCS temperature less than 350° F is an administrative requirement for opening SI-401A.

B: Incorrect: If a harsh environment exists in containment, OP-009-005 provides an administrative requirement that SI-401A should not be opened until RCS pressure is $\leq$ 358 psia. Part 2 is correct

C: **CORRECT:** SI-405A can only be opened if RCS pressure is $\leq$ 386 psia and can only be closed if LPSI pump A is off..

D. Incorrect: The administrative requirements for opening SI-401 A is RCS pressure is $\leq$ 386 psia and RCS temperature is less than 350° F.

Technical Reference(s): SD-SDC page 11
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-SDC00 obj 4 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

MAJOR VALVES ASSOCIATED WITH SDC SYSTEM

RCS Loop SDC System Isolation Valves (SI-401A(B), SI-405A(B), and SI-407A(B))

SI-401A and B are motor operated 14” gate valves. The valves have a backup power supply for OPEN/CLOSE indication, since the normal power breaker is locked open while in Modes 1-4 (>1750 PSIA) to prevent inadvertent operation. The valves can only be opened if RCS pressure less than or equal to 386 PSIA. The LPSI Pump must be off to close SI-401A(B) from the control panels. The valves are controlled from CP-8 and LCP-43. The valves have a 3-position keylock switch for positioning the valves either open or closed at CP-8. EC9720 changed the stroke logic to allow the valve to be “bumped” open to minimize any potential void induced water hammer. Once partially opened, the operator then holds the CS in OPEN to fully open SI-401. The key can only be removed in the CLOSED position. Valve position is controlled by OPEN/CLOSE hand switches at LCP-43. SI-401 stroke time is approximately 60 seconds.

SI-405A and B are pneumatic piston operated 14” gate valves and fail AS-IS. When in Mode 5, the valve actuator is normally gagged open to prevent inadvertent closure. Connections to the Instrument Air system and local accumulators ensure sufficient air is stored to operate each SI-405. A minimum air pressure of 55 psig is required to stroke the SI-405 valves. These valves are required for containment isolation and are the class boundary separating the Safety Class 1 RCS SDC line from the Safety Class 2 LPSI Pump suction piping. They are interlocked with Pressurizer pressure to prevent over pressurization of the LPSI Pump suction piping. Normally, the valves can only be opened if Pressurizer pressure is less than or equal to 386 PSIA. Two key switches mounted locally can be used to bypass the SI-401B and SI-405B pressure interlocks; but bypassing the pressure interlock is not normally performed. The valves are operated from either the Control Room (CP-8) or the Remote Shutdown Panel (LCP-43). The valves have a keylock switch for OPEN/LOCKED CLOSED positions at CP-8. The key can only be removed in the LOCKED CLOSED position. Valve position is also controlled by OPEN/CLOSE handswitches at LCP-43, which bypass the pressure interlock when enabled. SI-405 stroke time is approximately 300 seconds by design, but may take as long as 15 minutes to fully open. EC13462 modified the SI-405 valves to prevent hydraulic pressure lock from preventing these valves from opening. New pressure equalizing lines 1SI ¾-354 &1SI ¾-355 connect to existing vent/vacuum lines, 1SI1-34B and 1SI1-32A [upstream of valves SI-402A(B)]. In order for the bonnet cavity to connect to the new equalizing line the packing was modified. The lower packing was exchanged with a spacer ring. The new spacer ring and original lantern ring will create a flow path from the bonnet cavity to the pressure equalizing line. The upper packing is preformed grafoil with composite end rings, against the gland and lantern ring. Both the packing gland and lantern ring are the same as the original design.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K1.03	
	Importance Rating	3.0	

K/A Statement

K1.03 Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: RCS

Proposed Question: RO 33 Rev: 0

A Steam Generator Tube Rupture has occurred that resulted in an automatic SIAS/CIAS.

Which of the following could result in a Quench Tank Rupture Disc failure and rising containment pressure, due to **automatic** alignment to the Quench Tank?

A. RCP Control Bleed Off

B. RCP Vapor Seal Leak Off

C. Reactor Head Vent Header

D. Pressurizer Vent Header

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** RC-606, RCP Control Bleedoff Inside Containment Isolation Valve closes on a CIAS, redirecting RCP control bleedoff to the quench tank through a relief valve.
B. Incorrect: The RCP vapor seal leakoff is directed to the Reactor Drain Tank at all times.
C. Incorrect: The Reactor head vent must be aligned to the quench tank manually.
D. Incorrect: The Pressurizer vent must be aligned to the quench tank manually.

Technical Reference(s): OP-902-009 Appendix 4 revision 309
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-RCS00 obj. 2 (As available)

Question Source: Bank # 08393
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2007 NRC RO Makeup Exam

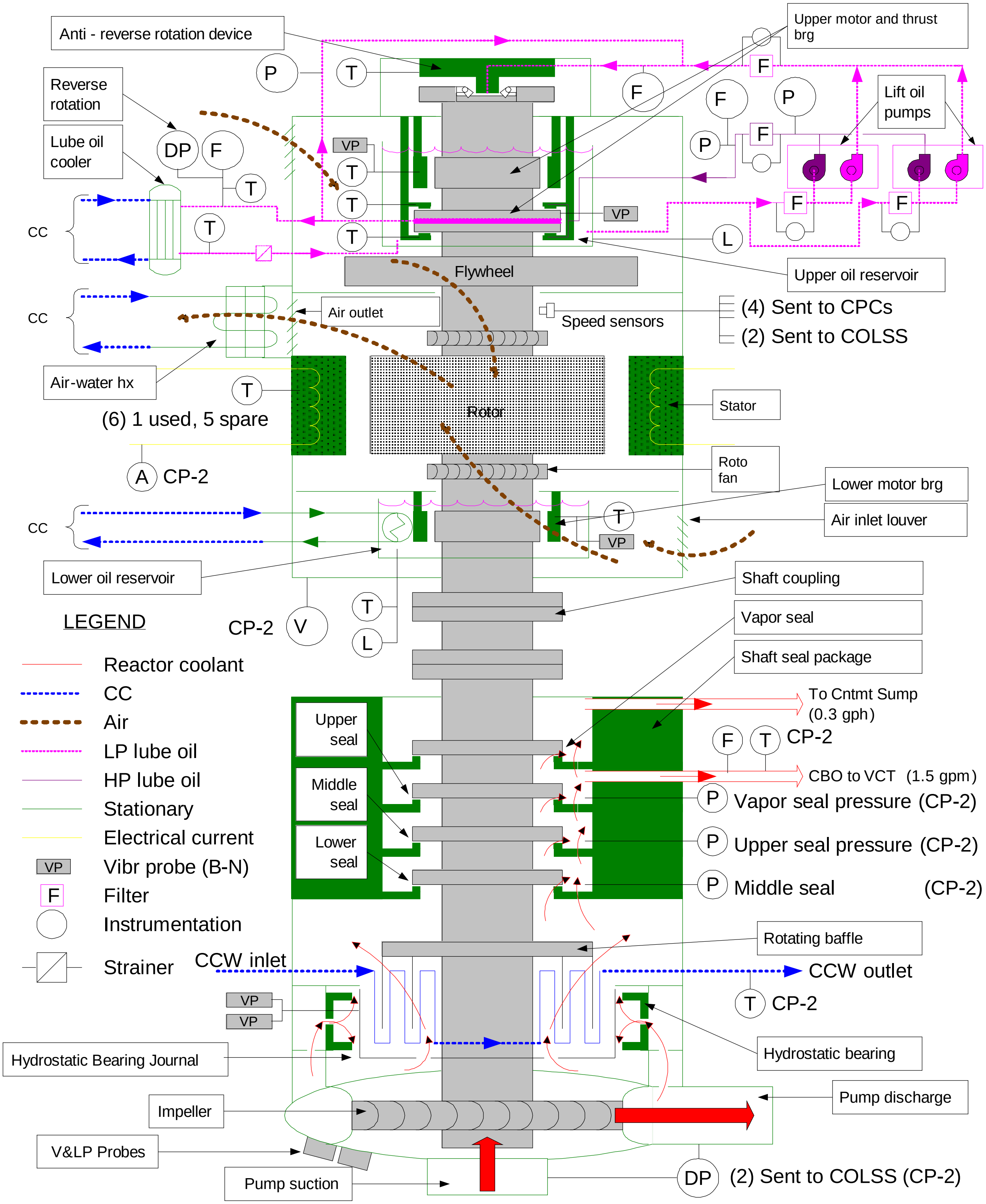
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 3
55.43

Comments:

Component Number	Component Noun Name	Location	Action	Verified
CP-8 Train B				
MS 120B	Main Steam Line 2 Drains Normal	CP-8	Close	
MS 119B	Main Steam Line 2 Drains Bypass	CP-8	Close	
PSL 105	RCS Sample Isol Hot Leg (In)	CP-8	Close	
PSL 203	RCS Sample Isol Pzr Surge (In)	CP-8	Close	
PSL 303	RCS Sample Isol Pzr Isol Vlv (In)	CP-8	Close	
BD 102A	SG Blowdown Isol Stm Gen 1 (In)	CP-8	Close	
BD 102B	SG Blowdown Isol Stm Gen 2 (In)	CP-8	Close	
SSL 8004A	Sampling Isolation SG 1	CP-8	Close	
SSL 8004B	Sampling Isolation SG 2	CP-8	Close	
SP 105	Cntmt Isolation Sump Pumps Outlet	CP-8	Close	
FP 601B	Cntmt Isolation Fire Water B	CP-8	Close	
NG 157	Cntmt Isolation Nitrogen	CP-8	Close	
CP-4 Train B				
CVC 103	Cntmt Isol Valves Letdown Isol Valve	CP-4	Close	
GWM 104	Cntmt Isol Valves Waste Gas	CP-4	Close	
RC 606	Cntmt Isol Valves RCP Bleedoff	CP-4	Close	
BM 109	Cntmt Isol Valves Reactor Drain Tank	CP-4	Close	

FI . 16 REACTOR COOLANT PUMP



REVISION 20 SD-RCS-16

FIG. 29 QUENCH TANK

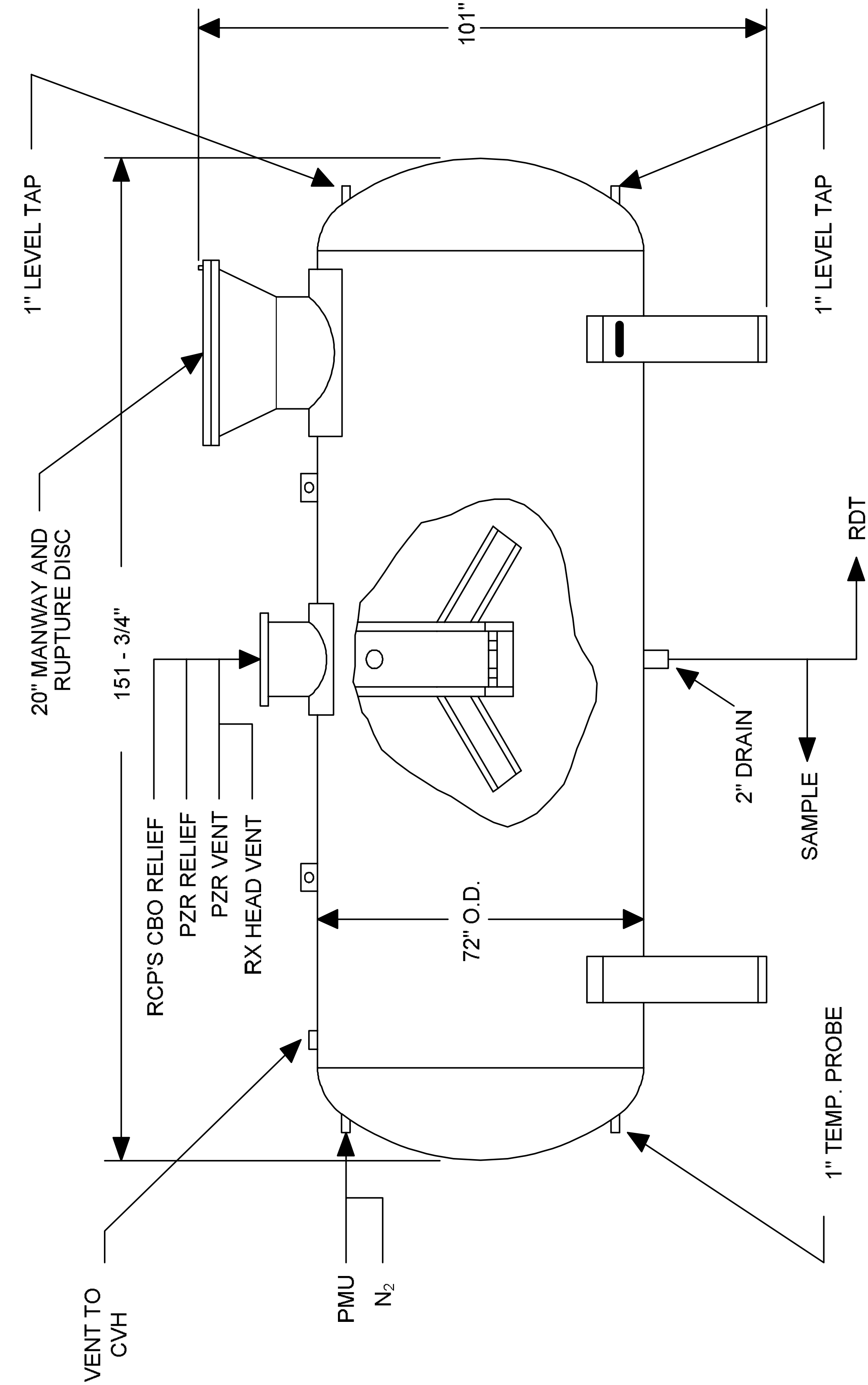


FIG. 31 REACTOR COOLANT SYSTEM VENTING SYSTEM

(REF. G-172)

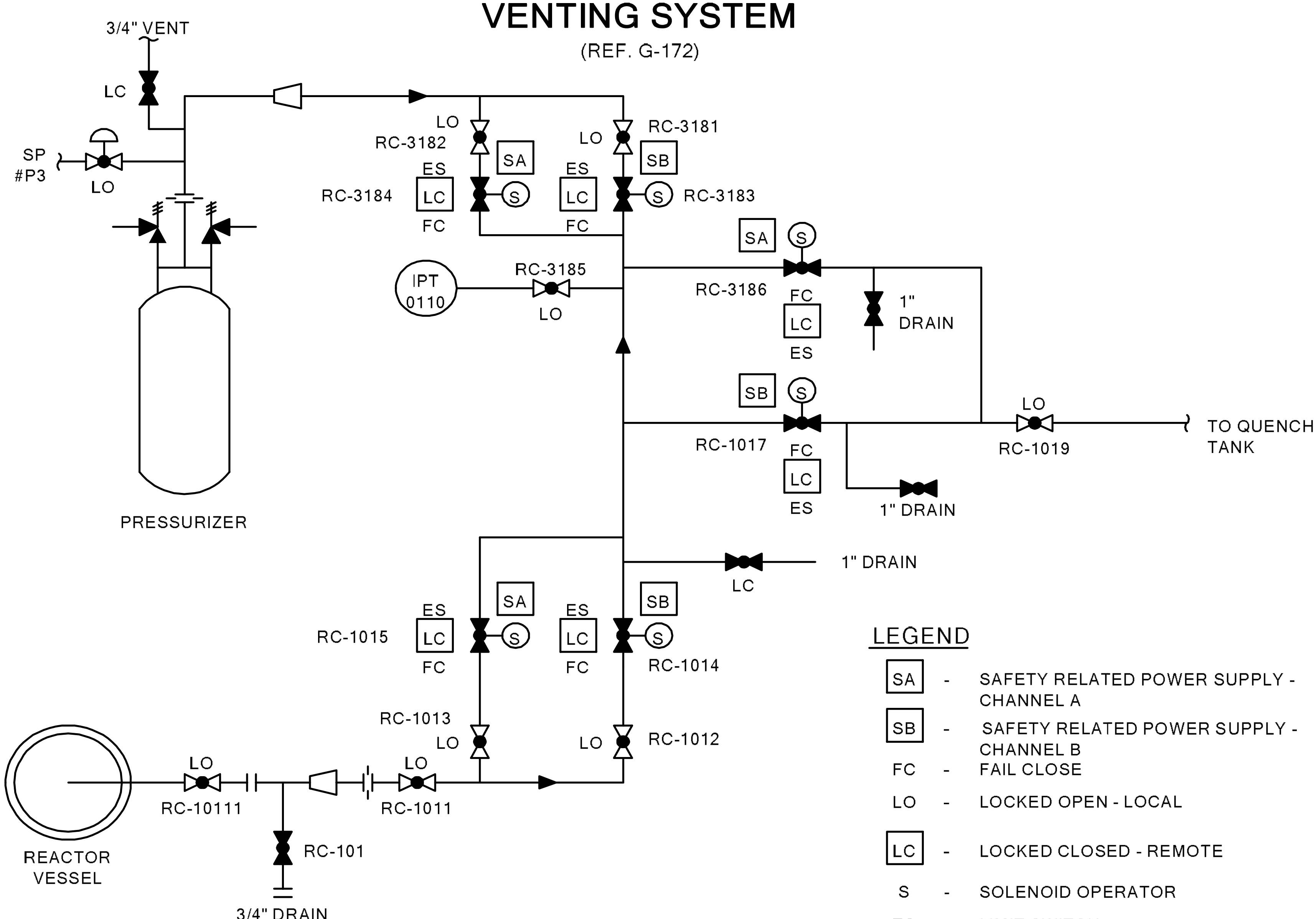
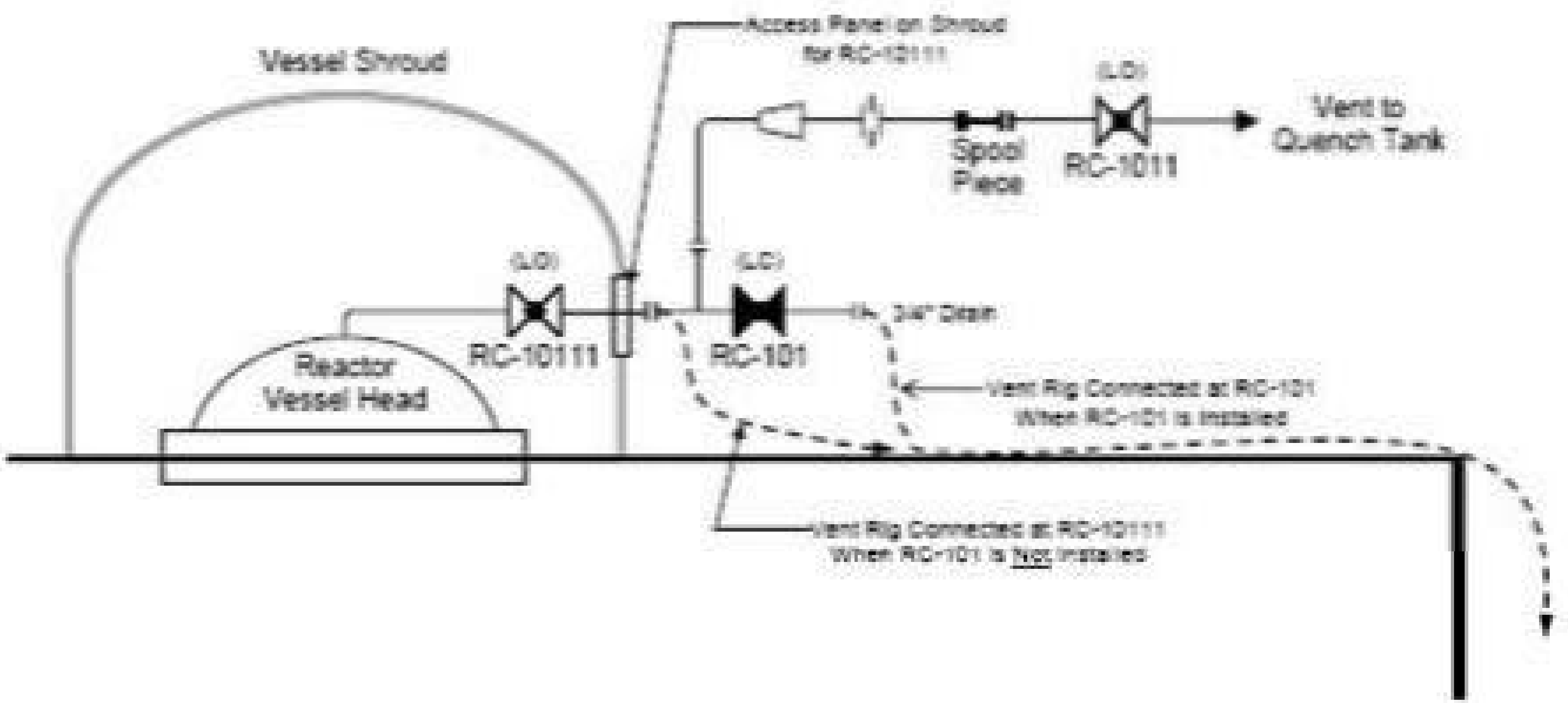


FIG. 48 RCS REACTOR VESSEL HEAD VENT RIG CONNECTION

(REF. OP 001-003)



makeup to the VCT by taking the control switch for BAM-143 to CLOSE and then opening the VCT makeup stop valve (CVC-510).

RCP CONTROLLED BLEEDOFF

Excess Flow Check Valves

Each operating RCP delivers approximately 1.5 gpm of seal bleedoff to the VCT. An excess flow check valve is provided in each RCP bleedoff line (RC-409A, RC-409B, RC-509A, and RC-509B). The valves are spring actuated to hold the valve disc open. On high forward flows (10 gpm) which might occur as a result of a reactor coolant pump gross seal failure, the fluid force is sufficient to overcome the spring thereby closing the valve. This action minimizes the reactor coolant inventory loss.

Reactor Coolant Pump Controlled Bleedoff Header Pressure, CVC-IPT-0215

A pressure measurement channel is provided to measure the pressure at the reactor coolant pump controlled bleedoff combined header. Indication and alarm annunciation is provided on CP-4 (0-300 psig). The HI alarm is set at 120 psig and may indicate that a valve in the line to the VCT has been improperly throttled or closed. The HI HI alarm, which is set at 250 psig, may indicate that the controlled bleedoff flow has stopped due to a valve closure or that a reactor coolant pump seal failure has occurred. Plant computer alarm and indication is also provided.

RCP Controlled Bleedoff Relief Stop Valve (RC-602)

Upstream of RC-603, is RC-602, the RCP Controlled Bleedoff Relief Stop Valve. This valve is keylocked open in the main control room on CP-4. RC-602 is a 3/4 inch spring opened, air diaphragm closed globe valve that fails open on a loss of air or power.

It is used to isolate the RCS leakage through the controlled bleedoff relief valve in the event of a LOCA. This action would minimize the loss of RCS inventory.

<u>RCP Controlled Bleedoff Header Relief Valve (RC-603)</u>

The relief valve RC-603 in the RCP controlled bleedoff header is located in containment and the valve allows the controlled bleedoff flow to continue to the Quench Tank in the event that a valve in the line to the VCT is closed. RC-603, with a 150 psig set point and in conjunction with the administrative control of valve RC-602, is credited

in providing over pressure protection for the piping between CVC-4063, RCP Controlled Bleedoff PCV, and the piping upstream that is rated for 2485 psig.

RC-603 is sized to pass the flow rate required to ensure closure of one excess flow check valve in the event of a RCP seal failure plus the normal bleedoff from the other reactor coolant pumps. The relief valve opening pressure of 150 psig is less than the controlled bleedoff high-high pressure alarm pressure.

RCP Controlled Bleedoff Containment Isolation Valves (RC-606 and CVC-401)

The normal bleedoff flow from all 4 RCPs joins together to form a common header. There are two containment isolation valves, one on either side of the containment wall. Valve RC-606 is located inside containment and CVC-401 is located outside containment. Both of these valves are 3/4-inch globe, spring closed, air diaphragm opened valves. They fail closed on loss of air or power.

RC-606 and CVC-401 are controlled from the main control room on CP-4 and are closed automatically by a CIAS signal. The CIAS signal can be overridden from CP-4. Normal flow is continuously directed to the VCT.

Containment Penetration 44 Thermal Relief, RC-6061

RC-6061 is a one inch relief valve designed to protect the RCP Controlled Bleedoff containment penetration from overpressure due to heating up the fluid between the two RCP Controlled Bleedoff Containment Isolation Valves, RC-606 and CVC-401. This could occur post accident when the RCP Controlled Bleedoff line is isolated at a low temperature and then heated up by the high temperature post accident environment seen following a LOCA or MSLB inside Containment. RC-6061 is set to lift at 2500 psig and discharges into Containment.

RCP Controlled Bleedoff Bypass (CVC-4061)

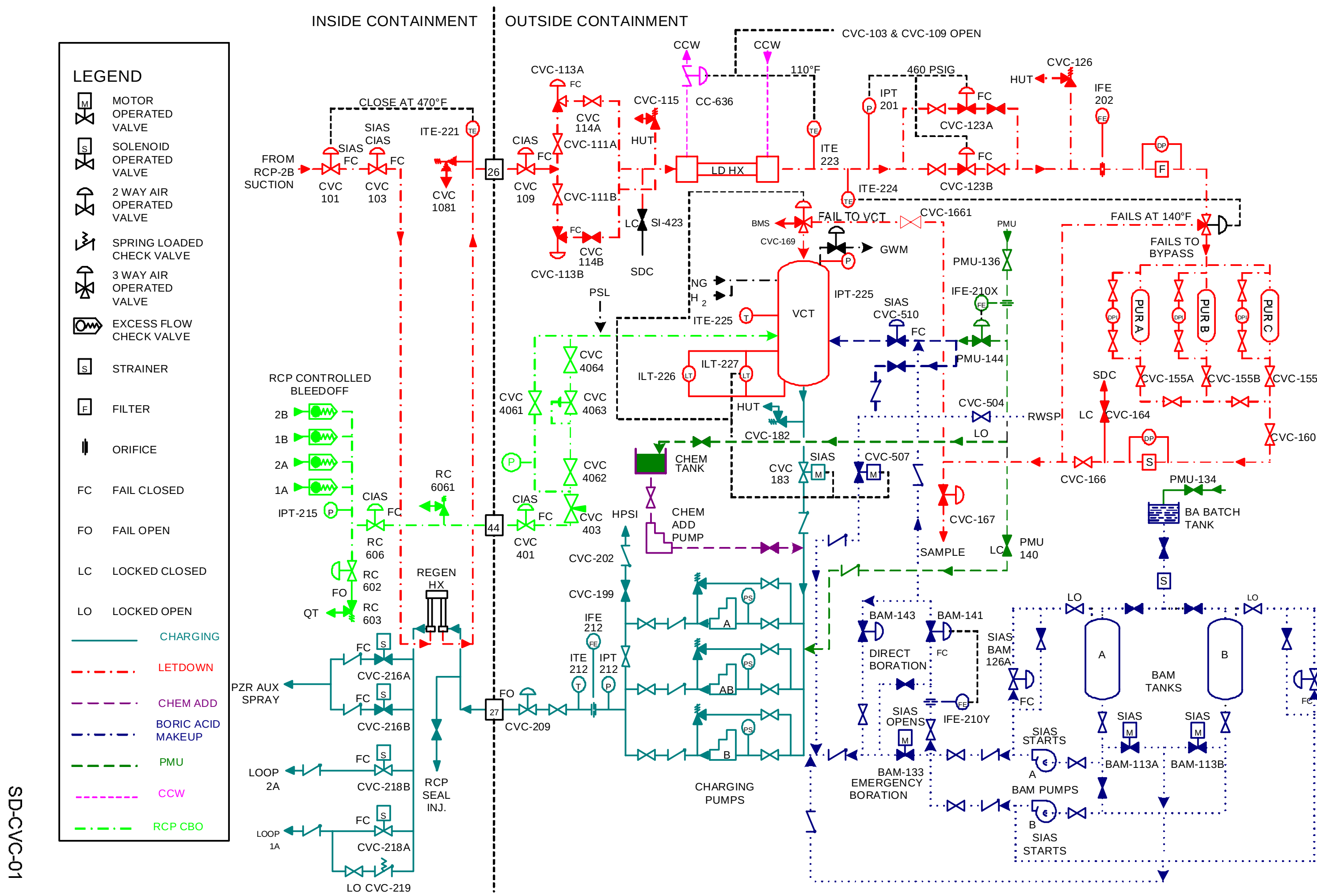
CVC–4061 is a 3/4 inch manual valve. This valve is throttled during operation to ensure minimum RCP Control Bleedoff Flow in the event CVC-4063, RCP Controlled Bleedoff Backpressure Control Valve fails closed.

RCP Controlled Bleedoff Backpressure Control Valve (CVC-4063)

CVC-4063 is a direct-acting 3/4 inch pressure control valve. The sensing line senses pressure upstream of the valve. The valve starts to open at 31 psig backpressure. The valve should maintain 40-65 psig backpressure on the RCP seals during normal

FIG. 01 CHEMICAL AND VOLUME CONTROL SYSTEM

(REF. G-168)



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K4.01	
	Importance Rating	2.6	

K/A Statement

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

Proposed Question: RO 34 Rev: 0

Given:

- The crew is performing a fill and drain of the Quench Tank to reduce Quench Tank temperature in accordance with OP-007-001, Boron Management

The crew will fill the Quench Tank with ____(1)____ and then drain the Quench Tank to the ____(2)____ .

	(1)	(2)
A.	Primary Makeup (PMU)	Equipment Drain Tank
B.	Primary Makeup (PMU)	Reactor Drain Tank
C.	Condensate Makeup (CMU)	Equipment Drain Tank
D.	Condensate Makeup (CMU)	Reactor Drain Tank

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: In accordance with OP-007-001 Section 8.14, PMU is un-isolated from containment to fill the Quench Tank and then the Quench Tank is drained to the Reactor Drain Tank. The EDT is plausible because it is the tank that the RDT is drained to.
B. **CORRECT:** In accordance with OP-007-001 Section 8.14, PMU is un-isolated from containment to fill the Quench Tank and then the Quench Tank is drained to the Reactor Drain Tank.
C. Incorrect: In accordance with OP-007-001 Section 8.14, PMU is un-isolated from containment to fill the Quench Tank and then the Quench Tank is drained to the Reactor Drain Tank. CMU is plausible because it also has piping to containment.
D. Incorrect: In accordance with OP-007-001 Section 8.14, PMU is un-isolated from containment to fill the Quench Tank and then the Quench Tank is drained to the Reactor Drain Tank.

Technical Reference(s): OP-007-001 section 8.14, Fill and Drain to reduce Quench Tank temperature, Revision 25
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-RCS00 obj. 2 (As available)

Question Source: Bank # X Question 34
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2011 RO Exam

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

10 CFR Part 55 Content: 55.41 3,7
55.43

Comments:

8.14 Fill and Drain to Reduce Quench Tank Temperature

8.14.1 Obtain SM/CRS permission to operate PMU Containment Isolation, PMU-151.

8.14.2 Unlock and slowly throttle Open PMU Containment Isolation, PMU-151.

8.14.3 Slowly fill Quench Tank to a level of 95 %.

NOTE
Maximum normal Quench Tank Pressure is 10 psig.

8.14.4 Close PMU Containment Isolation, PMU-151.

NOTE
Maintain a positive pressure on Quench Tank during draining process.

8.14.5 Open Quench Tank Drain to Reactor Drain Tank, RC-325.

NOTE
Normal RDT Pressure band is 1-14 PSIG. Maximum RDT level is 95 %.

8.14.6 Monitor RDT pressure and level.

8.14.7 Drain Quench Tank to a level of approximately 60 %.

8.14.8 Close Quench Tank Drain to Reactor Drain Tank, RC-325.

8.14.9 Slowly throttle Open PMU Containment Isolation, PMU-151.

8.14.10 Slowly fill Quench Tank to low end of normal operating band of 75 % - 82 %.

8.14.11 Close and Lock PMU Containment Isolation, PMU-151.

8.14.11.1 Document on Attachment 11.24, Boron Management System Isolation Valves Restoration.

Examination Outline Cross-Reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 K3.01	
	Importance Rating	3.4	

K/A Statement

K3.02 Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Proposed Question: RO 35 Rev: 0

Given:

- At time 0205, the crew manually initiated a SIAS and CIAS due to a Steam Generator Tube Rupture event.
- At time 0210, CCW Pump B tripped
- The crew has diagnosed to OP-902-007, Steam Generator Tube Rupture Recovery Procedure
- Station loads are powered from the Startup Transformers

To prevent damage to Emergency Diesel Generator B, the crew must align CCW flow before time ____(1)____ . If CCW flow cannot be restored to EDG B, the diesel will be secured using the _____(2)_____ .

	(1)	(2)
A.	0220	emergency Shutdown push buttons on CP-1
B.	0220	overspeed trip plunger
C.	0235	overspeed trip plunger
D.	0235	emergency Shutdown push buttons on CP-1

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: OI-038-000 step 5.2.3 states that EDGs can run without CCW for 10 minutes at 3.23 MW and for 25 minutes unloaded without causing damage to the diesels. In this situation, EDG B started and is running unloaded due to the SIAS, therefore the crew has 25 minutes to restore CCW flow. OP-902-007 requires the crew to pull the affected EDG overspeed trip plunger if CCW flow can not be restored..
B. Incorrect: OI-038-000 step 5.2.3 states that EDGs can run without CCW for 10 minutes at 3.23 MW and for 25 minutes unloaded without causing damage to the diesels. In this situation, EDG B started and is running unloaded due to the SIAS, therefore the crew has 25 minutes to restore CCW flow.
C. **CORRECT**: OI-038-000 step 5.2.3 states that EDGs can run without CCW for 10 minutes at 3.23 MW and for 25 minutes unloaded without causing damage to the diesels. In this situation, EDG B started and is running unloaded due to the SIAS, therefore the crew has 25 minutes to restore CCW flow upon the loss of CCW. OP-902-007 requires the crew to pull the affected EDG overspeed trip plunger if CCW flow can not be restored.
D. Incorrect: Part 1 is correct. OP-902-007 requires the crew to pull the affected EDG overspeed trip plunger if CCW flow can not be restored.

Technical Reference(s): OI-038-000 step 5.2.3 revision 7
(Attach if not previously provided) OP-902-007 Revision 15
(including version/revision number) ____

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE07 obj. 8 (As available)
WLP-OPS-PPE01 obj. 4

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

5.2.3 Verify Maintenance of Vital Auxiliaries

- The substeps listed for stopping steam emission to the main turbine are listed in a preferred order. If one of the Contingency Actions results in tripping the main turbine, it is not necessary or desired to perform the remaining substep.
- If the Exciter Field Breaker does not open and the Main Generator is tripped, the Exciter Field Breaker may be opened using the control switch on CP-1.
- The operator is required to check the status of all electrical buses listed in the step, since this will provide information to be used during the event diagnoses. CP-1 indications or the PMC may be used.
- The vital AC Instrument Channel that is required to be energized must be of the same train as the energized DC bus. CP-7 indications should be used to verify which vital AC Instrument Channels are energized.
- If a 3 Bus is de-energized an EDG should be verified to be operating. If the EDG is operating and the EDG output breaker has failed to close automatically the NPO should verify that EDG voltage is correct and that the 3-2 breaker has opened. If the EDG output breaker has failed to close automatically then two NAOs should be sent to close the output breaker locally, one NAO to perform the manual closure action and the other to act as a Safety Observer. Do not attempt to close the EDG output breaker if a fault exists on the safety bus.
  - A CCW pump should be verified to be operating for each operating EDG.
  - A calculation was performed that states the EDG’s can run without CCW for 10 minutes at 3.23 MW and for 25 minutes unloaded without causing damage to the EDG’s. This calculation is based on the EDG’s starting from a standby condition. **[EC-36621, ECM12-001]**
- The NPO should observe the sequencer is operating properly for a LOOP or SIAS. The NPO should not attempt to operate equipment that is not normally sequenced on until the sequencer has completely timeout.
- It is recommended to verify proper operation of EDGs within 20 minutes of starting.
- If power to the MSR control panel is not available, the Control Room should direct the NAO to isolate the MSR. The NAO should isolate the air and open the petcock to all 5 inch and 10 inch valves. The NAO should then verify all MSR TCVs close.
- The CRS should ask the NPO the status of Maintenance of Vital Auxiliaries on the substep level. The NPO will report the status of Maintenance of Vital Auxiliaries to the CRS on the substep level.
- The CRS should ensure the RO has reported the status of the Maintenance of Vital Auxiliaries acceptance criteria. The CRS may prompt the operator to obtain the necessary information.

INSTRUCTIONS	CONTINGENCY ACTIONS
Verify Proper CCW Operation	
* 10. Check a CCW pump is operating for each energized 4.16 KV safety bus. (continue)	10.1 **IF** the AB electrical bus is aligned to the side with the faulted CCW pump, **THEN** start CCW pump AB as follows: a. Place the CCW ASSIGNMENT switch to the desired position to replace the faulted pump. b. Verify open the CCW SUCT & DISCH HEADER TIE VALVES for the faulted CCW pump: **Train A** • CC 126A/CC 114A • CC 127A/CC 115A **Train B** • CC 126B/CC 114B • CC 127B/CC 115B c. Start CCW PUMP AB. d. **IF** CCW flow is **NOT** restored, **THEN** pull the affected emergency diesel generator overspeed trip.

INSTRUCTIONS	CONTINGENCY ACTIONS

* 10. (continued)

10.2 **IF** the AB electrical bus is **NOT** aligned to the side with the faulted CCW pump **AND** the sequencer has timed out, **THEN** <u>start</u> CCW pump AB as follows:

a. <u>Place</u> the CCW ASSIGNMENT switch to the desired position to replace the faulted pump.

b. <u>Verify</u> open the CCW SUCT & DISCH HEADER TIE VALVES for the faulted CCW pump:

Train A

- CC 126A/CC 114A
- CC 127A/CC 115A

Train B

- CC 126B/CC 114B
- CC 127B/CC 115B

c. <u>Start</u> CCW PUMP AB.

d. **IF** CCW flow is **NOT** restored, **THEN** <u>pull</u> the affected emergency diesel generator overspeed trip.

10.3 **IF** CCW Pump AB is the faulted pump **THEN**:

a. <u>Place</u> the CCW ASSIGNMENT switch to the neutral position **AND** <u>start</u> CCW Pump A(B).

b. **IF** CCW flow is **NOT** restored, **THEN** <u>pull</u> the affected emergency diesel generator overspeed trip.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 A1.02	
	Importance Rating	2.9	

K/A Statement

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

Proposed Question: RO 36 Rev: 0

Given:

- The crew has entered OP-901-130, Reactor Coolant Pump Malfunction due to a high bearing temperature on RCP 2B
- The CRS has directed the BOP operator to lower RCP 2B bearing temperature using DCT fans

Per OP-901-130, Reactor Coolant Pump Malfunction, CCW temperature should be maintained above ____(1)_____ °F to prevent _______(2)______ .

	(1)	(2)
A.	75	degradation of the reactor coolant pump seals
B.	75	essential chiller trips on low refrigerant pressure
C.	60	degradation of the reactor coolant pump seals
D.	60	essential chiller trips on low refrigerant pressure

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: OP-901-130 section E3 caution states that CCW temperatures < 75 °F could lead to essential chiller trips on low refrigerant pressure. The degredation of the reactor coolant pump seals is plausible because it is the second part of the same caution, but is due to a CCW temperature change of > 10°F per hour.
B. **CORRECT:** OP-901-130 section E3 caution states that CCW temperatures < 75 °F could lead to essential chiller trips on low refrigerant pressure.
C. Incorrect: The 60°F limit is plausible because it is the chillwater inlet temperature at which the essential chiller should be loaded manually, IAW OP-002-004, Chilled Water System. The degredation of the reactor coolant pump seals is plausible because it is the second part of the same caution, but is due to a CCW temperature change of > 10°F per hour.
D. Incorrect: The 60°F limit is plausible because it is the chillwater inlet temperature at which the essential chiller should be loaded manually, IAW OP-002-004, Chilled Water System.

Technical Reference(s): OP-901-130 Section E3 Revision 7
(Attach if not previously provided) OP-002-003 limitation 3.2.3 Revision 311
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CC00 obj. 8 (As available)
WLP-OPS-PPO30 obj. 3

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

E_3 BEARING TEMPERATURE HIGH

	PLACEKEEPER		
	START	DONE	N/A
1. IF ANY of the following occurs, THEN notify Duty Plant Manager and System Engineer:	☐	☐	☐
• ANY bearing temperature exceeds 205°F		☐	
• Reactor Coolant Pump Lube Oil Cooler differential temperature reaches 30°F		☐	
• A change in ANY bearing temperature of >10°F over 1 hour		☐	

CAUTION

(1) CCW TEMPERATURES OF <75° F COULD LEAD TO ESSENTIAL CHILLER TRIPS ON EVAPORATOR LOW REFRIGERANT PRESSURE.

(2) CCW TEMPERATURE SHOULD BE CHANGED AT A RATE OF ≤10°F IN ONE HOUR TO PREVENT DEGREDATION OF THE REACTOR COOLANT PUMP SEALS.

	START	DONE	N/A
2. Attempt to lower bearing temperature(s) by ANY of the following:	☐	☐	☐
• Start Dry Cooling Tower Fans.		☐	☐
• Start Auxiliary Component Cooling Water Pump(s) AND associated Wet Cooling Tower Fans.		☐	☐
• Start Auxiliary Component Cooling Water Pump(s) AND lower ACC-126A(B) setpoint.		☐	☐
• Start an oil lift pump on affected RCP for 5-10 minutes.		☐	☐

3.2.2.1 With the motor cold, do not attempt more than two consecutive starts, allowing the motor to come to rest between start attempts. It is considered a start when motor comes up to rated speed. The motor should be at rest for 30 minutes before next start attempt.

3.2.2.2 With the motor hot, do not attempt more than one start. Additional starts can be made if the following conditions are both met:

- The motor runs at least 15 minutes between starts.
- The motor has been at rest for at least 30 minutes between starts.

3.2.3 CCW Temperatures of <75° F could lead to Essential Chiller trips on evaporator low refrigerant pressure.

3.2.4 Notify Chemistry when placing any branch line into service that has been idle for more than 5 days.

3.2.5 Manipulating DCT Fans out of sequence has the potential to disrupt the DCT sequencing control logic. Disrupting the DCT sequencing control logic does not impact the safety related function of the DCT Fans to start in FAST speed. When possible, the DCT Fans should be started in sequential order when being started in manual (ascending when being started and descending when being secured). This does not prevent a fan from being started out of sequence (i.e. DCT Fan 14A could be started for retest when only DCT Fans 1A, 2A, and 3A are running is SLOW speed). If the DCT sequencing control logic has been disrupted, then it is recommended to take all DCT Fans to OFF and then take all DCT Fans back to AUTO. The DCT Fan sequencing control logic should then reset when the DCT Fans sequence back on. **[CR-WF3-2012-01094]**

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

K/A Statement

A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures

Proposed Question: RO 37 Rev: 0

Given:

- Plant is at 100% power
- A loss of all Pressurizer heaters has occurred
- The crew has entered OP-901-120, Pressurizer Pressure Control Malfunction
- Pressurizer Level Setpoint source input has been shifted at CP-31 and a plant shutdown has commenced in accordance with OP-010-005, Plant Shutdown

During the shutdown, the crew is required to ____(1)____ . Pressurizer level setpoint will be controlled _____(2)_____ .

	(1)	(2)
A.	maintain pressurizer level constant	manually by the operator
B.	raise pressurizer level	automatically from the Reactor Regulating System
C.	maintain pressurizer level constant	automatically from the Reactor Regulating System
D.	raise pressurizer level	manually by the operator

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: OP-901-120 requires the crew to maintain pressurizer level constant during the cooldown to conserve Pressurizer inventory and enthalpy. The pressurizer level controller is maintained in auto and the setpoint is placed in manual. Setpoint adjustment is performed manually.
B. Incorrect: Raising pressurizer level is plausible since the purpose of the actions in OP-901-120 is to conserve inventory. The pressurizer level controller is maintained in auto and the setpoint is placed in manual. The setpoint is normally adjusted by the Reactor Regulating System.
C. Incorrect: The pressurizer level controller is maintained in auto and the setpoint is placed in manual. The pressurizer level controller is maintained in auto and the setpoint is placed in manual. The setpoint is normally adjusted by the Reactor Regulating System..
D. Incorrect: Raising pressurizer level is plausible since the purpose of the actions in OP-901-120 is to conserve inventory. The pressurizer level controller is maintained in auto and the setpoint is placed in manual.

Technical Reference(s): OP-901-120 revision 302
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 obj. 3 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 5,10
55.43 ______

Comments:

E$_4$ LOSS OF ALL PRESSURIZER HEATERS

PLACEKEEPER: START | DONE | N/A

1. Place Pressurizer Spray Controller (RC-IHIC-0100) to MAN and adjust output to 0%. ☐ ☐

2. Verify Pressurizer Spray Valves (RC 301A and RC 301B) closed. ☐

NOTE

Maintaining Pressurizer Level constant will conserve Pressurizer inventory and enthalpy.

3. Shift PRESSURIZER LEVEL SETPOINT SOURCE input from RRS to RTGB as follows: ☐ ☐

   a. Place Pressurizer Level Controller (RC-ILIC-0110) in Manual. ☐

   b. At CP-31 Card Frame 1, Slot #36, place switches 1 and 2 to DEFEAT. ☐

   c. At Pressurizer Level Controller (RC-ILIC-0110) verify setpoint matches Pressurizer level. ☐

   d. Place Pressurizer Level Controller (RC-ILIC-0110) to AUTO. ☐

CAUTION

CVC-209, CVC-216A, & CVC-218A ARE CONTAINMENT ISOLATIONS. CVC-209 FAILS OPEN, AND CVC-216A & CVC-218A FAIL CLOSED. REFER TO TECHNICAL SPECIFICATION 3.6.3 AND TRM TABLE 3.6-2.

4. If Pressurizer Heaters were lost due to fire, then de-energize the following valves: ☐ ☐ ☐

   - Pressurizer Aux Spray Valve A (CVC-216A) by opening breaker (CVCEBKR90A 31) ☐
   - Charging Line To RCS Loop 1A Isolation (CVC-218A) by opening breaker (CVC-EBKR-90A-27) ☐
   - Charging Header Isolation (CVC-209) by opening breaker (CVC-EBKR-AB-38). ☐

E$_4$ LOSS OF ALL PRESSURIZER HEATERS (CONT'D)

Step		PLACEKEEPER START	DONE	N/A
5.	Refer to the following Technical Specifications and Technical Requirements: • TS 3.2.8 • TS 3.4.3.1 • TRM 3.4.3.1		Continuous	
6.	Perform a plant shutdown to Mode 3 in accordance with OP-010-005, Plant Shutdown, concurrently with this procedure.	☐	☐	
7.	<u>When</u> Plant Shutdown to Mode 3 is complete, <u>then</u> secure Reactor Coolant Pumps 1A, 1B, 2A.	☐	☐	
8.	Perform a Plant cooldown to Mode 5 in accordance with OP-010-005, Plant Shutdown.	☐	☐	

END

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

K/A Statement

K5.01 Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB

Proposed Question: RO 38 Rev: 0

Which of the following is correct in relation to the DNBR-Low trip?

A. The Reactor Protection Trip setpoint is 1.24 and is automatically bypassed below the 10 E-4 bistable setpoint.

B. The Reactor Protection Trip setpoint is 1.24 and may be bypassed by manually enabling the operating bypass below the 10 E-4 bistable setpoint.

C. The Reactor Protection Trip setpoint is 1.26 and is automatically bypassed below the 10 E-4 bistable setpoint.

D. The Reactor Protection Trip setpoint is 1.26 and may be bypassed by manually enabling the operating bypass below the 10 E-4 bistable setpoint.

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: The safety limit for DNBR is 1.24 but the RPS trip setpoint is 1.26. The DNBR-low trip can be bypassed by enabling the operating bypass below the 10 E-4 bistable setpoint. The operating bypass will automatically be removed above the 10 E-4 bistable setpoint.
B. Incorrect: The safety limit for DNBR is 1.24 but the RPS trip setpoint is 1.26. The DNBR-low trip can be bypassed by enabling the operating bypass below the 10 E-4 bistable setpoint. The operating bypass will automatically be removed above the 10 E-4 bistable setpoint.
C. Incorrect: The safety limit for DNBR is 1.24 but the RPS trip setpoint is 1.26. The DNBR-low trip can be bypassed by enabling the operating bypass below the 10 E-4 bistable setpoint. The operating bypass will automatically be removed above the 10 E-4 bistable setpoint.
D. **CORRECT:** The safety limit for DNBR is 1.24 but the RPS trip setpoint is 1.26. The DNBR-low trip can be bypassed by enabling the operating bypass below the 10 E-4 bistable setpoint. The operating bypass will automatically be removed above the 10 E-4 bistable setpoint.

Technical Reference(s): TS Table 2.2.1 and note 5 of the same table
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPS00 obj. 1 and 4 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 2,6
55.43 ______

Comments:

TABLE 2.2-1
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
Four Reactor Coolant Pumps Operating	$\leq$ 108% of RATED THERMAL POWER	$\leq$ 108.76% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.257% of RATED THERMAL POWER (6)	$\leq$ 0.280% of RATED THERMAL POWER (6)
4. Pressurizer Pressure - High	$\leq$ 2350 psia	$\leq$ 2359 psia
5. Pressurizer Pressure - Low	$\geq$ 1684 psia (2)	$\geq$ 1649.7 psia (2)
6. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.4 psia
7. Steam Generator Pressure - Low	$\geq$ 666 psia (3)	$\geq$ 652.4 psia (3)
8. Steam Generator Level - Low	$\geq$ 27.4% (4)	$\geq$ 26.48% (4)
9. Local Power Density - High	$\leq$ 21.0 kW/ft (5)	$\leq$ 21.0 kW/ft (5)
10. DNBR - Low	$\geq$ 1.26 (5)	$\geq$ 1.26 (5)
11. DELETED		
12. Reactor Protection System Logic	Not ApplicableNot Applicable	Not Applicable
13. Reactor Trip Breakers	Not ApplicableNot Applicable	Not Applicable
14. Core Protection Calculators	Not ApplicableNot Applicable	Not Applicable
15. CEA Calculators	Not ApplicableNot Applicable	Not Applicable
16. Reactor Coolant Flow - Low	$\geq$ 19.00 psid (7)	$\geq$ 18.47 psid (7)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) The operating bypass may be enabled above the 10^{-4}% bistable setpoint and shall be capable of automatic removal whenever the operating bypass is enabled and logarithmic power is below the 10^{-4}% bistable setpoint. Trip may be manually bypassed during physics testing pursuant to Special Test Exception 3.10.3.

(2) Value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. The operating bypass may be enabled below the 10^{-4}% bistable setpoint and shall be capable of automatic removal whenever the operating bypass is enabled and logarithmic power is above the 10^{-4}% bistable setpoint. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; the 10^{-4}% bistable setpoint may be changed to less than or equal 5% RATED THERMAL POWER to perform the automatic removal function.

(6) As measured by the Logarithmic Power Channels

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

Examination Outline Cross-Reference:		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 A4.03	
	Importance Rating	4.5	

K/A Statement

A4.03 Ability to manually operate and/or monitor in the control room: ESFAS initiation

Proposed Question: RO 39 Rev: 0

Given:

- Pressurizer pressure is 1650 PSIA and slowly lowering
- Containment pressure is 16.7 and slowly rising
- Steam Generator 1 pressure is 700 PSIA and lowering
- Steam Generator 2 pressure is 680 PSIA and lowering

Which of the following ESFAS signal(s) will have been generated?

A. SIAS only

B. SIAS and CIAS only

C. SIAS, CIAS, and MSIS only

D. SIAS, CIAS, MSIS, and CSAS

Proposed Answer: B

Explanation: (Optional)

A. Incorrect. SIAS and CIAS will be generated together at PZR pressure of 1684 PSIA.
B. **CORRECT:** PZR Pressure at 1684 PSIA will generate an SIAS and CIAS.
C. Incorrect. MSIS will not be generated until one SG pressure drops below 666 PSIA or Containment pressure rises to 17.1 PSIA.
D. Incorrect. MSIS will not be generated until one SG pressure drops below 666 PSIA or Containment pressure rises to 17.1 PSIA. CSAS will not be generated until Containment pressure rises to 17.7 PSIA.

Technical Reference(s): TS 2.2.1
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPS00 obj. 3 (As available)

Question Source: Bank # X Question #39
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2012 NRC Exam

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

10 CFR Part 55 Content: 55.41 4,7
55.43

Comments:

TABLE 2.2-1
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
Four Reactor Coolant Pumps Operating	$\leq$ 108% of RATED THERMAL POWER	$\leq$ 108.76% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.257% of RATED THERMAL POWER (6)	$\leq$ 0.280% of RATED THERMAL POWER (6)
4. Pressurizer Pressure - High	$\leq$ 2350 psia	$\leq$ 2359 psia
5. Pressurizer Pressure - Low	$\geq$ 1684 psia (2)	$\geq$ 1649.7 psia (2)
6. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.4 psia
7. Steam Generator Pressure - Low	$\geq$ 666 psia (3)	$\geq$ 652.4 psia (3)
8. Steam Generator Level - Low	$\geq$ 27.4% (4)	$\geq$ 26.48% (4)
9. Local Power Density - High	$\leq$ 21.0 kW/ft (5)	$\leq$ 21.0 kW/ft (5)
10. DNBR - Low	$\geq$ 1.26 (5)	$\geq$ 1.26 (5)
11. DELETED		
12. Reactor Protection System Logic	Not ApplicableNot Applicable	Not Applicable
13. Reactor Trip Breakers	Not ApplicableNot Applicable	Not Applicable
14. Core Protection Calculators	Not ApplicableNot Applicable	Not Applicable
15. CEA Calculators	Not ApplicableNot Applicable	Not Applicable
16. Reactor Coolant Flow - Low	$\geq$ 19.00 psid (7)	$\geq$ 18.47 psid (7)

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	12
b. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
d. Automatic Actuation - Logic	4	2	3	1, 2, 3	12
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	12
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	13*, 14*
c. Automatic Actuation Logic	4	2	3	1, 2, 3	12
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	12
b. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
d. Automatic Actuation Logic	4	2	3	1, 2, 3	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION					
a. Manual (Trip Buttons)	2 sets of 2 per steam generator	1 set of 2 per steam generator	2 sets of 2 per operating steam generator	1, 2, 3	16
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	13*, 14*
c. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
d. Automatic Actuation Logic	4	2	3	1, 2, 3	12
5. SAFETY INJECTION SYSTEM SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Refueling Water Storage Pool - Low	4	2	3	1, 2, 3, 4	19a*, 19b, 20
c. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	12
6. LOSS OF POWER (LOV)					
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	3/bus	3/bus	3/bus	1, 2, 3	17, 18
b. 480 V Emergency Bus Undervoltage (Loss of Voltage)	3/bus	3/bus	3/bus	1, 2, 3	17, 18
c. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	3/bus	3/bus	3/bus	1, 2, 3	17, 18

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 K2.01	
	Importance Rating	3.0	

K/A Statement

K2.01 Knowledge of power supplies to the following: Containment cooling fans

Proposed Question: RO 40 Rev: 0

Containment Fan Cooler C is powered from which bus?

A. SWGR 31A

B. SWGR 31B

C. MCC 317A

D. MCC 317B

Proposed Answer: C

Explanation: (Optional)

A. Incorrect. Credible to use the 31 bus because it is 480 volt safety related power supply and is the power supply to the Control Element Drive Mechanism (CEDM) Fans.
B. Incorrect. Credible to use the 31 bus because it is 480 volt safety related power supply and is the power supply to the Control Element Drive Mechanism (CEDM) Fans. Credible to choose B Train supply because the applicant may not understand that Containment Fan Cooler C is an A train component
C. **CORRECT**: The power supply to Containment Fan Cooler C is MCC 317A.
D. Incorrect. Credible to choose B Train supply because the applicant may not understand that Containment Fan Cooler C is an A train component

Technical Reference(s): OP-008-003 revision 302
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CC00 Obj. 4 (As available)

Question Source: Bank # 08410
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2007 NRC RO Written Exam

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

10 CFR Part 55 Content: 55.41 4
55.43

Comments:

11.1 CONTAINMENT COOLING SYSTEM STANDBY BREAKER LINEUP

COMPONENT NUMBER	COMPONENT DESCRIPTION	LOCATION	REQUIRED POSITION	PERFORMED BY (INITIAL/DATE)	IV BY (INITIAL/DATE)
CCS-EBKR-317A-2M	CONTAINMENT COOLING HVAC AH-1 (3A-SA) BKR	RAB +21 COL 9A&H MCC 3A317	ON		
CCS-EBKR-317A-3M	CONTAINMENT COOLING HVAC AH-1 (3C-SA) BKR	RAB +21 COL 9A&H MCC 3A317	ON		
CCS-EBKR-317B-2M	CONTAINMENT COOLING HVAC AH-1 (3D-SB) BKR	RAB +21 COL 10AZ&L MCC 3B317	ON		
CCS-EBKR-317B-3M	CONTAINMENT COOLING HVAC AH-1 (3B-SB) BKR	RAB +21 COL 10AZ&L MCC 3B317	ON		
CCS-EBKR-60A-13	CONTAINMENT COOLING HVAC AH-1 (3A-SA) SPACE HTR BKR	RAB +21 COL 11A&H MCC 3A312	ON		
CCS-EBKR-60A-15	CONTAINMENT COOLING HVAC AH-1 (3C-SA) SPACE HTR BKR	RAB +21 COL 11A&H MCC 3A312	ON		
CCS-EBKR-61B-13	CONTAINMENT COOLING HVAC AH-1 (3B-SB) SPACE HTR BKR	RAB +21 COL 9A&K MCC 3B312	ON		
CCS-EBKR-61B-15	CONTAINMENT COOLING HVAC AH-1 (3D-SB) SPACE HTR BKR	RAB +21 COL 9A&K MCC 3B312	ON		
CCS-EBKR-60A-17	CFC SYSTEM A VLVS BKR	RAB +21 COL 11A&H MCC 3A312	ON		
CCS-EBKR-61B-17	CFC SYSTEM B VLVS BKR	RAB +21 COL 9A&K MCC 3B312	ON		
CCS-EBKR-45AB-3	CFC CONDENSATE POT FLOW DETECTORS BKR	RAB +35 COL 11A&H PDP 345AB	ON		

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A3.01	
	Importance Rating	4.1	

K/A Statement

A3.01 Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

Proposed Question: RO 41 Rev: 0

Given:

- Plant is at 100% power
- Containment Fan Coolers A, B, & D are in operation
- CEDM Fans A and D are in operation
- Reactor is manually tripped and SIAS is manually actuated

Which of the following describes the response during the event?

A. Containment Fan Coolers A, B, C, & D operate in slow speed. CEDM Fans A and D trip.

B. Only Containment Fan Coolers A, B, & D operate in slow speed. CEDM Fans A and D trip.

C. Containment Fan Coolers A, B, C, & D operate in slow speed. CEDM Fans A and D continue to operate.

D. Only Containment Fan Coolers A, B, & D operate in slow speed. CEDM Fans A and D continue to operate.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: On a SIAS, all four Containment Fan Coolers start in slow speed and any running CEDM fans will trip.
B. Incorrect. On a SIAS, all four Containment Fan Coolers start in slow speed and any running CEDM fans will trip.
C. Incorrect. On a SIAS, all four Containment Fan Coolers start in slow speed and any running CEDM fans will trip.
D. Incorrect. On a SIAS, all four Containment Fan Coolers start in slow speed and any running CEDM fans will trip.

Technical Reference(s): OP-008-004, Control Element Drive Mechanism Cooling System, Rev. 7

(Attach if not previously provided) OP-008-003, Containment Cooling System, Rev. 301

(including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CCS00 obj. 1 (As available)

Question Source: Bank # X Question #40
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2012 NRC Exam

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

10 CFR Part 55 Content: 55.41 4
55.43 ______

Comments:

9.0 AUTOMATIC FUNCTIONS

9.1	Containment Fan Coolers A, B, C & D start in slow if not running, shift to slow if running in fast	SIAS
9.2	Containment Fan Cooler Header A CC Return TCV, CC-835A, full open	SIAS
9.3	Containment Fan Cooler Header B CC Return TCV, CC-835B, full open	SIAS
9.4	Containment Cooling HVAC Safety Discharge Damper A, CCS-102A, full open	SIAS
9.5	Containment Cooling HVAC Safety Discharge Damper B, CCS-102B, full open	SIAS

9.0 AUTOMATIC FUNCTIONS

9.1 CEDM Cooling Fans, CDC-0002A, CDC-0002B, CDC-0002C, or CDC-0002D Stop .... CEDM area ambient temperature lowers to < 80°F.

9.2 CEDM Cooling Fans, CDC-0002A, CDC-0002B, CDC-0002C, and CDC-0002D Stop.................................... SIAS actuation

9.3 CEDM Cooling Fans, CDC-0002A CDC-0002B, CDC-002C, and CDC-0002D Stop .... Associated Inlet Damper CDC-101A(B,C,D) Closes

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 A1.03	
	Importance Rating	3.5	

K/A Statement

A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including:
Containment sump level

Proposed Question: RO 42 Rev: 0

Given:

- A large break LOCA occurred
- Containment Spray Line A pipe failure occurred in the -35 Wing Area

Which of the following conditions describes the operational concern for the Emergency Core Cooling systems following RAS initiation?

The ___(1)___ will not have adequate water inventory for operation of the ___(2)___ pumps.

	(1)	(2)
A.	RWSP	LPSI
B.	RWSP	HPSI
C.	SI sump	LPSI
D.	SI sump	HPSI

Proposed Answer: D

Explanation: (Optional)

A. INCORRECT: RWSP is NOT available once RAS is initiated. HPSI pumps remain running after an RAS.
B. INCORRECT: RWSP is NOT available once RAS is initiated. The SIS strainers are designed for CS and HPSI flow only following an RAS.
C. INCORRECT: The RWSP inventory is going to the -35 and not the SIS which is a concern since suction swaps to the SIS on the RAS.
D. **CORRECT:** The RWSP inventory is going to the -35 and not the SIS which is a concern since suction swaps to the SIS on the RAS. HPSI pumps remain running after an RAS.

Technical Reference(s): OP-902-002 step 43
(Attach if not previously provided) TG-OP-902-002 step 43
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CS00 Obj. 1, 5 (As available)

Question Source: Bank # 6504-A
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 8
55.43

Comments:

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INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

SI 120A(B) and SI 121A(B), SI PUMPS RECIRC ISOL, should be closed within two minutes of receipt of RAS to prevent recirculating SI sump water to the RWSP.

RAS Initiation Criteria

* 43. **IF** the break is inside containment, **AND** RWSP level is less than 10%, **THEN**:

a. Verify the RAS is initiated.

b. Verify that **BOTH** LPSI pumps are stopped.

b.1 **IF** LPSI pump continues to run **THEN** continue with Instruction Step 43.c.

c. Verify that ESF PUMPS SUCTION SI SUMP valves are open:

- SI 602A
- SI 602B

d. Close the SI PUMPS RECIRC ISOL VALVES within two minutes of receipt of RAS:

- SI 120A
- SI 120B
- SI 121A
- SI 121B

(continue)

INSTRUCTIONS	CONTINGENCY ACTIONS
* 43. (continued)	
e. Close the ESF PUMPS SUCTION RWSP: • SI 106A • SI 106B	

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

-- **NOTE** --

If a LPSI Pump does not trip, perform step 43.f.1-2 actions within five minutes. Time starts at SI Pumps Recirc Isol Valves closed and ends at SI FCVs closed.

----

43. (continued)

f. **IF** any LPSI pump operating, **THEN** perform the following:

1) Throttle Open SI-135A(B) RC LOOP 2(1) SHDN COOLING WARM-UP approximately 10 seconds.

2) Reset Open then Close LPSI HEADER TO RC LOOP FLOW CONTROL:

- SI-138A(B)
- SI-139A(B)

3) Throttle Open SI-135A(B) to establish approximately 4000 gpm.

4) Stop operating LPSI Pump A(B).

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

43. (continued)

g. Place **ALL** charging pumps in "OFF."

h. Close CVC 209, CHARGING HEADER ISOLATION.

i. Verify CVC 507,RWSP TO CHARGING PUMPS SUCTION ISOLATION closed.

SI-602 Override

* 44. **IF** in the opinion of the Emergency Coordinator, closing SI 602A(B) Safety Injection System Sump Isolation valve to stop Emergency Core Cooling System leakage is in the best interest of protecting the public health and safety, **THEN** REFER TO Appendix 29, "SI-602 Override" and isolate SI 602A(B).

Step Number 43 RAS Initiation Criteria Note

Objective

The intent of this Note is to ensure the operators are made aware that SI-120A(B) and SI-121A(B) should be closed within two minutes of receipt of an RAS to prevent recirculating SI sump water to the RWSP before the step for aligning the Safety Injection system for a RAS.

Justification for Deviations

The EPG does not include this Note. Waterford has committed to have these valves closed within two minutes of the RAS signal.

References

None

Step Number 43 RAS Initiation Criteria

Objective

The intent of this step is to ensure that RAS is initiated when required and that the associated equipment and systems function as designed.

Instructions

For breaks located inside containment, if the refueling water storage pool level lowers to less than the RAS setpoint the operator should perform the following:

- The operator firsts verifies that RAS has automatically initiated. If it did not, then the operator should manually initiate RAS. Recirculation is actuated either automatically or manually in order to maintain a continuous flow of safety injection fluid to the RCS (required for inventory control) and a continuous flow of containment spray water (required for containment temperature and pressure control).

- The operator then verifies that the LPSI pumps stop, since they are not required following RAS. If the LPSI pumps do not stop the operator should secure them. The SI sump is not designed for full HPSI, LPSI and CS pump flows. A contingency step is added to direct Operations to continue through the steps if a LPSI pump does not stop.

- The operator then verifies that the ESF pump suction valves from the containment sump are open. This will provide a suction flowpath for the operating ESF pumps. If the valves do not open automatically, then the operator should open them.

- The operator then ensures that the SI pump minimum recirculation flow valves are closed to prevent contaminated containment sump water from returning to the RWSP. Waterford has committed to ensure these valves are closed within two minutes of the RAS signal.

- After verifying the above steps the operator should close the RWSP outlet valves one at a time. In addition, for events where high containment pressure is present, the check valves in the RWSP outlet line may be forced shut and the RWSP fluid will remain unavailable while the containment is pressurized.

Step Number 43 RAS Initiation Criteria (cont)

- Steps are added to direct the operator to take action if a LPSI pump did not stop. These actions are added to protect the containment sump from additional debris buildup that is not analyzed. The containment sump long term design operation is for two HPSI pumps and two Containment Spray Pumps all at full flow. Five minutes is allotted to complete the action.

- Throttle open SI-135 approximately 10 seconds, this is to establish a minimum flow through the LPSI pump prior to closing the FCVs to prevent dead heading the pump.

- Closing the FCVs stops the LPSI pump from pulling flow and additional debris through the containment sump screens.

- Open SI-135 to establish 4000 gpm flow through the LPSI pump as indicated on SI-IFIC-0306(-0307) or SI-IFI-1306-B1(-1307-A1).

- The maximum expected temperature is 378°F for train A and 326°F for train B. Both of these temperatures are below the maximum operating temperature for the LPSI pumps of 400°F and are therefore acceptable. The time to approach within 10°F of these maximum temperatures is 187 hours for train A and 156 hours for train B. If all power to the pumps goes into heating the water and it is assumed that no heat is lost to CCW, the time for the fluid to reach 400°F would be 4.4 hours for train A and 5.6 hours for train B. This is considered adequate time for operator response to secure the operating pumps or to take action to flush the hot water from the lines.

- The charging pumps are secured following RAS, since they are no longer required and are disabled to prevent charging pump damage when the adequate suction can no be longer provided.

- The operator is directed to isolate the charging penetration. Since the charging pumps are no longer in service the charging penetration is not being used and should be isolated.

- CVC-507 is verified closed in the event that charging pumps had been aligned to the RWSP as a suction source.

Step Number 43 RAS Initiation Criteria (cont)

Contingency Actions

A contingency step is added to direct Operations to continue through the steps if a LPSI pump does not stop.

Justification for Deviations

Waterford reverses EPG steps 35.d and 35.e. Step 35.d is to verify the RWSP outlet valves are closed and Step 35.e is to close the SI pumps recirc flowpath back to the RWSP. Waterford is committed to minimize exposure of plant personnel and to limit offsite dose. Closing the recirculation flowpath back to the RWSP will help to minimize this exposure, therefore these steps have been reversed for the operator to take the action to isolate the flowpath back to the RWSP sooner. Waterford includes a step to close the charging header isolation which will ensure the containment isolation safety function is met for the charging penetration with no charging pumps running.

References

1.) Commitment P 21391
2.) EC 26496 for actions of a LPSI Pump failing to stop on a RAS.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 K4.04	
	Importance Rating	2.9	

K/A Statement

K4.04 Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Utilization of steam pressure program control when steam dumping through atmospheric relief/dump valves, including T-ave. limits

Proposed Question: RO 43 Rev: 0

Given:

- A reactor trip occurred
- The crew is performing OP-902-000, Standard Post Trip Actions
- SG #1 and SG #2 pressures are 1060 psia
- SG #1 and SG #2 NR levels are 5% and steady
- Main Feedwater is available
- Tc indicate 555° F and stable

The crew is required to raise feedwater flow to maintain SG levels ____(1)____ NR and verify _____(2)_____ being restored back into the bands required by OP-902-000 using the SBCS.

	(1)	(2)
A.	10-76%	RCS temperature only is
B.	10-76%	RCS temperature and SG pressures are
C.	55% - 70%	RCS temperature and SG pressures are
D.	55% - 70%	RCS temperature only is

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: The crew is required to check SG levels are 10-76% NR Post trip. But are required to maintain level at 55-70% NR. RCS temperature (limit 550°F) and SG pressure (limit 1040 psia) are being exceeded and must be restored.
B. Incorrect: The crew is required to check SG levels are 10-76% NR following a Rx Trip. But are required to maintain level at 55-70% NR. RCS temperature (limit 550°F) and SG pressure (limit 1040 psia) are being exceeded and must be restored.
C. **CORRECT**: The crew is required to maintain SG level at 55-70% NR. RCS temperature (limit 550°F) and SG pressure (limit 1040 psia) are being exceeded and must be restored.
D. Incorrect: RCS temperature (limit 550°F) and SG pressure (limit 1040 psia) are being exceeded and must be restored.

Technical Reference(s): OP-902-000 revision 15
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 9 (As available)

Question Source: Bank # None
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS
Verify Core Heat Removal	
___5. Determine Core Heat Removal acceptance criteria are met:	
___a. Check at least one RCP is operating.	
___b. Check operating loop ΔT is less than 13ºF.	
___c. Check RCS subcooling is greater than or equal to 28ºF.	
Verify RCS Heat Removal	
___6. Determine RCS Heat Removal acceptance criteria are met:	
___a. Check that at least one steam generator has **BOTH** of the following: • Steam generator level is 10% to 76% NR. • Main Feedwater is available to restore level within 55% to 70% NR **[60% to 80% NR].**	a.1 Verify Main Feedwater is restoring level in at least one steam generator within 55% to 70% NR **[60% to 80% NR]**. a.2 Verify Emergency Feedwater is available to restore level in at least one steam generator within 55% to 70% NR **[60% to 80% NR]**.

(continue)

INSTRUCTIONS	CONTINGENCY ACTIONS
6. (continued)	
___ b. Check RCS T_C is 530ºF to 550ºF (continue)	b.1 **IF** RCS T_C is greater than 550ºF, **THEN** confirm that at least one steam generator is removing RCS heat: • Verify level is being restored to at least one steam generator • Verify Steam Bypass or ADVs are restoring RCS T_C 530ºF to 550ºF b.2 **IF** RCS T_C is less than 530ºF, **THEN** perform the following: • Verify feedwater flow is **NOT** excessive • Verify Steam Bypass or ADVs are restoring RCS T_C 530ºF to 550ºF • **IF** RCS T_C is less than 382ºF, **THEN** verify no more than two RCPs are operating • **IF** RCS T_C is being controlled by an ESD, **THEN** REFER TO Appendix 13, "Stabilize RCS Temperature" and stabilize RCS temperature using the least affected steam generator

INSTRUCTIONS	CONTINGENCY ACTIONS
6. (continued)	
___ c. Check steam generator pressure is 885 psia to 1040 psia.	c.1 **IF** steam generator pressure is less than 885 psia, **THEN** perform **ALL** of the following: 1) Verify steam bypass valves are closed. 2) Verify ADVs are closed. c.2 **IF** steam generator pressure is less than or equal to 666 psia, **THEN** verify MSIS is initiated. c.3 **IF** steam generator pressure is greater than 1040 psia, **THEN** verify that Steam Bypass or ADVs are restoring steam generator pressure to less than 1040 psia.
___ d. Check Feedwater Control in Reactor Trip Override: • MAIN FW REG valves are closed • STARTUP FW REG valves are 13% to 21% open • Operating main feedwater pumps are 3800 rpm to 4000 rpm	c.4 Manually operate the Feedwater Control system and restore level in at least one steam generator within 55% to 70% NR **[60%-80% NR]**.

(continue)

INSTRUCTIONS	CONTINGENCY ACTIONS
6. (continued)	
___ e. Reset moisture separator reheaters, and check the temperature control valves closed.	e.1 Isolate the moisture separator reheaters by locally closing the air operated temperature control valves.
Verify Containment Isolation	
___ 7. Determine Containment Isolation acceptance criteria are met:	
___ a. Check containment pressure is less than 16.4 psia.	a.1 **IF** containment pressure is greater than or equal to 17.1 psia, **THEN** verify the following: • CIAS is initiated • SIAS is initiated • MSIS is initiated
___ b. Check **NO** containment area radiation monitor alarms **OR** unexplained rise in activity.	
___ c. Check **NO** steam plant activity monitor alarms **OR** unexplained rise in activity.	

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 A3.03	
	Importance Rating	2.5	

K/A Statement

A3.03 Ability to monitor automatic operation of the MFW, including: Feedwater pump suction flow pressure

Proposed Question: RO 44 Rev: 0

Given:

- The plant is at 100% power
- ES-205, ES to #2 Heaters Isolation closed causing all three heater drain pumps to trip
- The crew has entered OP-901-221, Secondary System Transient

Which of the following describes the response during the event?

A. Condensate Polisher D/P will lower, both Main Feedwater Pumps will trip 10 seconds after reaching the suction pressure trip setpoint.

B. Condensate Polisher D/P will lower, one Main Feedwater Pump will trip 10 seconds after reaching the suction pressure trip setpoint.

C. Condensate Polisher D/P will rise, both Main Feedwater Pumps will trip 10 seconds after reaching the suction pressure trip setpoint.

D. Condensate Polisher D/P will rise, one Main Feedwater Pump will trip 10 seconds after reaching the suction pressure trip setpoint.

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: Condensate Polisher D/P will rise because more flow is being forced through the polisher on HDP trips. Only Main Feedwater Pump A will trip 10 seconds after reaching 250 psig for FWP suction pressure. Main Feedwater Pump B has a 30 second time delay to trip on low suction pressure. Suction pressure should restore after the MFP A trip.
B. Incorrect: Condensate Polisher D/P will rise because more flow is being forced through the polisher on HDP trips. The second part is correct.
C. Incorrect: The first part is correct. Only Main Feedwater Pump A will trip 10 seconds after reaching 250 psig for FWP suction pressure. Main Feedwater Pump B has a 30 second time delay to trip on low suction pressure. Suction pressure should restore after the MFP A trip.
D. **CORRECT**: Condensate Polisher D/P will rise because more flow is being forced through the polisher on HDP trips. Only Main Feedwater Pump A will trip 10 seconds after reaching 250 psig for FWP suction pressure. Main Feedwater Pump B has a 30 second time delay to trip on low suction pressure. Suction pressure should restore after the MFP A trip.

Technical Reference(s): OP-901-221 revision 0.
(Attach if not previously provided) OP-003-033 revision 313 step 9.1
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-FWP00 obj. 2 (As available)
WLP-OPS-PPO20 obj. 3

Question Source: Bank # None
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 4
55.43 ______

Comments:

C AUTOMATIC ACTIONS

1. Condensate Pumps trip on a Main Condenser Hotwell level of 1.5 feet (18”).

2. Hotwell Emergency M/U Valve, CMU-715, opens on a Main Condenser Hotwell level of 2.5 feet (30”) and closes at 2.75 feet (33”).

3. SGFP(s) trip on low suction pressure as follows:
   - SGFP A trips at 250 PSIG (reset at 280 PSIG) with a 10 sec time delay
   - SGFP B trips at 250 PSIG (reset at 280 PSIG) with a 30 sec time delay

9.0 AUTOMATIC FUNCTIONS

9.1 Main Feedwater Pump Turbine Trip

9.1.1	FWPT A RECIRC FAILURE	SGFP A(B) discharge pressure <900 PSIG and either of the following: • Flow demand vs. actual suction flow differential $\geq$562 GPM or • either recirc isolation valve not fully open
9.1.2	FWPT A LUBE OIL PRESS LO (LOF-IPS-3006A1, A2, A3)	10 PSIG
9.1.3	FWPT A OVERSPEED (FW-IST-2010A)	5720 RPM
9.1.4	FWPT A VACUUM LO (FW-IPS-3007A5)	14” Hg vac.
9.1.5	FWPT A SUCTION PRESS LO (CD-IPS-2203A, 2204A, 2205A) (2/3) (Time Delay 10 Seconds)	250 PSIG (R: 280 PSIG)
9.1.6	FWPT A FLOW LO (CD-IDPIS-2207A, 2208A and CD-IFIS-2202A) (2/3)	2700 GPM and SGFP A(B) discharge pressure >900 PSIG (R: 760 PSIG)
9.1.7	FWPT A CD PUMP LOST	A and B FWPT HP governor valves open and A and C Condensate Pumps tripped.
9.1.8	FWPT A THRUST BEARING WEAR (FW-IZD-3016A) (2/2)	Channel A Normal 25 mils Channel A Reverse -10 mils Channel B Normal 25 mils Channel B Reverse -10 mils

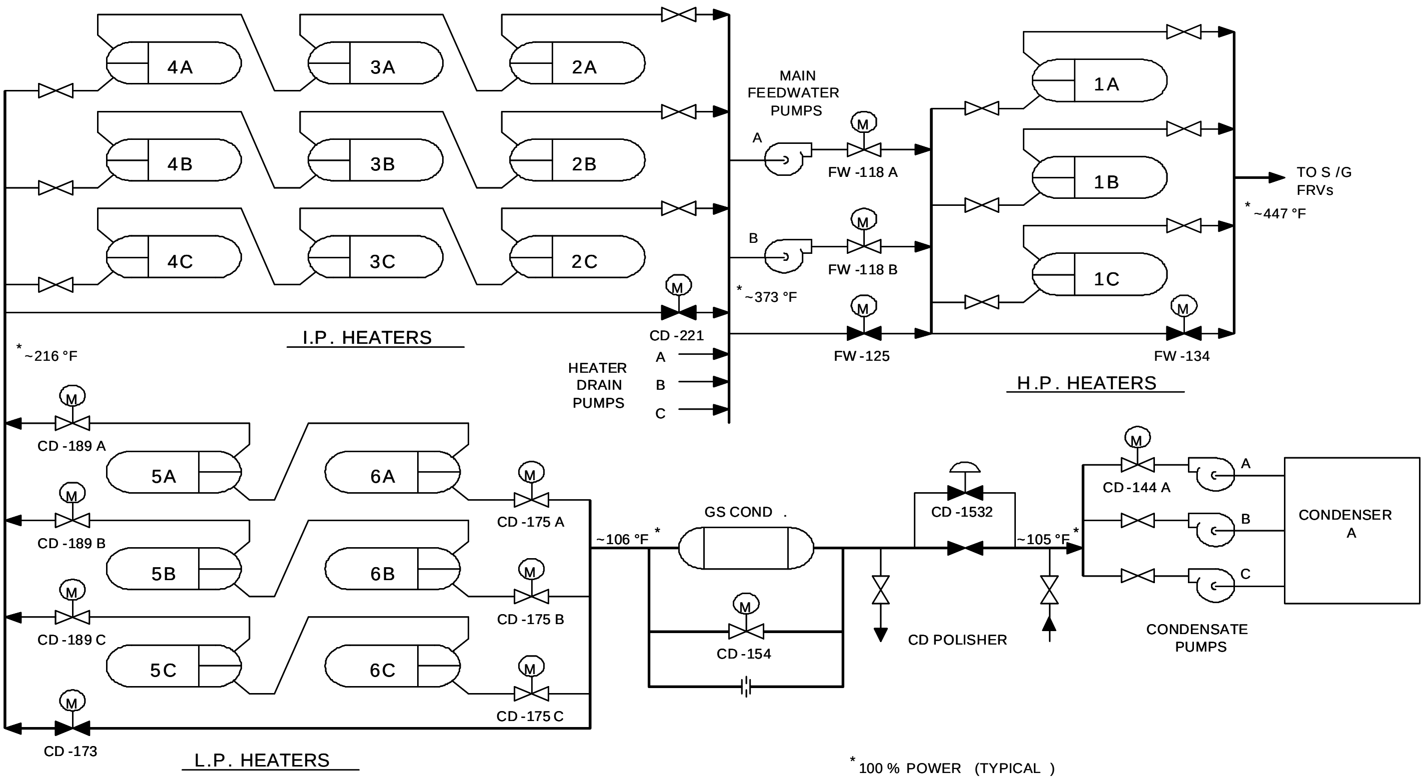
9.1.9	FWPT B RECIRC FAILURE	SGFP A(B) discharge pressure <900 PSIG and either of the following: • Flow demand vs. actual suction flow differential ≥562 GPM or • either recirc isolation valve not fully open
9.1.10	FWPT B LUBE OIL PRESS LO (LOF-IPS-3006B1, B2, B3)	10 PSIG
9.1.11	FWPT B OVERSPEED (FW-IST-2010B)	5720 RPM
9.1.12	FWPT B VACUUM LO (FW-IPS-3007B5)	14" Hg vac.
9.1.13	FWPT B SUCTION PRESS LO (CD-IPS-2203B, 2204B, 2205B) (2/3) (Time Delay 30 Seconds)	250 PSIG (R: 280 PSIG)
9.1.14	FWPT B FLOW LO (CD-IDPIS-2207B, 2208B and CD-IPIS-2202B) (2/3)	2700 GPM and SGFP A(B) discharge pressure >900 PSIG (R: 760 PSIG)
9.1.15	FWPT B CD PUMP LOST	A and B FWPT HP governor valves open and B and A, or C Condensate Pumps tripped.
9.1.16	FWPT B THRUST BEARING WEAR (FW-IZD-3016B) (2/2)	Channel A Normal 25 mils Channel A Reverse -10 mils Channel B Normal 25 mils Channel B Reverse -10 mils

9.2 Main Feedwater Pump Turbine Speed lowers to "0"

9.2.1	SSPU A FAILURE (TGB +40 MUX TB4005 ONLY)	Both speed sensors fail, or loss of power to cabinet (4 second time delay)
9.2.2	SSPU B FAILURE (TGB +40 MUX TB4005 ONLY)	Both speed sensors fail, or loss of power to cabinet (4 second time delay)

FIG . 01 CD /FW FLOWPATH THROUGH FEEDWATER HEATERS

(REF . G-176 , G-178)



SD-ES-01

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 2.1.28	
	Importance Rating	4.1	

K/A Statement

2.1.28 Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO 45 Rev: 0

Given:

- A plant power reduction is in progress
- Reactor power is 58% and slowly lowering
- Main Feedwater pumps A & B are providing Feedwater in AUTO mode
- The BOP reports Feedwater Master Controller #1 output has failed LOW
- The BOP reports NO response while attempting to operate the Master Controller #1 in MANUAL

Which of the following should be operated in MANUAL mode to restore and maintain Steam Generator level within band?

A. Main Feedwater Regulating valve #1 M/A station only

B. Startup Feedwater Regulating valve #1 M/A station only

C. Main Feedwater Pump Speed #1 M/A station only

D. Both the SUFRV #1 M/A station and Main Feedwater Pump Speed #1 M/A station

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** MFW regulating valve controller adjustment is the proper action at this power level. The function of the SUFRV is to maintain level at low power levels.
B. INCORRECT: At 58% power, the SUFRV capacity is inadequate to restore level. The Main FWRV must be opened.
C. INCORRECT: MFW Pump Speed will take the high select from FWC Master #1 or #2. Pump speed will be adequate.
D. INCORRECT: The SUFRV capacity is inadequate to restore SG level at this power Feedwater flow at this power level. MFW Pump Speed will take the high select from FWC Master #1 or #2. Pump speed will be adequate.

Technical Reference(s): OP-901-201 Attachment 1 Revision 6
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO20 obj. 3 (As available)

Question Source: Bank # 02188a
Modified Bank # ____ (Note changes or attach parent)
New ____

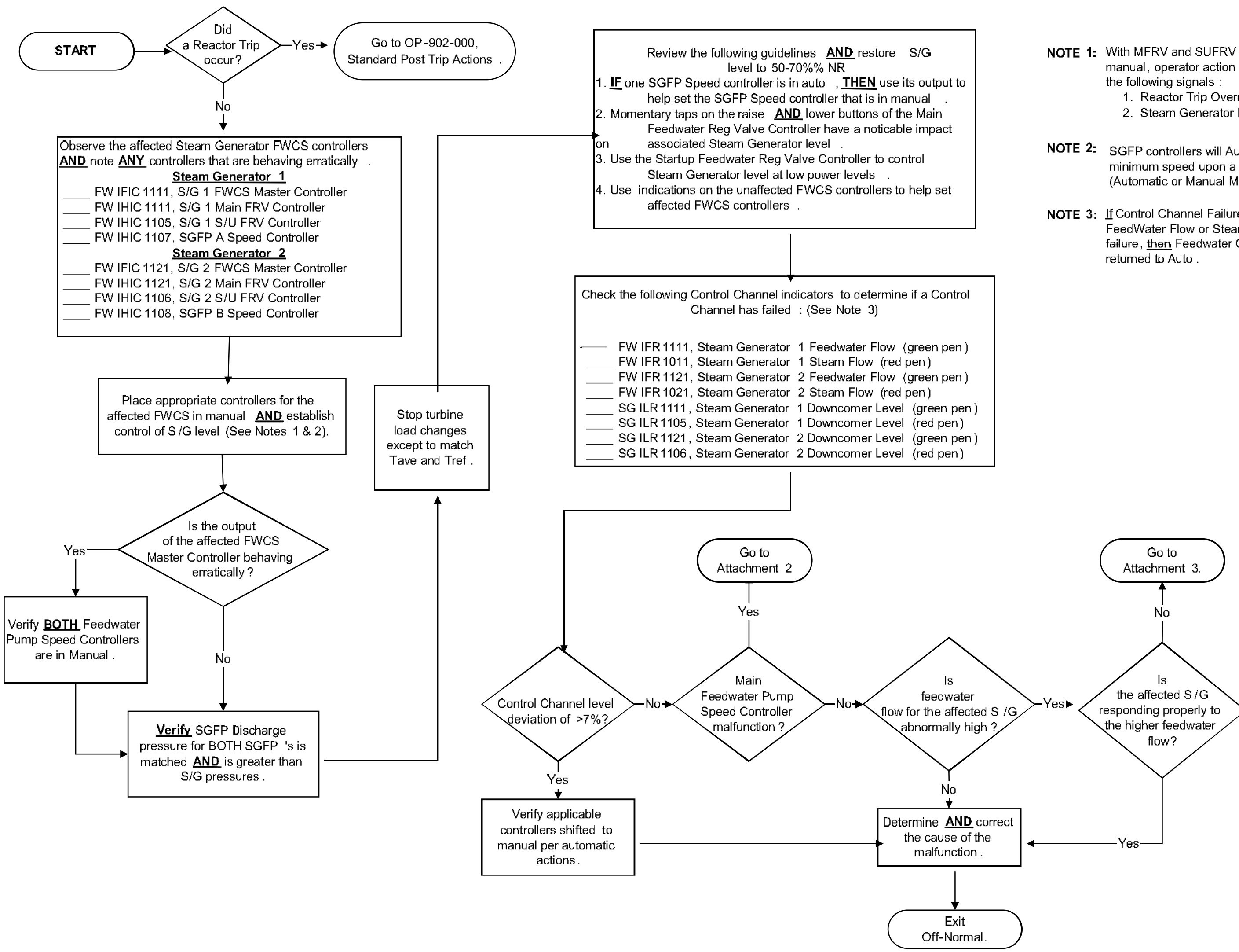
Question History: Last NRC Exam 2009 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 4,10
55.43 ____

Comments:

ATTACHMENT 1: GENERAL ACTIONS



NOTE 1: With MFRV and SUFRV controllers in manual, operator action will be required for the following signals :
1. Reactor Trip Override .
2. Steam Generator High Level Override

NOTE 2: SGFP controllers will Automatically go to minimum speed upon a Reactor Trip (Automatic or Manual Mode of Operation).

NOTE 3: If Control Channel Failure was due to FeedWater Flow or Steam Flow instrument failure, then Feedwater Controllers may be returned to Auto .

Examination Outline Cross-Reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 A1.05	
	Importance Rating	3.6	

K/A Statement

A1.05 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: AFW flow/motor amps

Proposed Question: RO 46 Rev: 0

Given:

- The reactor has tripped
- Pressurizer Pressure is 1800 PSIA and lowering
- SG #1 Level is 51% WR and lowering
- SG #2 Level is 43% WR and stable
- SG #1 Pressure is 645 PSIA
- SG #2 Pressure is 710 PSIA
- Containment Pressure is 15.2 PSIA and stable

The initial response of Emergency Feedwater flows to the Steam Generators is ____(1)____ gpm to Steam Generator #1 and ____(2)____ gpm to Steam Generator #2.

	(1)	(2)
A.	200	400
B.	200	200
C.	0	400
D.	0	200

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: There will be no flow to SG #1 because an EFAS 1 is blocked because SG pressure is less than 666 psia. The flow would be 200 gpm if there was an EFAS 1. For SG#2, flow will be 400 gpm EFW flow because an EFAS 2 is present and level has dropped below 45% WR.
B. Incorrect: There will be no flow to SG #1 because an EFAS 1 is blocked because SG pressure is less than 666 psia. The flow would be 200 gpm if there was an EFAS 1. For SG#2, flow will be 400 gpm EFW flow because an EFAS 2 is present and level has dropped below 45% WR. (would be 200 gpm between 55% and 45%) WR.
C. **CORRECT:** There will be no flow to SG #1 because an EFAS 1 is blocked because SG pressure is less than 666 psia. For SG#2, flow will be 400 gpm EFW flow because an EFAS 2 is present and level has dropped below 45% WR.
D. Incorrect: There will be no flow to SG #1 because an EFAS 1 is blocked when SG pressure is less than 666 psia. For SG#2, flow will be 400 gpm EFW flow because an EFAS 2 is present and level has dropped below 45% WR. (would be 200 gpm between 55% and 45%) WR.

Technical Reference(s): OP-009-003 Revision 305, Automatic Actions section.
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EFW00 obj. 6 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

9.0 AUTOMATIC FUNCTIONS

9.1 Emergency Feedwater Pump AB Turbine Electrical Overspeed Trip (EFW-IST-8350AB) .... 4895 RPM

9.2 Emergency Feedwater Pump AB Turbine Mechanical Overspeed Trip (EFW-MPMP-0001AB) .... 4930-4980 RPM

9.3 Main Steam to Emergency Feedwater Pump AB Turbine Drain Leg Level Hi to Alarm and Open Hi Drain Valve, MS-407, (MS-ILIS-0311) .... 8.0 INWC

9.4 Main Steam to Emergency Feedwater Pump AB Turbine Drain Leg Level Hi to Open Normal Drain Valve, MS-408, (MS-ILIS-0311) 5.5 INWC

9.5 EFAS-1 Train A or B Logic Initiated .... SG1 ■27.4% NR

SG1 ■666PSIA

or

SG1 ■27.4% NR

SG1 123 PSID >SG2

9.6 EFAS-2 Train A or B Logic Initiated .... SG2 ■27.4% NR

SG2 ■666 PSIA

or

SG2 ■27.4% NR

SG2 123 PSID >SG1

9.7 DEFAS Actuation .... DRTS signal present with the following:

- Both SG1 and SG2 WR levels $\leq$ 55%
- Both SG1 and SG2 pressures $\geq$750 PSIA
- No EFAS-1 or EFAS-2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 A4.01	
	Importance Rating	3.3	

K/A Statement

A4.01 - Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard)

Proposed Question: RO 47 Rev: 0

Given:

- An Excess Steam Demand has occurred
- SIAS, CIAS, and MSIS are present
- The crew is performing actions of OP-902-004, Excess Steam Demand
- Station loads are energized from the Startup transformers

The CRS has directed the ATC to stabilize RCS pressure.

To restore pressurizer back-up heaters, the crew will ________.

A. reset the back-up heater banks from CP-2

B. verify the 32A and 32 B feeder breakers closed on the sequencer after .5 seconds

C. manually close the 32A and 32B feeder breakers after 205 seconds

D. reset the SIAS, then close the 32A and 32B feeder breakers

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: After a SIAS, the 32 feeder breakers open and must be manually closed after 205 seconds. The back-up heaters banks are not required to be reset, only the proportional heaters.
B. Incorrect: The 32 feeder breakers close on the .5 second load block of the sequencer if an UV only has occurred.
C. **CORRECT:** After a SIAS, the 32 feeder breakers open and must be manually closed after 205 seconds.
D. Incorrect: The SIAS would be required to be reset if a UV and SIAS had occurred to close the 32 feeder breakers. In this instance, only a SIAS has occurred.

Technical Reference(s): TG-OP-902-009 appendix 25
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-ED00 obj. 7 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Appendix 25 Restore Pressurizer Heater Control

Objective

The intent of this Standard Appendix is restore power to the pressurizer heaters following a Loss of Offsite Power (LOOP) or a Safety Injection Actuation Signal (SIAS). Restoring the pressurizer heaters aids in maintaining the RCS Pressure Control Safety Function.

Instructions

Pressurizer level must be greater than the low level heater cutout in order to energize pressurizer heaters. The operator must also monitor EDG loading to prevent overloading the EDG with the Pzr heaters.

The sequencer is required to be timed out in order to close the 32 feeder breakers. The operator should reset all of the Pzr heaters and maintain pressure using the Pressurizer Pressure Control System in automatic or manually cycling pressurizer heaters and spray.

Contingency Actions

None

References

1. NUREG 0737, Clarification of TMI Action Plan Requirements Section
2. Section II.E.3.1 Emergency Power Supply for Pressurizer Heaters
3. Commitment P4115
4. ECS98-001 L.03

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 2.4.8	
	Importance Rating	3.8	

K/A Statement

2.4.8 - Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question: RO 48 Rev: 0

Given:

- The plant has experienced a Station Blackout
- The crew has entered OP-902-005, Station Blackout
- The crew has completed the step for reducing battery loads

The crew will monitor Spent Fuel Pool level using guidance located in ____________.

A. OP-500-008, Annunciator response procedure

B. OP-002-006, SFP Cooling and Purification

C. OP-901-513, SFP Cooling Malfunction

D. OP-902-008, Functional Recovery procedure

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: The note before step 22 of OP-902-005 states that SFP temperature and level annunciators are disabled once battery loads are stripped. The same note states that OP-901-513, SFP Malfunction, provides for local/alternate monitoring.
B. Incorrect: The note before step 22 of OP-902-005 states that SFP temperature and level annunciators are disabled once battery loads are stripped. The same notes states that OP-901-513, SFP Malfunction, provides for local/alternate monitoring. OP-002-006 is the normal operating procedure for the Spent Fuel Pool.
C. **CORRECT**: The note before step 22 of OP-902-005 states that SFP temperature and level annunciators are disabled once battery loads are stripped. The same notes states that OP-901-513, SFP Malfunction, provides for local/alternate monitoring.
D. Incorrect: The note before step 22 of OP-902-005 states that SFP temperature and level annunciators are disabled once battery loads are stripped. The same notes states that OP-901-513, SFP Malfunction, provides for local/alternate monitoring. The applicant may assume that this information is in OP-902-008.

Technical Reference(s): OP-902-005 revision 17
(Attach if not previously provided) OP-901-513 revision 9
(including version/revision number) OP-500-008 Attachment 4.92 revision 35

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE05 obj. 7 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10,13
55.43 ____

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS

Reset MSIS Initiation Setpoints

* 20. **IF** MSIS is **NOT** present, **THEN** lower the automatic initiation setpoints as the cooldown and depressurization proceed for MSIS (low SG Pressure).

Reset SIAS Initiation Setpoints

* 21. **IF** SIAS is **NOT** present, **THEN** lower the automatic initiation setpoints as the cooldown and depressurization proceed for SIAS (low PZR Pressure).

NOTE

SFP Temperature and Level annunciators are not functional when DC loads are stripped. OP-901-513, SFP Malfunction, provides for local/alternate monitoring.

Monitor Spent Fuel Pool

INSTRUCTIONS	CONTINGENCY ACTIONS
* 22. Monitor Spent Fuel Pool Time to 200ºF **AND** Level. a. Determine SFP Time to 200ºF using OP-901-513, SFP Malfunction. b. Monitor SFP level locally. c. Maintain TSC informed of SFP conditions.	22.1 For SFP alarm **OR** abnormal conditions refer to OP-901-513, SFP Malfunction.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 K2.03	
	Importance Rating	3.2	

K/A Statement

K2.03 - Knowledge of bus power supplies to the following: Control power

Proposed Question: RO 49 Rev: 0

Given:

- Plant is at 100% power
- EDG A is synchronized to the grid for a post-maintenance test
- EDG A output is 2.2 MW

If a loss of A-DC occurs, Emergency Diesel Generator A will ____(1)____ and the Emergency Diesel Generator A output breaker will _____(2)_____ .

	(1)	(2)
A.	trip	trip open immediately
B.	trip	remain closed
C.	remain running	trip open immediately
D.	remain running	remain closed

Proposed Answer: B

Explanation: (Optional)

A. Incorrect. The EDG will trip on a loss of control power (A-DC), the EDG output breaker does not trip because it has lost control power (loss of all auto trips).
B. **CORRECT:** The EDG will trip on a loss of control power (A-DC), the EDG output breaker does not trip because it has lost control power (loss of all auto trips).
C. Incorrect. The EDG will trip on a loss of control power (A-DC), the EDG output breaker does not trip because it has lost control power (loss of all auto trips).
D. Incorrect. The EDG will trip on a loss of control power (A-DC), the EDG output breaker does not trip because it has lost control power (loss of all auto trips).

Technical Reference(s): SD-DG Table 2
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EDG00 Obj 3 (As available)
WLP-OPS-ED00, Obj 2

Question Source: Bank # 08207
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 RO NRC Exam

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

10 CFR Part 55 Content: 55.41 5, 7
55.43

Comments:

TABLE 2 DIESEL ENGINE/GENERATOR TRIPS

TRIP SIGNAL	SETPOINT (ALARM SWITCH)	ACTUATING DEVICE
Low lube oil pressure	30 psig EGL-IPS-3019A (B)	PEV-14 EGL-IPEV-3014A(B)
Turbo low lube oil pressure	3 psig EGL-IPS-3017A (B)	PDEV-18 EGL-IPDEV-3018A(B)
Jacket water low pressure	5 psig EGC-IPS-3004A (B)	PEV-28 EGC-IPEV-3028A (B)
Turbocharger thrust bearing wear	0.011 inches(228°F) EG-ITS-3001A (B)	XEV-16 EG-IXEV-3016A (B)
Engine overspeed	660 rpm EGA-ISS-3009A1,A2 (B1,B2) EGF-ISS-3010A (B)	SSCV-6A, SSCV-6B EGF-ISSV-3006-1A (B)
Main bearing high temperature	228°F EG-ITS-3002A (B)	TEV-15
Connecting rod bearing high temperature	197°F EG-ITS-3002A (B)	TEV-15
Generator bearing high temperature	228°F EG-ITS-3004A (B)	TEV-19 EG-ITEV-3019A (B)
Jacket water high temperature	205°F EGC-ITS-3010A (B)	TEV-17 EGC-ITEV-3017A (B)
Diesel lube oil high temperature	185°F EGL-ITS-3003A (B)	TEV-31 EGL-ITEV-3031A (B)
Loss of field		EG-EREL-4766B1 (2)
Reverse power		EG-EREL-4766A1 (2)
Generator differential	0.14 amperes	EG-EREL-2316 (2366A) EG-EREL-2171I1 (2) EG-EREL-2172H1 (2)
Overcurrent with undervoltage	4 amperes with voltage < 95% of nominal	EG-EREL-4766F1 (2) EG-EREL-4766G1 (2) EG-EREL-4766H1 (2)
Loss of 125VDC		EG-ISV-3001A (B)

B_2 INDICATIONS

1. Loss of 125 Volt DC Bus A-DC
   - ■ A-DC BUS VOLTAGE (DC-IEI-7010A1) indicates zero (CP-1)
   - The following breakers on CP-1 will exhibit loss of indications:
     - EMERGENCY DIESEL GEN A (EGA 0001A)
     - BUS A3 TO A2 TIE BKR (4KV-3A-11S)
     - BUS TIE A TO AB (A3S1)
     - EMERGENCY DIESEL GEN BREAKER A (4KV-3A-14S)
     - SST A32 FEEDER (SSD-3A-8)
     - SST A315 FEEDER (SSD-3A-13S)
     - SST A31 FEEDER (SSD-3A-15S)
     - COMPUTER PRIMARY FEEDER (COMPUTER FDR PRIMARY)
     - SUT A TO BUS A1 FEEDER (A1-4)
     - UAT A TO BUS A1 FEEDER (A1-1)
     - A31 TO AB31 TIE BREAKER (A31S4B).
   - The following equipment will exhibit loss of indications:
     - ACCW PUMP A on CP-33
     - CHILLER A on CP-18
     - RAB NORMAL EXHAUST FAN A on CP-18
     - HPSI PUMP A on CP-8
     - LPSI PUMP A on CP-8
     - CS PUMP A on CP-8
     - CCW PUMP A on CP-8
     - EFW PUMP A on CP-8
     - MSIV (MS-124A) on CP-8
     - MFIV (FW-184A) on CP-8
     - EFW TO SG 1 PRIMARY ISOL (EFW 228A) on CP-8
     - EFW TO SG 2 BACKUP ISOL (EFW 229B) on CP-8
     - CNTMT CCW RET HDR INSIDE CNTMT ISOL (CC 710) on CP-8

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 K5.03	
	Importance Rating	2.9	

K/A Statement

K5.03 - Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Relationship between radiation intensity and exposure limits

Proposed Question: RO 50 Rev: 0

Given:

- Plant is at 100% power
- The following alarm is received on Cabinet L:
    - o A-9, RAD MONITORING SYS ACTIVITY HI-HI
- ARM-IRE-5021, Letdown HX/Blowdown Pump Hallway, is in alarm
- Chemistry sample of the RCS indicates elevated Iodine-131 activity levels

Which of the following describes the reason for notification of Radiation Protection to initiate surveys in the Reactor Auxiliary Building?

A. Area radiation levels may require additional controls.

B. Surveys are required for confirmation of ARM-IRE-5021 detector response.

C. ARM-IRE-5021 reads in CPM and therefore has no correlation to area radiation levels.

D. Area radiation levels are used to determine the need for additional letdown flow.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** Surveys of the area or taken because additional controls may be required due to reduce exposure to personnel.

B. Incorrect. In this case, confirmation is provided by Chemistry sample.

C. Incorrect. ARM reads in mr/hr, and is directly related to changes in radiation levels. CPM would be the units if activity were being measured.

D. Incorrect. Letdown flow is maximized based on RCS activity readings, not area radiation levels.

Technical Reference(s): OP-901-410, Page 6, Step 4 8 & 9, Step 6
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PP040 obj. 3 (As available)

Question Source: Bank # X
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10,11,12
55.43

Comments:

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

	PLACEKEEPER		
	START	DONE	N/A
1. Request Chemistry Department to sample and analyze the following:	☐	☐	
• RCS for Dose Equivalent I-131 concentration and isotopics		☐	
• RCS for gross activity		☐	
• Purification Ion Exchanger influent and effluent to determine the Decontamination Factor (D.F.)		☐	
2. Advise the Shift Manager to perform the following:	☐	☐	
• Implement EP-001-001, Recognition and Classification of Emergency Conditions		☐	
• Refer to NF-102, Corporate Fuel Reliability		☐	
3. Using the Plant Paging System, make the following announcement twice:	☐	☐	
3.1 "ATTENTION ALL PERSONNEL, ATTENTION ALL PERSONNEL. RISING RADIATION LEVELS ARE IN THE REACTOR AUXILIARY BUILDING (state affected areas). ALL PERSONNEL EXIT THESE AREAS AND PROCEED TO A RADIATION PROTECTION CONTROL POINT."		☐	
4. Advise Radiation Protection of the possibility of higher dose levels in the Reactor Auxiliary Building and to consider additional controls.		☐	

E$_0$ GENERAL (CONT'D)

	PLACEKEEPER		
	START	DONE	N/A
5. If the specific activity of the primary coolant is > 1.0 microcuries/gram Dose Equivalent I-131 or > 100/E microcuries/gram, then perform the following:	☐	☐	☐
• Comply with the actions of Technical Specification 3.4.7, Reactor Coolant System Specific Activity.	☐	Continuous ↓	
• Comply with the sampling frequency of Table 4.4-4.	☐	Continuous ↓	
5.1 If a Plant Shutdown is required, then perform OP-010-005, Plant Shutdown, concurrently with this Procedure.	☐	Continuous ↓	
6. If sample analysis indicates exhaustion of Purification Ion Exchanger (D.F. less than 1.0), then perform the following:	☐	☐	☐
6.1 Isolate the in service Purification Ion Exchanger in accordance with OP-002-005, Chemical and Volume Control.	☐	☐	☐
6.2 Place standby Purification Ion Exchanger in service in accordance with OP-002-005, Chemical and Volume Control.	☐	☐	☐
6.3 If standby Purification Ion Exchanger is not available, then contact Chemistry and Radiation Protection to expedite flushing and replacement of resin in Purification Ion Exchanger.	☐	☐	☐
7. If purification system is operating properly, then maximum Letdown flow should be established as follows:	☐	☐	☐
7.1 Start one available Backup Charging Pump by placing Control Switch to ON.		☐	
7.2 Verify that COLSS Charging and Letdown constants are correct for Charging Pump configuration in accordance with OP-004-005, Core Operating Limits Supervisory System Operation.		☐	

E0 GENERAL (CONT'D)

		START	DONE	N/A
7.3	Monitor Pressurizer Level and Pressure, allowing time for stabilization.	☐	Continuous ↓	
7.4	Verify Letdown Backpressure Controller, CVC-IPIC-0201, maintains approximately 460 PSIG.	☐	Continuous ↓	
7.5	Verify Letdown Temperature Controller, CVC-ITIC-0223, maintains approximately 120°F.	☐	Continuous ↓	
7.6	<u>If</u> available, <u>then</u> start second Backup Charging Pump by placing Control Switch to ON.		☐	☐
7.6.1	Verify that COLSS Charging and Letdown constants are correct for Charging Pump configuration in accordance with OP-004-005, Core Operating Limits Supervisory System Operation.		☐	☐
7.7	Monitor Pressurizer Level <u>and</u> Pressure, allowing time for stabilization.	☐	Continuous ↓	
7.8	Verify Letdown Backpressure Controller, CVC-IPIC-0201, maintains approximately 460 PSIG.	☐	Continuous ↓	
7.9	Verify Letdown Temperature Controller, CVC-ITIC-0223, maintains approximately 120°F.	☐	Continuous ↓	
8.	Continue to comply with the sampling frequency of Technical Specification Table 4.4-4.	☐	Continuous ↓	☐
9.	<u>When</u> maximum Letdown flow is no longer required, <u>then</u> return Charging system to desired number of pumps operating.	☐	☐	

END

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 K1.20	
	Importance Rating	3.4	

K/A Statement

K1.20 – Knowledge of the physical connections and/or cause- effect relationships between the SWS and the following systems: AFW

Proposed Question: RO 51 Rev: 0

Given:

- Reactor tripped due to a loss of Main Feedwater
- Crew entered OP-902-006, Loss of Main Feedwater Recovery
- Condensate Storage Pool (CSP) level is 35% and lowering with CMU-141, CSP LCV Bypass, open

To prevent cavitation of the EFW Pumps, the transfer of EFW Pump suction to ____(1)____ side(s) of Auxiliary Component Cooling Water (ACCW) must be completed before CSP level lowers to ____(2)____.

	(1)	(2)
A.	both	11%
B.	both	25%
C.	one	11%
D.	one	25%

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: OP-902-009 Appendix 10 states that 11% CSP level is the level at which EFW pump suction transfer should be completed to prevent from cavitating the EFW pump. Step 1.1 directs the crew to align the suction to only one side.

B. Incorrect: OP-902-009 Appendix 10 states 11% CSP level is the level at which EFW pump suction transfer should be completed to prevent from cavitating the EFW pump. 25% level in the CSP is when the procedure directs the transfer to take place. Any operating train of ACCW is aligned to EFW. Step 1.1 directs the crew to align the suction to only one side.

C. **CORRECT**: OP-902-009 Appendix 10 states 11% CSP level is the level at which EFW pump suction transfer should be completed to prevent from cavitating the EFW pump. Any operating train of ACCW is aligned to EFW.

D. Incorrect: OP-902-009 Appendix 10 states 11% CSP level is the level at which EFW pump suction transfer should be completed to prevent from cavitating the EFW pump. 25% level in the CSP is when the procedure directs the transfer to take place. Any operating train of ACCW is aligned to EFW.

Technical Reference(s): OP-902-006, Loss of Main Feedwater Recovery, Rev. 13

(Attach if not previously provided) OP-902-009, Standard Appendices, Appendix 10, Rev. 307

(including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE06 obj. 9 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS

Makeup to CSP

* 30. Maintain CSP level greater than 92%:

a. Locally throttle open CMU 141, Condensate Storage Pool LCV Bypass.

b. **IF** CSP level is less than 25%, **THEN** REFER TO Appendix 10, "Transferring EFW Pump Suction", and transfer EFW pump suction to **ONE** operating train of ACCW.

Cooldown NOT Desired

* 31. **IF** a plant cooldown is **NOT** desired, **THEN**:

a. Maintain the plant in a stabilized condition.

b. **WHEN BOTH** of the following conditions are met,

- At least one RCP is operating
- MFW is available to the steam generators

THEN GO TO the appropriate general operating procedure.

Transferring EFW Pump Suction

INSTRUCTIONS CONTINGENCY ACTIONS

----- **NOTE** -----

CSP Indicated level will be lower than actual when drawing suction from the CSP. CSP Indicated level will be higher than actual when drawing suction from the ACCW system. When EFW suction is drawn from the CSP, consideration should be given to reducing flow to less than 500 gpm to read CSP level.

----- **NOTE** -----

Transfer of EFW Pump suction should be completed by a CSP level of 11% to prevent cavitation of EFW Pumps.

INSTRUCTIONS	CONTINGENCY ACTIONS

____ 1.1 Transfer Emergency Feedwater Pump suction to **ONE** side of the Auxiliary Component Cooling System as follows:

Train A

a. Verify Auxiliary Component Cooling Water Pump A operating.

b. Close ACC 115A, Auxiliary Component Cooling Header A to Emergency Feedwater Drain.

c. Unlock and open the following valves:

- ACC 116A, Auxiliary Component Cooling Header A to Emergency Feedwater Isolation
- ACC 114A, Auxiliary Component Cooling Header A Supply to EFW Header Isolation

(continue)

INSTRUCTIONS

CONTINGENCY ACTIONS

____ 1.1 (continued)

Train B

a. Verify Auxiliary Component Cooling Water Pump B operating.

b. Close ACC 115B, Auxiliary Component Cooling Header B to Emergency Feedwater Drain.

c. Unlock and open the following valves:

- ACC 116B, Auxiliary Component Cooling Header B to Emergency Feedwater Isolation
- ACC 114B, Auxiliary Component Cooling Header B Supply to EFW Header Isolation

____ 1.2 **WHEN** 30 minutes has elapsed, **THEN** close and lock the valves for the Train aligned in step 1:

- ACC 116A
- ACC 116B
- ACC 114A
- ACC 114B

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 1.3 Open Auxiliary Component Cooling Header to Emergency Feedwater Drain for the Train aligned in step 1: • ACC 115A • ACC 115B	

End of Appendix 10

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 K3.07	
	Importance Rating	3.7	

K/A Statement

K3.07 - Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads

Proposed Question: RO 52 Rev: 0

Given:

- A Loss Of Coolant Accident (LOCA) is in progress
- Auxiliary Component Cooling Water (ACCW) Pump A tripped on overcurrent
- Component Cooling Water temperature is 90°F and rising
- All Dry Cooling Tower A fans are in SLOW speed operating in AUTO

If Component Cooling Water System temperature continues to rise, all Dry Cooling Tower A fans start simultaneously in FAST speed when CCW temperature reaches ____(1)____ and Essential Chillers will swap to the WET tower mode when CCW temperature reaches _____(2)_____ .

	(1)	(2)
A.	100 °F	95 °F
B.	92 °F	102 °F
C.	100 °F	102 °F
D.	92 °F	95 °F

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: All Dry cooling tower fans start simultaneously when CCW temperature reaches a setpoint of 100 °F. The essential chillers swap to the WET mode when CCW temperature reaches 102 °F. 95 °F is the temperature at which the essential chillers swap back to the dry mode.
B. Incorrect: 92 °F is the temperature at which DCT fans sequence on at fast speed. The essential chillers swap to the WET mode when CCW temperature reaches 102 °F.
C. **CORRECT**: All Dry cooling tower fans start simultaneously when CCW temperature reaches a setpoint of 100 °F. The essential chillers swap to the WET mode when CCW temperature reaches 102 °F.
D. Incorrect: 92 °F is the temperature at which DCT fans sequence on at fast speed. 95 °F is the temperature at which the essential chillers swap back to the dry mode.

Technical Reference(s): OP-901-510 revision 301
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CC00 obj. 12 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

C AUTOMATIC ACTIONS

1. At 70% dropping CCW Surge Tank level, CMU-226, WATER STORAGE MAKEUP CCW SURGE TANK, Opens.

2. At 52% dropping CCW A(B) Surge Tank Level, CMU-538A(B), CCW MAKEUP VALVE A(B), Opens.

NOTE 3

1. WITH CCW Makeup Pumps aligned for Automatic Operation, the CCW Makeup Pumps will start on a low CCW Surge Tank Level (20%) AND automatically stop three minutes later OR on a high level (52%).
2. IF the CCW Makeup Pump(s) are automatically stopped by the three minute timer AND CCW Surge Tank Level is NOT ≥ 24%, THEN further Operation of the CCW Makeup Pump(s) will require placing the Control Switch to START.

3. At 20% dropping CCW A(B) Surge Tank level, the following actions occur:

• CMU-0004A(B)	CCW MAKEUP PUMP A(B)	Starts
• CC-134A(B)	CCW A(B) DRY COOLING TOWER BYPASS	Opens
• CC-135A(B)	CCW A(B) DRY COOLING TOWER ISOLATION	Closes

C AUTOMATIC ACTIONS (Continued)

NOTE 4

1. CCW Suct & Disch Header Tie Valves AB to A remain Open when CCW ASSIGNMENT Switch is aligned to A.
2. CCW Suct & Disch Header Tie Valves AB to B remain Open when CCW ASSIGNMENT Switch is aligned to B.

4. At 16% dropping CCW A(B) Surge Tank level, the following valves Close, to split the A AND B CCW trains:
   - CC-126A/CC-114A — CCW SUCT & DISCH HEADER TIE VALVES AB TO A
   - CC-127A/CC115A — CCW SUCT & DISCH HEADER TIE VALVES AB TO A
   - CC-126B/CC-114B — CCW SUCT & DISCH HEADER TIE VALVES AB TO B
   - CC-127B/CC-115B — CCW SUCT & DISCH HEADER TIE VALVES AB TO B
   - CC-200A/CC-727 — CCW SUCT & DISCH HEADER TIE VALVES A TO AB
   - CC-200B/CC-563 — CCW SUCT & DISCH HEADER TIE VALVES B TO AB
5. IF loss of CCW Pump(s) occurs due to loss of Vital Bus, THEN applicable pump will start 7 seconds after bus is energized by the Emergency Diesel Generator.
6. WHEN CCW temperature ≥ 92°F, THEN CCW Dry Cooling Tower Fans in Auto start sequentially in Slow speed at 60 second intervals.
7. WHEN ALL CCW Dry Cooling Tower Fans in AUTO are operating in Slow speed AND CCW temperature ≥ 92°F, THEN CCW Dry Cooling Tower Fans shift sequentially to Fast speed at 60 second intervals.
8. WHEN CCW temperature > 100°F, then applicable Auxiliary Component Cooling Water Pump Starts AND ALL CCW Dry Cooling Tower Fans in Auto shift to Fast speed.

C AUTOMATIC ACTIONS (Continued)

9. WHEN CCW temperature > 102°F, THEN Essential Chillers shift from Dry Tower mode to Wet Tower mode.

10. High Reactor Coolant Pump CCW Return temperature (155°F) actuates automatic isolation of applicable RCP Seal Cooler by Closing the following valves:

- CC-679A/CC-6651A 1A RCP Seal Cooler
- CC-679B/CC-6651B 1B RCP Seal Cooler
- CC-680A/CC-666A 2A RCP Seal Cooler
- CC-680B/CC-666B 2B RCP Seal Cooler

END

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 K1.02	
	Importance Rating	2.7	

K/A Statement

K1.02 – Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Service air

Proposed Question: RO 53 Rev: 0

Given:

- Plant is at 100% power
- Instrument Air Header pressure is 115 PSIG

An air leak occurs, causing Instrument Air Header pressure to drop to 92 PSIG. The crew enters OP-901-511, Instrument Air Malfunction.

The crew will verify that SA-125, SA Backup Supply for IA Press Cntl valve, is ____(1)____ and IA-123, Instrument Air Dryer Bypass Solenoid valve, is ____(2)____ .

	(1)	(2)
A.	open	open
B.	closed	open
C.	open	closed
D.	closed	closed

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** SA-125, SA Backup Supply for IA Press Cntl valve opens at 105 psig. Instrument Air Dryer Bypass Solenoid valve opens at 95 psig. With instrument Air pressure at 92 psig, both valves will be open.
B. Incorrect: SA-125, SA Backup Supply for IA Press Cntl valve opens at 105 psig. Instrument Air Dryer Bypass Solenoid valve opens at 95 psig. With instrument Air pressure at 92 psig, both valves will be open.
C. Incorrect: SA-125, SA Backup Supply for IA Press Cntl valve opens at 105 psig. Instrument Air Dryer Bypass Solenoid valve opens at 95 psig. With instrument Air pressure at 92 psig, both valves will be open.
D. Incorrect: SA-125, SA Backup Supply for IA Press Cntl valve opens at 105 psig. Instrument Air Dryer Bypass Solenoid valve opens at 95 psig. With instrument Air pressure at 92 psig, both valves will be open.

Technical Reference(s): OP-901-511 revision 14
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-AIR obj. 5 (As available)
WLP-OPS-PP050 obj. 3

Question Source: Bank # 08217
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2008 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

C AUTOMATIC ACTIONS

NOTE

Safety class valves listed in Attachment 1, Valves Supplied with N_2/Instrument Air, and Attachment 2, Valves Supplied with Air Accumulators, have accumulators for operation after failure of Instrument Air System. Nitrogen accumulators have provisions for nitrogen makeup for extended periods of operation without instrument air.

1. The following are occurring as Instrument Air pressure is dropping:
   - SA Backup Supply for IA Press Cntl valve (SA-125) opens at SA-IPIC-9821 setpoint (normally >105 PSIG)
   - Instrument Air Compressor selected for standby starts at 105 PSIG Instrument Air Receiver pressure
   - Instrument Air Dryers Bypass Solenoid valve (IA-123) Opens at 95 PSIG Instrument Air Dryer Outlet pressure (IA-IPS-9719).

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 A4.01	
	Importance Rating	3.1	

K/A Statement

A4.01 - Ability to manually operate and/or monitor in the control room: Pressure gauges.

Proposed Question: RO 54 Rev: 0

On a Loss of TGB-DC Bus, the reactor will be manually tripped when Instrument Air header pressure lowers to ____(1)___ due to a loss of power to _____(2)_____.

	(1)	(2)
A.	80	IA and SA compressor unloader valves
B.	80	Instrument Air Dryers
C.	65	IA and SA compressor unloader valves
D.	65	Instrument Air Dryers

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: The reactor will not be tripped until 65 psig (not 80) per OP-901-313 step 13. The 80 psig value is when a plant shutdown must occur. The reason for the loss of instrument air is presented in a caution statement preceding step 13.
B. Incorrect: The reactor will not be tripped until 65 psig (not 80) per OP-901-313 step 13. The 80 psig value is when a plant shutdown must occur. The reason for the loss of instrument air is presented in a caution statement preceding step 13. However, IA dryer malfunctions can significantly affect IA pressure making it plausible.
C. **CORRECT:** The reactor will not be tripped until 65 psig per OP-901-313 step 13. The reason for the loss of instrument air is presented in a caution statement preceding step 13.
D. Incorrect: The reactor will not be tripped until 65 psig per OP-901-313 step 13. The reason for the loss of instrument air is presented in a caution statement preceding step 13. However, IA dryer malfunctions can significantly affect IA pressure making it plausible.

Technical Reference(s): OP-901-313 revision 303
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO30 obj. 3 (As available)

Question Source: Bank # X Question #54
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

E$_4$ LOSS OF 125 VOLT DC BUS TGB-DC (CONT'D)

CAUTION

AUTO VOLTAGE REGULATOR OPERATION MAY BE ERRATIC OR LOST. INFORM PINE BLUFF LOAD DISPATCHER UNIT WILL BE COMING OFF LINE AND MAY EXPERIENCE AUTO VOLTAGE REGULATOR PROBLEMS.

	PLACEKEEPER START	DONE	N/A
10.6 Verify all equipment serviced by TGB-DC bus is restored.	☐	☐	
10.7 Exit this procedure.		☐	
11. If bus TGB-DC cannot be re-energized, then conduct a Rapid Plant Power Reduction in accordance with OP-901-212 and cooldown to Mode 5 in accordance with OP-010-005, Plant Shutdown, in conjunction with this procedure.	☐	☐	☐
12. If a Reactor Trip occurs, initiate a MSIS, then refer to OP-902-000, Standard Post Trip Actions, and perform concurrently with this procedure.	☐	Continuous ■	

CAUTION

(1) POWER WILL BE LOST TO ALL STATION AIR COMPRESSORS AND INSTRUMENT AIR COMPRESSORS UNLOADER VALVES. THIS WILL RESULT IN A LOSS OF INSTRUMENT AIR AND STATION AIR.

(2) THE MAIN TURBINE CANNOT BE TRIPPED FROM CP-1 AND CAN ONLY BE TRIPPED LOCALLY.

(3) THE MAIN FEEDWATER PUMPS WILL NOT GO ON THE TURNING GEAR.

(4) IF TGB-DC POWER CANNOT BE RESTORED, THEN ALL MAIN STEAM BYPASS VALVES CANNOT BE OPENED FROM CP-1 AND WILL NOT OPEN ON A REACTOR TRIP OR CUTBACK.

	START	DONE	N/A
13. If Instrument Air pressure drops to 65 psig, then Trip the Reactor, initiate a MSIS, and refer to OP-902-000, Standard Post Trip Actions, and perform concurrently with this procedure.	☐	Continuous ■	

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 A3.01	
	Importance Rating	3.9	

K/A Statement

A3.01 - Ability to monitor automatic operation of the containment system, including: Containment isolation

Proposed Question: RO 55 Rev: 0

Given:

- Containment Pressure is 16.9 PSIA
- RCS Pressure is 1600 PSIA
- CC-641, CCW RCP Inlet Outside Isol Valve is OPEN
- BAM-133, Emergency Boration Valve is OPEN
- SI-343, SI Tanks Drain to RWSP Valve is CLOSED
- MS-120A, Main Steam Line 1 Normal Drain Valve is OPEN

Which ONE of the following valves requires operator action for the current plant conditions?

A. CC-641

B. BAM-133

C. SI-343

D. MS-120A

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: CC-641 closes on a CSAS. The present conditions indicate only a SIAS and CIAS are present.
B. Incorrect: BAM-133 should be open on a SIAS.
C. Incorrect: SI-343 closes on a CIAS.
D. **CORRECT:** MS-120A should have closed on a CIAS and will require manual action to close it.

Technical Reference(s): OP-902-009 Appendix 4
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 6 (As available)

Question Source: Bank # 08261
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Component Number	Component Noun Name	Location	Action	Verified
CP-8 Train A CCW				
CC 0001A	CCW Pump A	CP-8	Start	
CC 126A/ 114A	CCW Suct & Disch Header Tie Vlvs AB to A	CP-8	Close (1)	
CC 126B/ 114B	CCW Suct & Disch Header Tie Vlvs AB to B	CP-8	Close (4)	
CC 501	CCW NNS Loop Supply Isol	CP-8	Close	
CP-4 Train A				
CVC 0001A	Charging Pump A	CP-4	Start	
BAM 0001A	Boric Acid Pump A	CP-4	Start	
BAM 0001B	Boric Acid Pump B	CP-4	Start	
BAM 126A	Boric Acid Makeup Pump A Recirc Valve	CP-4	Close	
BAM 126B	Boric Acid Makeup Pump B Recirc Valve	CP-4	Close	
BAM 133	Emergency Boration Valve	CP-4	Open	
CVC 510	VCT Makeup Valve	CP-4	Close	
CVC 101	Cntmt Isol Valves Letdown Stop Valve	CP-4	Close	
CP-1 Train A				
EGA 0001A	Emergency Diesel Gen A	CP-1	Start	

(1) - **IF** CCW Pump AB is replacing CCW Pump A, **THEN** verify valve is open.
(4) - **IF** CCW Pump AB is replacing CCW Pump B, **THEN** verify valve is open.

ESFAS Auto Actions

Attachment 4-D: CIAS Automatic Actions

Component Number	Component Noun Name	Location	Action	Verified
CP-8 Train A				
MS 120A	Main Steam Line 1 Drains Normal	CP-8	Close	
MS 119A	Main Steam Line 1 Drains Bypass	CP-8	Close	
PSL 107	RCS Sample Isol Hot Leg (Out)	CP-8	Close	
PSL 204	RCS Sample Isol Pzr Surge (Out)	CP-8	Close	
PSL 304	RCS Sample Isol Pzr Steam (Out)	CP-8	Close	
BD 103A	SG Blowdown Isol Stm Gen 1 (Out)	CP-8	Close	
BD 103B	SG Blowdown Isol Stm Gen 2 (Out)	CP-8	Close	
SI 343	SI Tanks Drain to RWSP	CP-8	Close	
SSL 8006A	Sampling Isolation SG 1	CP-8	Close	
SSL 8006B	Sampling Isolation SG 2	CP-8	Close	
SP 106	Cntmt Isolation Sump Pumps Outlet	CP-8	Close	
FP 601A	Cntmt Isolation Fire Water A	CP-8	Close	
IA 909	Cntmt Isolation Instrument Air	CP-8	Close	
SI 14023A	LPSI A to RC Loop 2B Upstr Auto Vent Contmt Isol	CP-8	Close	
SI 14024A	LPSI A to RC Loop 2B Upstr Auto Vent Auto Isol	CP-84	Close	

ESFAS Auto Actions

Attachment 4-F: CSAS Automatic Actions

Component Number	Component Noun Name	Location	Action	Verified
CP-8 Train A				
CS 0001A	Cntmt Spray Pump A	CP-8	Start	
CS 125A	Cntmt Spray Header A Isol	CP-8	Open	
CC 963A	CCW Shdn Hx A Outlet	CP-8	Open	
CC 200A/ 727	CCW Suct & Disch Header Tie Valves A to AB	CP-8	Close	
CC 710	CCW RCP Outlet Inside Isol	CP-8	Close	
CP-8 Train B				
CS 0001B	Cntmt Spray Pump B	CP-8	Start	
CS 125B	Cntmt Spray Header B Isol	CP-8	Open	
CC 713	CCW RCP Outlet Outside Isol	CP-8	Close	
CC 641	CCW RCP Inlet Outside Isol	CP-8	Close	

End of Appendix 4

Examination Outline Cross-Reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001 K4.05	
	Importance Rating	3.9	

K/A Statement

K4.05 - Knowledge of CRDS design feature(s) and/or interlock(s) which provide for the following: Boration and dilution.

Proposed Question: RO 56 Rev: 0

Given:

- Plant is at 90% power
- An inadvertent dilution event is in progress
- The crew entered OP-901-104, Inadvertent Positive Reactivity Addition
- The ATC reports that “Reactor Coolant Tavg-Tref Hi” annunciator is in
- Steam Bypass Control Valves are closed

The crew will verify a ____(1)____ signal being sent to CEDMCS. This signal will prevent withdrawal of _____(2)_____ control element assemblies in the Auto Sequential mode.

	(1)	(2)
A.	Automatic Withdrawal Prohibit (AWP)	regulating group
B.	Automatic Withdrawal Prohibit (AWP)	all
C.	Automatic Motion Inhibit (AMI)	regulating group
D.	Automatic Motion Inhibit (AMI)	all

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: RCS Tave-Tref High alarms when Tave is 6.8°F higher than Tref. An AWP signal is generated at this same setpoint and will prevent withdrawal of regulating group CEAS in the AS mode.
B. Incorrect: RCS Tave-Tref Low alarms when Tave is 6.8°F higher than Tref. An AWP signal is generated at this same setpoint and will prevent withdrawal of regulating group CEAS in the AS mode.
C. Incorrect: RCS Tave-Tref Low alarms when Tave is 6.8°F higher than Tref. An AWP signal is generated at this same setpoint and will prevent withdrawal of regulating group CEAS in the AS mode. An AMI is generated on a SBCS demand signal.
D. Incorrect: RCS Tave-Tref Low alarms when Tave is 6.8°F higher than Tref. An AWP signal is generated at this same setpoint and will prevent withdrawal of regulating group CEAS in the AS mode. An AMI is generated on a SBCS demand signal.

Technical Reference(s): OP-500-008 att. 4.81,4.97 revision 26
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CED00 obj. 5 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 6,7
55.43 ____

Comments:

REACTOR COOLANT
TAVG-TREF HI

INITIATING DEVICE	SETPOINT
K-6 Relay on selected RRS	6.8°F TAVG > TREF

POSSIBLE EFFECTS

1. AWP sent to CEDMCS
2. Rise in Pressurizer Level

CONTROL ROOM INDICATIONS	LOCAL INDICATIONS
TAVG-TREF Recorder TAVG-TREF Hi amber light on drawer	NONE

POSSIBLE CAUSES	RECOMMENDED ACTIONS
1. Power mismatch (primary > secondary)	1.1 Take corrective action to match primary and secondary plant power.
2. Failed RTD	2.1 Check T_H and T_C inputs to RRS.
	2.2 Refer to OP-901-110, Pressurizer Level Control Malfunction.

CEA AUTO
WITHDRAWAL PROHIBIT

NOTE
AWP signal prevents regulating CEA withdrawal in Automatic Sequential mode.

INITIATING DEVICE	SETPOINT
AWP Relay	N/A

POSSIBLE EFFECTS

NONE

CONTROL ROOM INDICATIONS	LOCAL INDICATIONS
D36600 CEDMCS - CEA AWP	NONE
Tc indication	
SBCS indication, Red AWP light on SBCS Cabinet	

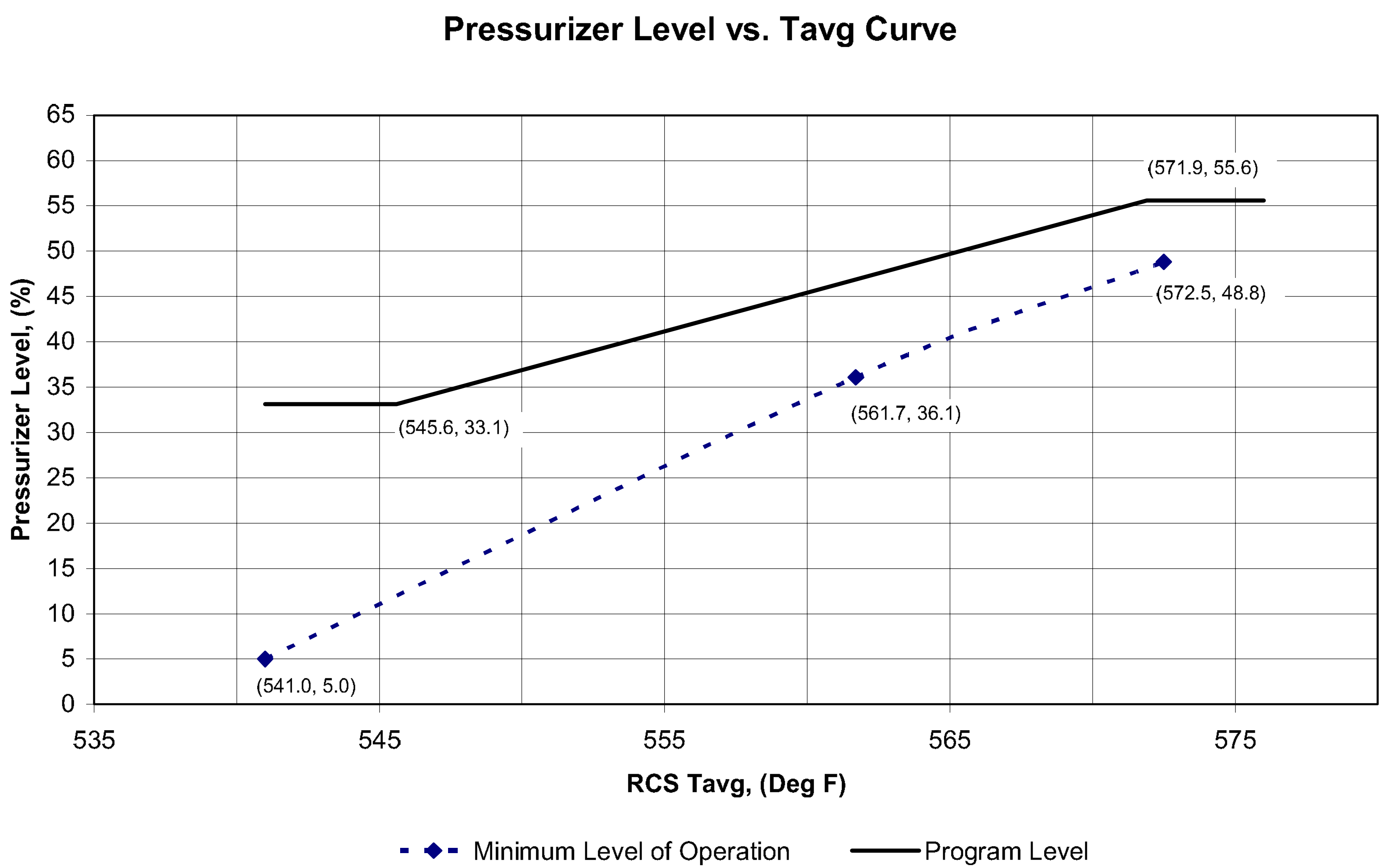
POSSIBLE CAUSES	RECOMMENDED ACTIONS
1. Reactor coolant $T_{AVG} > T_{REF}$ by 6.8°F.	1.1 Verify no Auto-Withdrawal in progress for regulating CEA groups. 1.2 Take action to match T_{AVG} and T_{REF}.

CEA AUTO
WITHDRAWAL PROHIBIT

POSSIBLE CAUSES	RECOMMENDED ACTIONS
2. Tc > 548°F	2.1 Verify Steam Bypass Control System operating properly to remove excess energy from RCS.
3. Steam Bypass Permissive and Demand signals present	3.1 Verify Steam Bypass Valves in proper configuration for signals present.
4. AMI signal present	4.1 Verify no Auto Motion in progress for regulating CEA groups.

ATTACHMENT 1: PRESSURIZER LEVEL VERSUS TAVE CURVE

(Reference: Calc. CN-OA-09-12)



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011 A3.02	
	Importance Rating	2.6	

K/A Statement

A3.02 - Ability to monitor automatic operation of the PZR LCS, including: Reactor power

Proposed Question: RO 57 Rev: 0

Given:

- Plant is steady state at 75% power
- Tavg is being maintained on program
- A Letdown malfunction has occurred, requiring letdown flow control valves to be placed in manual

The CRS directs Pressurizer level to be restored to program level prior to placing the letdown flow control valves back in auto. What is the target level?

A. 45%

B. 46.5%

C. 49.5%

D. 51.5%

Proposed Answer: C

Explanation: (Optional)
A. Incorrect: 45% corresponds to Tave of 559.5°F.
B. Incorrect: 46.5% corresponds to Tave of 561°F.
C. **CORRECT**: 75% power corresponds to 564°F Tave On Fig 1. Corresponds to 49% PZR level on Fig 3.
D. Incorrect: 51.5% corresponds to Tave of 567°F.

Technical Reference(s): OP-901-112 revision 5
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: OP-901-112 attachment 1

Learning Objective: WLP-OPS-PPO10 odj. 3 (As available)

Question Source: Bank # 07927
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2006 NRC RO Exam

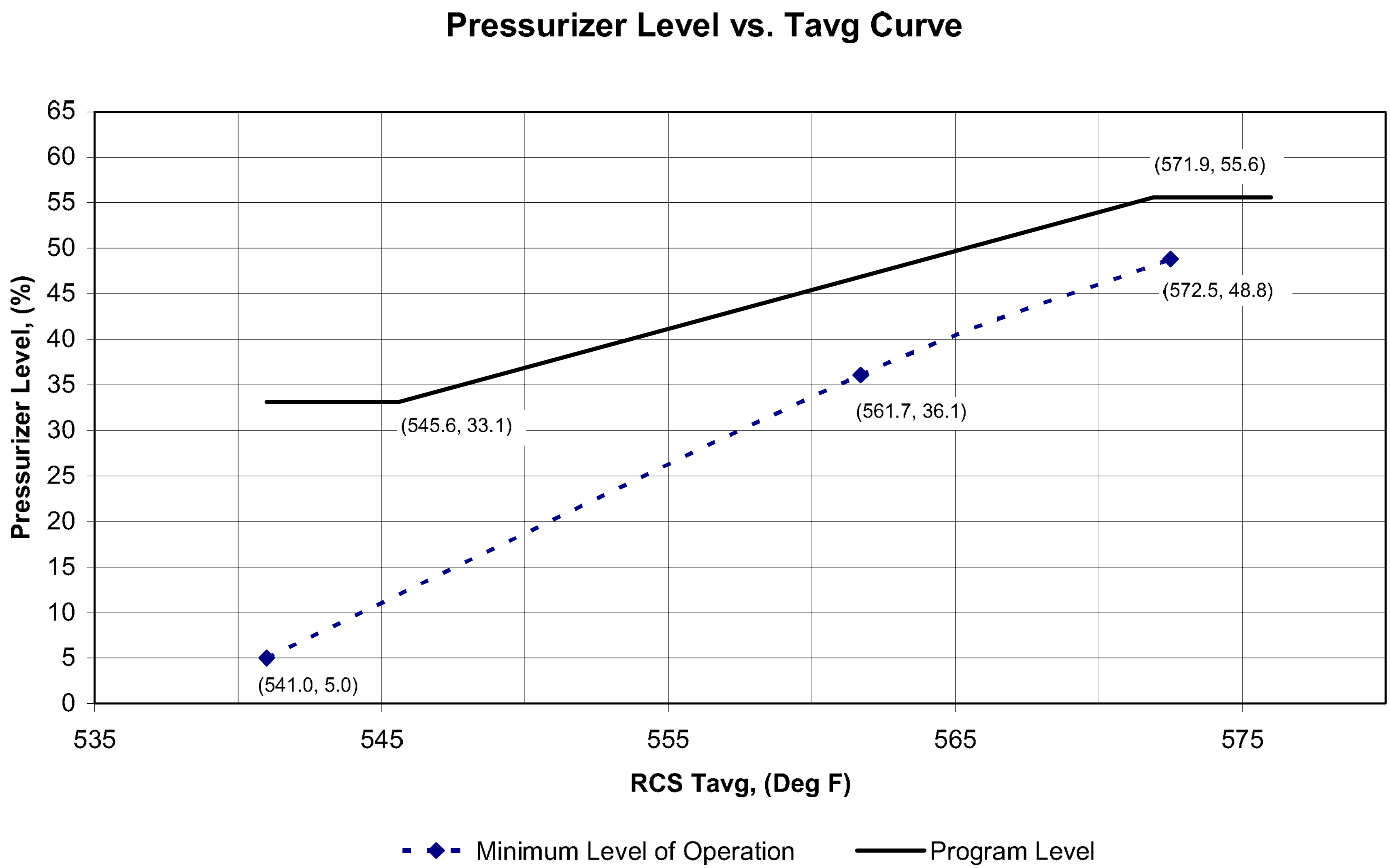
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7,10
55.43

Comments:

ATTACHMENT 1: PRESSURIZER LEVEL VERSUS TAVE CURVE

(Reference: Calc. CN-OA-09-12)



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015 K6.03	
	Importance Rating	2.6	

K/A Statement

K6.03 - Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Component interconnections

Proposed Question: RO 58 Rev: 0

Given:

- Plant power is 100%
- Safety Channel 1 and 2 High Voltage On Select Switches are positioned to Primary (PRI)
- Safety Channel A middle detector has just failed low

Describe the status of the Startup Channels.

A. Only Startup Channel 1 high voltage is on

B. Only Startup Channel 2 high voltage is on

C. Both Startup Channels high voltage are on

D. Both Startup Channels high voltage is off

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Safety Channel A and B middle detector affects S/U channel #2. The input will come from Safety Channel A if the high voltage select switch is selected to PRI. Startup channel #1 is unaffected.
B. **CORRECT:** Safety Channel A and B middle detector affects S/U channel #2. The input will come from Safety Channel A if the high voltage select switch is selected to PRI.
C. Incorrect: Safety Channel A and B middle detector affects S/U channel #2. The input will come from Safety Channel A if the high voltage select switch is selected to PRI. Startup Channel #1 is unaffected.
D. Incorrect: Safety Channel A and B middle detector affects S/U channel #2. The input will come from Safety Channel A if the high voltage select switch is selected to PRI.

Technical Reference(s): OP-500-008 att. 4.84 revision 26
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-ENI00 obj. 2 (As available)

Question Source: Bank # 28-A
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 6
55.43 ______

Comments:

4.84 STARTUP CHANNEL 2 NEUTRON FLUX HI (K-4) REV 026

INITIATING DEVICE	SETPOINT
ENI-IJAC-0006	5 X 10^4 CPS

POSSIBLE EFFECTS

1. High source level during refueling operations.
2. High log counts rate during startup. Indicates Startup Channel High Volts has not turned Off at > 5.3X10^{-6}% power.

CONTROL ROOM INDICATIONS	LOCAL INDICATIONS
S27100, SU CH 1 NEUTRON FLUX LVL	NONE
S27101, SU CH 2 NEUTRON FLUX LVL	
ENIIJI0006(5), S/U Channel 2(1) Indication	
ENIIJR0006(5), S/U Channel 2(1) Recorder	

STARTUP CHANNEL 2 NEUTRON FLUX HI

POSSIBLE CAUSES

1. Either failure of log channel governing $5.3X10^{-6}$% relay or failure of $5.3X10^{-6}$% relay.

RECOMMENDED ACTIONS

1.1 If reactor power is > $5.3X10^{-6}$% then select Primary (Log Channel A) or Alternate (Log Channel B) as appropriate in Startup Channel 2 drawer to deenergize Startup Channel 2.

1.2 If reactor power < $5.3X10^{-6}$% then allow continued operation of Startup Channel 2.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	016 K5.01	
	Importance Rating	2.7	

K/A Statement

K5.01 - Knowledge of the operational implication of the following concepts as they apply to the NNIS: Separation of control and protection circuits

Proposed Question: RO 59 Rev: 0

Given:

- The in-service Pressurizer pressure control channel has just failed low

Which of the following prevents a simultaneous Reactor Protection System (RPS) Trip bistable actuation?

A. Separate circuits are used for **protection** channel and **control** channel functions.

B. Each circuit is provided with two separate outputs; one for **control** channel, one for **protection** channel.

C. Isolation amplifiers from the output of each circuit ensure that feedback from the **control** signal will not affect the **protection** channel.

D. Circuit outputs are multiplexed to be processed separately by the **control** channel and **protection** channel circuitry.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: Pressurizer pressure inputs are specifically separated for protection and control. Different transmitters are used for each function, as well as input for other indications

B. Incorrect: Protection channel instruments are required to be separate and redundant from controlling channels.

C. Incorrect: Isolation amplifiers are typically used in PPS circuitry to prevent a fault in a non-safety system from propagating back to PPS (e.g. Reactor Trip on Turbine Trip)

D. Incorrect: Multiplexing is used in PPS to carry many signals over a limited number of circuits

Technical Reference(s): SD-PAC
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPS00 obj. 1 (As available)

Question Source: Bank # 08264
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2008 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

SYSTEM PURPOSE

The purpose of the Process Analog Control (PAC) system is to measure various plant process parameters in the field, convert these measurements into electrical signals, condition them appropriately, and route them to various devices and components for indication, control, and protection purposes.

DESIGN BASES

The PAC system has both class-1E and non-class-1E components. The design bases for the PAC system are as follows:

- The system shall, with precision and reliability, automatically initiate appropriate protection action whenever a condition monitored by the system variable and preset level is achieved. The system must remain functional during and after a design basis event (DBE).
- Equipment which is not safety-related is designated as nonsafety-related and shall be physically separated and electrically isolated from class 1E, safety-related equipment.
- There is sufficient independence and redundancy to assure that (1) no single failure results in the loss of a protective function, (2) the removal from service of any component or channel does not result in loss of the minimum redundancy, and (3) the environment resulting from DBEs and natural phenomena do not result in the loss of the protective function.
- The system shall be capable of performing its functions during and after design basis operating and safe shutdown earthquakes (OBEs and SSEs), high winds, or tornados.
- The system shall be capable of performing its function in the event of a probable maximum flood or internal missiles.
- The system shall be designed and located to minimize the probability and effects of fires and explosions, consistent with other safety requirements.
- Any equipment that is used for both protective and control functions shall be classified as part of the protection system.
- All PAC cards are purchased class 1E.
- The transmission of signals from protection circuits to control circuits shall be through isolation devices that are classified as part of the protective system and shall be qualified such that credible failures at the output of the isolation devices cannot propagate to the protection circuits and prevent them from performing their protective functions.
- If the control and protection systems utilize common sensors such that a sensor failure could cause a control system action that requires protective action and

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017 K1.02	
	Importance Rating	3.3	

K/A Statement

K1.02 - Knowledge of the physical connections and/or cause-effect relationships between the ITM system and the following systems: RCS

Proposed Question: RO 60 Rev: 0

Core Exit Thermocouples are located ____(1)____ the fuel alignment plate and is the instrumentation used when verifying RCS temperature _____(2)_____ .

(1)	(2)

A. below — anytime less than four Reactor Coolant Pumps are running

B. below — during natural circulation conditions

C. above — during natural circulation conditions

D. above — anytime less than four Reactor Coolant Pumps are running

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: The CETs are located above the fuel alignment plate. The fuel alignment plate is at the top of the fuel. OI-038-000 states that CETs are used when in natural circulation conditions (no RCPs running). The operator is to use Th and Tc indications whenever there is forced flow.
B. Incorrect: The CETs are located above the fuel alignment plate. The fuel alignment plate is at the top of the fuel. OI-038-000 states that CETs are used when in natural circulation conditions (no RCPs running). The operator is to use Th and Tc indications whenever there is forced flow.
C. **CORRECT**: The CETs are located above the fuel alignment plate. The fuel alignment plate is at the top of the fuel. OI-038-000 states that CETs are used when in natural circulation conditions (no RCPs running). The operator is to use Th and Tc indications whenever there is forced flow.
D. Incorrect: The CETs are located above the fuel alignment plate. The fuel alignment plate is at the top of the fuel. OI-038-000 states that CETs are used when in natural circulation conditions (no RCPs running). The operator is to use Th and Tc indications whenever there is forced flow.

Technical Reference(s): OI-038-000 step 5.1 revision 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-INI00 obj. 1 (As available)
WLP-OPS-PPE01 obj. 4

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 2
55.43 ______

Comments:

5.0 PROCEDURE

5.1 GENERAL EXPECTATIONS

- Where multiple indications for one parameter exist, more than one instrument should be used to obtain a particular reading. Accidents may cause irregularities in particular instrument readings.
- Systems should not normally be placed in manual unless it is determined that manual operation will provide better control based on the plant status or misoperation in automatic is apparent. Systems placed in manual should be monitored to ensure proper operation.
- Safety Functions, as specified on the Safety Function Status Checklist, shall be continuously monitored throughout the use of this procedure.
- When responding to a request for a parameter, the operator should voice the parameter as well as the trend of the parameter.
- Verification of an RCS temperature response to a plant change during natural circulation cannot be accomplished until approximately 5 to 15 minutes following the action due to increased cycle times. CETs should be used during natural circulation conditions.
- Hot leg temperatures and Cold leg temperatures may be influenced by Safety Injection flow. Multiple indications and CET temperatures should be used to determine the Reactor Coolant System temperature.
- Parameter values that are bracketed should be used if a harsh environment in containment exists. A harsh environment is defined as Containment Temperature greater than or equal to 200°F.

FIG. 48
REACTOR VESSEL VERTICAL CROSS-SECTION

(REF. FSAR SECTION 5.4)

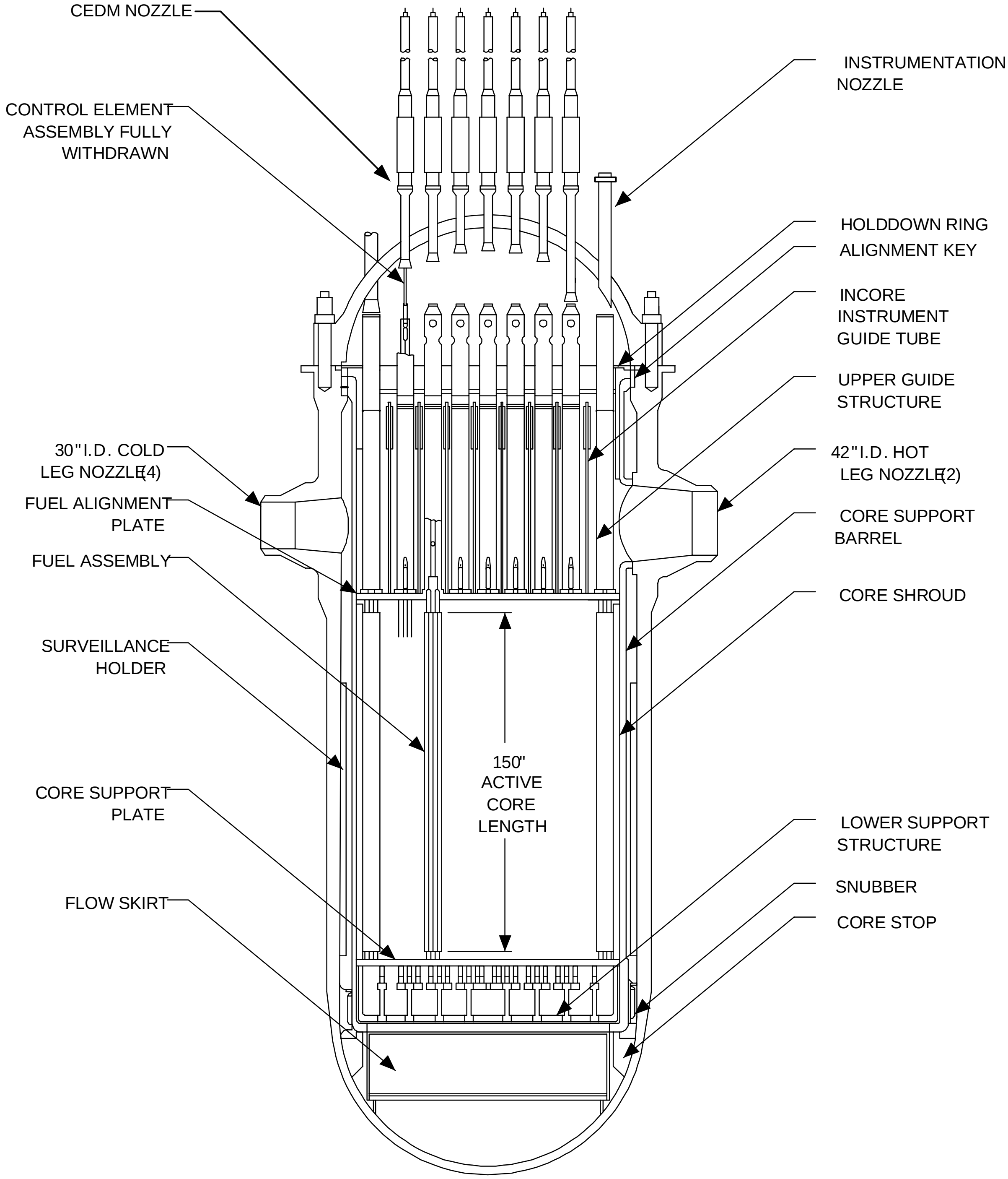
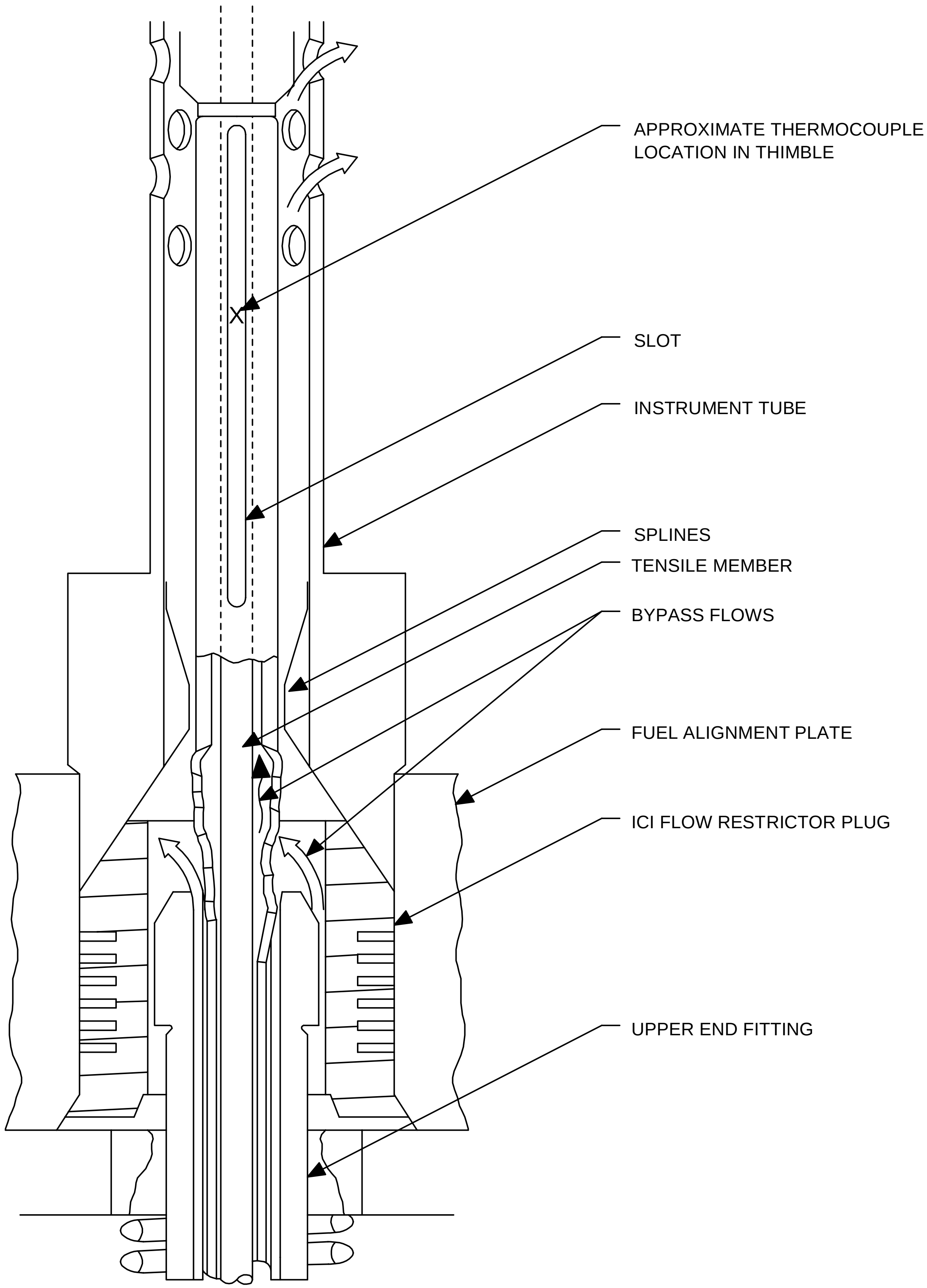


FIG. 55
CORE EXIT TEMPERATURE MEASUREMENT SCHEME

(REF. FSAR APPENDIX 1.9A)



6.0 NORMAL OPERATION

6.1 RECOMBINER STARTUP

6.1.1 Calculate required Hydrogen Recombiner Power Control potentiometer setting by performing following:

6.1.1.1 Record present Post-LOCA Containment pressure from Containment Atmosphere Pressure Indicator, ESF-IPI-6750A, on Attachment 11.2, Hydrogen Recombiner Power Control Setting Data Sheet.

6.1.1.2 Record Pre-LOCA Containment Average Temperature from OP-903-001, Technical Specification Surveillance Logs, on Attachment 11.2.

6.1.1.3 Determine Pressure Factor (Cp) from Attachment 11.4, Dry Containment Recombiner Power Correction Factor Graph.

6.1.1.3.1 Record Pressure Factor (Cp) on Attachment 11.2.

6.1.1.4 Determine Hydrogen Recombiner Power Control Setting by multiplying a reference power of 48 KW by Cp.

6.1.1.4.1 Record Hydrogen Recombiner Power Control Setting on Attachment 11.2.

6.1.2 Continuously monitor the Hydrogen Recombiner A(B) Power Meter, HRA-EM-960(962), when power level is being changed.

NOTE

Adjusting the Hydrogen Recombiner Power Control potentiometer slowly compensates for the lag between the meter and the potentiometer adjustments

6.1.3 Verify Hydrogen Recombiner A(B) Power Control potentiometer is set at zero (000).

6.1.4 Place Hydrogen Recombiner A(B) Power control switch, HRA-0001A(B), to ON.

6.1.5 Slowly adjust Hydrogen Recombiner Power Control potentiometer for Hydrogen Recombiner A(B) until 5 KW is indicated on Hydrogen Recombiner A(B) Power Meter, HRA-EM-960(962).

6.1.5.1 Hold reading for 10 minutes.

6.1.6 Verify Hydrogen Thermocouple Temperatures trend upward when adjusting Power Control Potentiometer, as indicated on Hydrogen Recombiner A(B) Outlet Air Temperature Indicator, HRA-ITI-0001A(B). Use Temperature Select switch to read thermocouple temperatures.

6.1.7 Adjust Hydrogen Recombiner Power Control potentiometer for Hydrogen Recombiner A(B) until 10 KW indicated on Hydrogen Recombiner A(B) Power Meter, HRA-EM-960(962).

6.1.7.1 Hold reading for 10 minutes.

6.1.8 Adjust Power Control Potentiometer for Hydrogen Recombiner(s) A(B) until 20 KW is indicated on Hydrogen Recombiner A(B) Power meter, HRA-EM-960(962).

6.1.8.1 Hold reading for 10 minutes.

CAUTION

DO NOT EXCEED 75 KW.

6.1.9 Adjust Hydrogen Recombiner Power Control Potentiometer for Hydrogen Recombiner A(B) to setting calculated on Attachment 11.2.

CAUTION

DO NOT EXCEED 1400°F.

6.1.10 Adjust Hydrogen Recombiner Power Control potentiometer as necessary, within the following guidelines, to maintain heater temperature > 1225°F to 1400°F, as read on Hydrogen Recombiner A(B) Outlet Air Temperature Indicator, HRA-ITI-0001A(B):

- Use the average of all three thermocouples temperatures to obtain a heater temperature. Example: 1200, 1210, and 1220, use 1210°F.
- If only two thermocouples are within 50°F of each other, then use average of the closest two temperatures. Examples: 1200, 1210, and 1270, use 1205°F.
- The following computer points can be used to trend operation of the Hydrogen Recombiner Operation:

Hydrogen Recombiner A	Hydrogen Recombiner B
A42700 – Temp 1	A42703 – Temp 1
A42701 – Temp 2	A42704 - Temp 2
A42702 - Temp 3	A42705 - Temp 3

6.1.11 Record Containment hydrogen concentration, Date, and Time on Attachment 11.3 when Hydrogen Recombiner Heater temperature reaches > 1225°F.

6.1.12 Verify proper Hydrogen Recombiner operation in accordance with Section 6.2, Verification of Recombiner Operation.

11.2 HYDROGEN RECOMBINER POWER CONTROL SETTING DATA SHEET

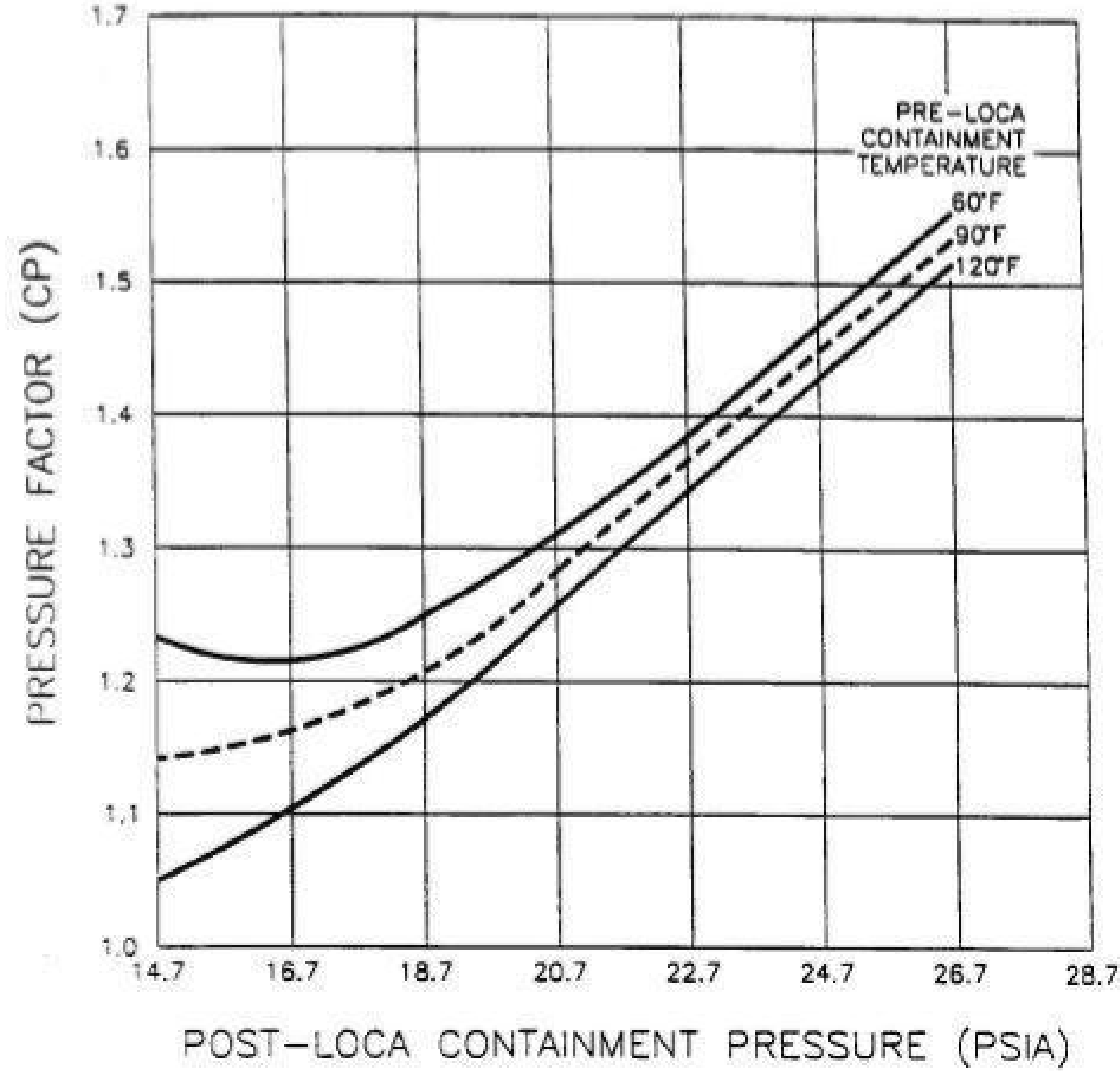
SECTION 6.1 (6.2)

STEP #		
6.1.1.1 (6.2.4.1)	Post-LOCA Containment Pressure, ESF-IPI-6750A (CP-8)	____________
6.1.1.2 (6.2.4.2)	Pre-LOCA Containment Temperature (from OP-903-001)	____________
6.1.1.3.1 (6.2.4.3.1)	______________ Cp (from Attachment 11.4)	
6.1.1.4.1 (6.2.4.4.1)	48 KW X Cp = Recombiner Power Control Setting	
	48 KW X __________ = ________________	

Performed by: ________________________________ ___________
(Signature) (Date)

IV by: ________________________________ ___________
(Signature) (Date)

11.4 DRY CONTAINMENT RECOMBINER POWER CORRECTION FACTOR GRAPH



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	028 A1.02	
	Importance Rating	3.4	

K/A Statement

A1.02 - Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Containment pressure

Proposed Question: RO 61 Rev: 0

Given:

- Plant tripped due to a Large Break LOCA event and TSC directed that both Hydrogen Recombiners were to be placed in service
- Hydrogen Recombiner A failed and was removed from service
- Hydrogen Recombiner B is operating with a setting of 25 KW output
- Hydrogen Analyzers indicate Containment H_2 concentration of 4.2% and rising slowly

The CRS directs the Hydrogen Recombiner B output raised in accordance with OP-008-006, using Attachment 11.2, to reduce Hydrogen concentration.

Post-LOCA Containment pressure is 17.7 PSIA and Pre-LOCA Containment Temperature was 105° F.

The final setting on the B potentiometer will be ____ KW to establish the required value.

A. 53

B. 55

C. 57

D. 59

Proposed Answer: B

Explanation: (Optional)

A. INCORRECT: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: 48 X 1.10 = 53 for (16.7 and 120)
B. **CORRECT**: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: **48 X 1.15 = 55** for (17.7 and 105)
C. INCORRECT: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: 48 X 1.19 = 57 for (18.0 and 90)
D. INCORRECT: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: 48 X 1.23 = 59 for (18.0 and 60)

Technical Reference(s): OP-008-006 revision 301
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: OP-008-006 Attachment 11.2 and 11.4

Learning Objective: WLP-OPS-HRA obj. 6 (As available)

Question Source: Bank # X Question #32
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

6.0 NORMAL OPERATION

6.1 RECOMBINER STARTUP

6.1.1 Calculate required Hydrogen Recombiner Power Control potentiometer setting by performing following:

6.1.1.1 Record present Post-LOCA Containment pressure from Containment Atmosphere Pressure Indicator, ESF-IPI-6750A, on Attachment 11.2, Hydrogen Recombiner Power Control Setting Data Sheet.

6.1.1.2 Record Pre-LOCA Containment Average Temperature from OP-903-001, Technical Specification Surveillance Logs, on Attachment 11.2.

6.1.1.3 Determine Pressure Factor (Cp) from Attachment 11.4, Dry Containment Recombiner Power Correction Factor Graph.

6.1.1.3.1 Record Pressure Factor (Cp) on Attachment 11.2.

6.1.1.4 Determine Hydrogen Recombiner Power Control Setting by multiplying a reference power of 48 KW by Cp.

6.1.1.4.1 Record Hydrogen Recombiner Power Control Setting on Attachment 11.2.

6.1.2 Continuously monitor the Hydrogen Recombiner A(B) Power Meter, HRA-EM-960(962), when power level is being changed.

NOTE

Adjusting the Hydrogen Recombiner Power Control potentiometer slowly compensates for the lag between the meter and the potentiometer adjustments

6.1.3 Verify Hydrogen Recombiner A(B) Power Control potentiometer is set at zero (000).

6.1.4 Place Hydrogen Recombiner A(B) Power control switch, HRA-0001A(B), to ON.

6.1.5 Slowly adjust Hydrogen Recombiner Power Control potentiometer for Hydrogen Recombiner A(B) until 5 KW is indicated on Hydrogen Recombiner A(B) Power Meter, HRA-EM-960(962).

6.1.5.1 Hold reading for 10 minutes.

6.1.6 Verify Hydrogen Thermocouple Temperatures trend upward when adjusting Power Control Potentiometer, as indicated on Hydrogen Recombiner A(B) Outlet Air Temperature Indicator, HRA-ITI-0001A(B). Use Temperature Select switch to read thermocouple temperatures.

6.1.7 Adjust Hydrogen Recombiner Power Control potentiometer for Hydrogen Recombiner A(B) until 10 KW indicated on Hydrogen Recombiner A(B) Power Meter, HRA-EM-960(962).

6.1.7.1 Hold reading for 10 minutes.

6.1.8 Adjust Power Control Potentiometer for Hydrogen Recombiner(s) A(B) until 20 KW is indicated on Hydrogen Recombiner A(B) Power meter, HRA-EM-960(962).

6.1.8.1 Hold reading for 10 minutes.

CAUTION
DO NOT EXCEED 75 KW.

6.1.9 Adjust Hydrogen Recombiner Power Control Potentiometer for Hydrogen Recombiner A(B) to setting calculated on Attachment 11.2.

CAUTION
DO NOT EXCEED 1400°F.

6.1.10 Adjust Hydrogen Recombiner Power Control potentiometer as necessary, within the following guidelines, to maintain heater temperature > 1225°F to 1400°F, as read on Hydrogen Recombiner A(B) Outlet Air Temperature Indicator, HRA-ITI-0001A(B):

- Use the average of all three thermocouples temperatures to obtain a heater temperature. Example: 1200, 1210, and 1220, use 1210°F.
- If only two thermocouples are within 50°F of each other, then use average of the closest two temperatures. Examples: 1200, 1210, and 1270, use 1205°F.
- The following computer points can be used to trend operation of the Hydrogen Recombiner Operation:

Hydrogen Recombiner A	Hydrogen Recombiner B
A42700 – Temp 1	A42703 – Temp 1
A42701 – Temp 2	A42704 - Temp 2
A42702 - Temp 3	A42705 - Temp 3

6.1.11 Record Containment hydrogen concentration, Date, and Time on Attachment 11.3 when Hydrogen Recombiner Heater temperature reaches > 1225°F.

6.1.12 Verify proper Hydrogen Recombiner operation in accordance with Section 6.2, Verification of Recombiner Operation.

11.2 HYDROGEN RECOMBINER POWER CONTROL SETTING DATA SHEET

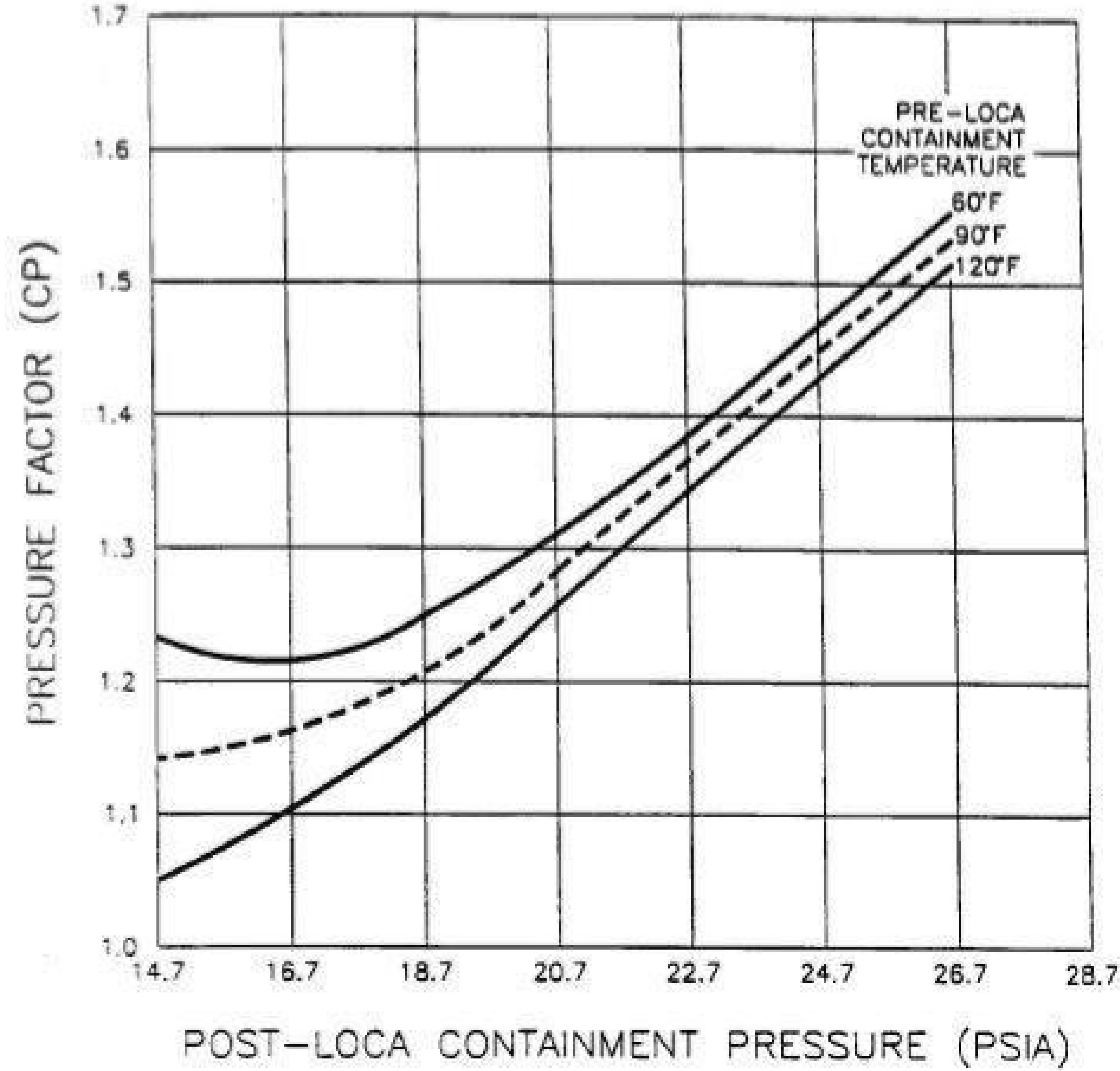
SECTION 6.1 (6.2)

STEP #		
6.1.1.1 (6.2.4.1)	Post-LOCA Containment Pressure, ESF-IPI-6750A (CP-8)	____________
6.1.1.2 (6.2.4.2)	Pre-LOCA Containment Temperature (from OP-903-001)	____________
6.1.1.3.1 (6.2.4.3.1)	______________ Cp (from Attachment 11.4)	
6.1.1.4.1 (6.2.4.4.1)	48 KW X Cp = Recombiner Power Control Setting	
	48 KW X __________ = ________________	

Performed by: ________________________________ ____________
(Signature) (Date)

IV by: ________________________________ ____________
(Signature) (Date)

11.4 DRY CONTAINMENT RECOMBINER POWER CORRECTION FACTOR GRAPH



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	029 K3.01	
	Importance Rating	2.9	

K/A Statement

K3.01 - Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following: Containment parameters

Proposed Question: RO 62 Rev: 0

Given:

- Plant is at 100% power
- Containment Purge is in progress in accordance with OP-002-010, Reactor Auxiliary Building HVAC and Containment Purge
- CAP-203, Containment Purge Exhaust Inside Containment, has failed closed

In relation to the Containment Purge Malfunction and three minutes into the event, Containment Pressure will ______(1)______ because ______(2)______ .

	(1)	(2)
A.	remain the same	containment purge inlet dampers close when CAP-203 closes
B.	remain the same	RAB normal Exhaust fan trips on low flow
C.	continue to rise	containment purge outlet dampers are closed
D.	continue to rise	RAB normal Exhaust fan trips on low flow

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** An interlock is associated with the Containment Purge Inlet dampers such that they will close 120 seconds into the event if CAP-203 is not full open. Therefore with these conditions, both inlet and exhaust valves isolate and containment pressure is unaffected.
B. Incorrect: Part 1 is correct. The Containment Purge system discharges to the RAB Normal Exhaust fans and the RAB normal exhaust fans do have a low flow trip. But losing containment purge flow does not lower flow enough to trip a RAB normal Exhaust fan. Pressure remaining the same in containment is plausible because the inlet CAP valves would close if RAB exhaust flow were low.
C. Incorrect: An interlock is associated with the Containment Purge Inlet dampers such that they will close if CAP-203 is not full open. Containment pressure is unaffected and the second portion is plausible if the applicant is not aware of the interlock that the makeup valves have with CAP-203.
D. Incorrect: An interlock is associated with the Containment Purge Inlet dampers such that they will close if CAP-203 is not full open. Therefore with these conditions, both inlet and exhaust valves isolate and containment pressure is unaffected. Containment pressure rising here is plausible if the applicant determines that we are not longer exhausting but are still supplying makeup air.

Technical Reference(s): OP-002-010 revision 306
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-HVR00 obj. 4 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

Fans

The fans are, direct-drive centrifugal type, sized to handle sufficient air volume necessary to maintain specific room air temperatures. Each fan is driven by a horizontal electric motor connected to a 480-volt, 60-cycle, 3-phase circuit and is interlocked with safety related pumps when the fan cooler provides cooling to a safety related pump room. Others are operated as required by a room temperature sensing element.

All of these fan coolers have a local control switch for starting or stopping (local control switch "STOP" only works when the "START" switch has been used to start the fan).

CONTAINMENT ATMOSPHERIC PURGE DAMPERS AND VALVES

Makeup Air Inlet Damper (CAP-101)

The air inlet makeup damper allows air into the makeup portion of the CAP system. CAP-101 is an air operated damper that fails closed on a loss of air.

The damper will open if the RAB Ventilation Mode Selector Switch on CP-18 is in CNTMT PURGE or REFUEL VENT and both of the following are true:

- RAB normal exhaust flow is greater than 69,000 scfm
- Containment pressure is less negative than -8.4 inches of water gauge (in. WG) below atmospheric

If the conditions previously mentioned are satisfied, CAP-101 will remain open for at least 120 seconds. The valve will close if any of the following conditions exists after 120 seconds:

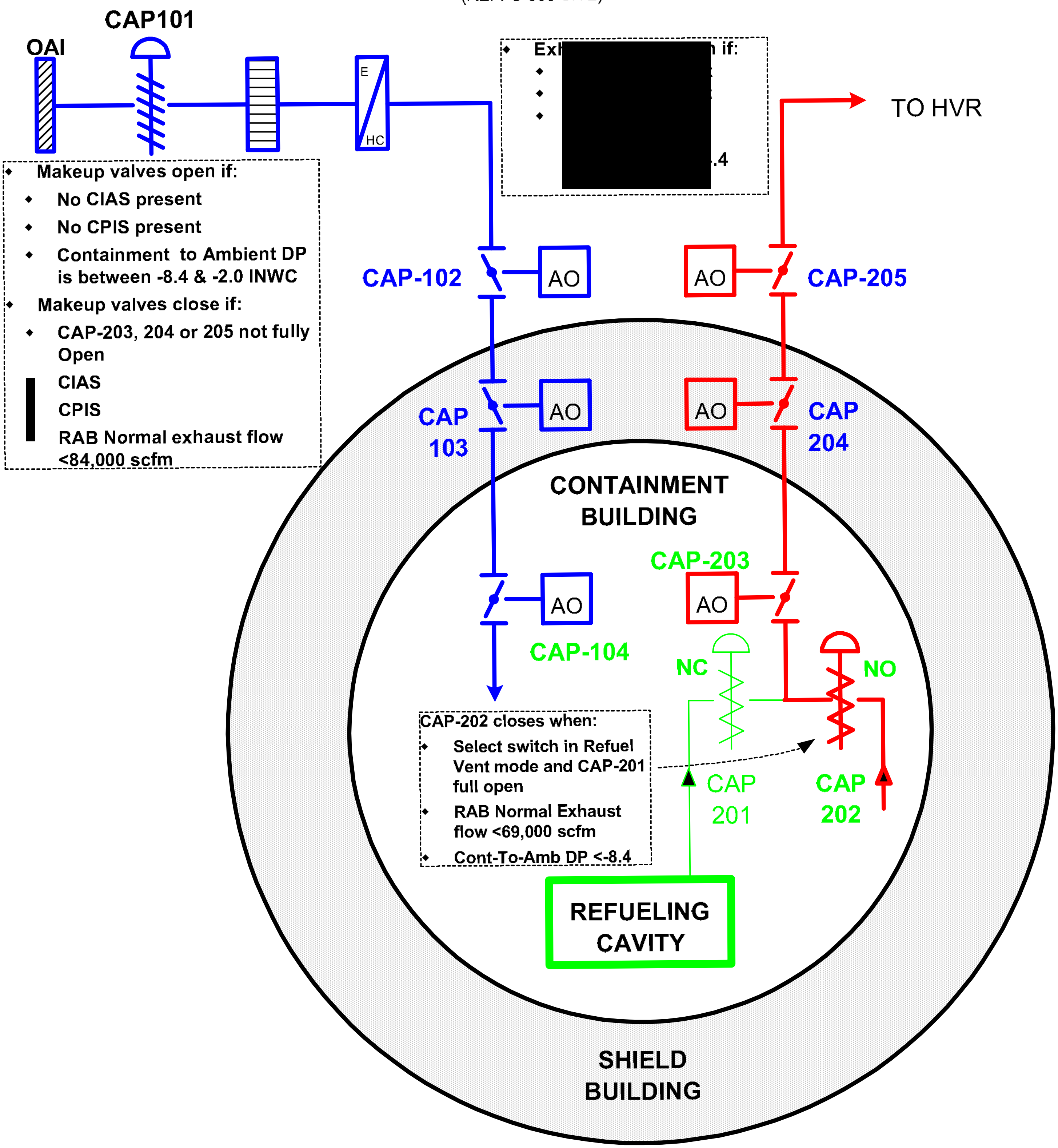
- CAP-203, -204, or -205 are not fully open
- CIAS is present
- HI HI alarm on plant stack process or containment atmospheric radiation monitor
- RAB normal exhaust flow drops below 84,000 scfm

Makeup Air Butterfly Valves (CAP-102, -103, and -104)

Valves CAP-103 and -104 serve as containment isolation valves and are located on the outboard and inboard sides of containment respectively. CAP-103 and -104 share a

FIG. 01C CONTAINMENT ATMOSPHERIC PURGE

(REF. G-853 SH 2)



All makeup and exhaust dampers and valves will close to isolate the purge path in the event of any of the following:

- Containment Isolation Actuation Signal (CIAS) or a
- Containment Purge Isolation Signal (CPIS) generated by on of the following:
  - HI-HI alarm on either Plant Stack process radiation monitor
  - Hi-Hi alarm on any of the four containment purge radiation monitors

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	071 A2.02	
	Importance Rating	3.3	

K/A Statement

A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Use of waste gas release monitors, radiation, gas flow rate, and totalizer

Proposed Question: RO 63 Rev: 0

Given:

- Plant is at 100% power
- A Gaseous Waste Discharge is in progress when the Waste Gas Flow and Radiation Recorder, GWM-IFRR-0648, fails
- The recorder cannot be immediately repaired

Which of the following describes the impact of the failure, and the action required?

A. Waste Gas Discharge automatic isolation is inoperable; stop the release until the recorder is returned to operable status. The release may resume ONLY when a new release permit is issued.

B. Waste Gas Discharge automatic isolation is inoperable; stop the release until the recorder is returned to operable status. The release may resume under the existing release permit after verifying NO new additions to the WGD Tank(s) being released.

C. Waste Gas Discharge automatic isolation remains operable but flow indication is lost; verify the flow setpoint on GWM-309 remains below the limit specified on the release permit, and continue the release.

D. Waste Gas Discharge automatic isolation remains operable but flow indication is lost; estimate discharge flow rate by determining the change in GDT pressure over time to ensure flow rate remains within limits.

Proposed Answer: D

Explanation: (Optional)

A. Incorrect. Loss of the flow recorder does not cause loss of isolation capability. It does cause loss of indication, but the flow recorder is not input to the valve. It is taken directly from the monitor

B. Incorrect. Actions are for unexpected change in radiation levels during a release and isolation capability is not lost.

C. Incorrect. Response is correct but the action is incorrect. Verifying the flow setting does not necessarily indicate flow, as tank backpressure will change.

D. **CORRECT:** The waste gas discharge automatic isolation remains operable but the flow indication is lost. The action is to estimate discharge flow rate by determining the change in GDT pressure over time to ensure flow rate remains within limits. A reference is given but the applicant must know (outside of the procedure) that the flow recorder does not have an effect on automatic isolation.

Technical Reference(s): OP-007-003, Page 15, Step 6.4.14 revision 304
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-RMS00 obj. 4 (As available)

Question Source: Bank # 08269
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 11
55.43

Comments:

NOTE

Successful performance of step 6.4.12 satisfies TRM Table 4.3-9 Channel Check.
[P-2390, P-2414]

6.4.12 Verify indication of GWM discharge flow using any of the following indications:

- GWM-IFRR-0648 Waste Gas Flow & Rad Recorder (CP-4)
- GWM-IFIT-0648 Waste Gas Flow Indic Transmitter (local)
- GWM-IFI-6712 Waste Gas Decay Tanks to Plant Vent Flow Indic (LCP-42A)
- PMC PID A41300 Gas to Stack Flow
- GWM-IFQI-0648 Waste Gas Flow (CP-4) counting upward

6.4.12.1 On the Gaseous Release Permit, initial for satisfactory performance of the Channel Check of GWM-IFIT-0648.

6.4.13 Verify discharge flow and activity are within the limits specified on the Gaseous Release Permit.

6.4.13.1 Record data at the required intervals on the Gaseous Release Permit.

6.4.14 If the Waste Gas Flow and Radiation Recorder, GWM-IFRR-0648, is out of service, then estimate the Waste Gas discharge flow every four hours by performing the following:

6.4.14.1 Multiply Gas Decay Tank (GDT) change in pressure by the conversion factor, 40.97 scf/psig.

6.4.14.2 Divide result in step 6.4.14.1 by the change in time.

6.4.14.3 Record results in step 6.4.14.2 on Technical Specification Addendum Logsheet.

6.4.14.4 Verification of results required on Technical Specification Addendum Logsheet.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	075 K2.03	
	Importance Rating	2.6	

K/A Statement

K2.03 - Knowledge of bus power supplies to the following: Emergency/essential SWS pumps

Proposed Question: RO 64 Rev: 0

Auxiliary Component Cooling Water (ACCW) Pump B is powered from which bus?

A. SWGR 31B

B. SWGR 3B

C. SWGR 1B

D. SWGR 2B

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: ACCW Pump B is powered from the 3B bus. The 31B is a 480 V safety bus fed from the 3B bus.
B. **CORRECT:** ACCW Pump B is powered from the 3B bus.
C. Incorrect: ACCW Pump B is powered from the 3B bus. The 1B bus is a 6.9 Kv which carries large loads.
D. Incorrect: ACCW Pump B is powered from the 3B bus. The 2B bus is a 4.16 Kv non-safety bus.

Technical Reference(s): OP-002-001 Attachment 11.2 Revision 305
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CC00 obj. 4 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 4
55.43

Comments:

AUXILIARY COMPONENT COOLING WATER SYSTEM STANDBY BREAKER LINEUP (CONT'D)

COMPONENT NUMBER	COMPONENT DESCRIPTION	LOCATION	REQUIRED POSITION	PERFORMED BY (INITIAL/DATE)	IV BY (INITIAL/DATE)
ACC-EBKR-3B-6A	AUX COMPONENT COOLING WATER PUMP B MOTOR HEATER	RAB +21 COL 10 A&K	ON		
ACC-EBKR-3B-6	AUXILIARY COMPONENT COOLING WATER PUMP B	RAB +21 COL 10 A&K	RACKED IN		
ACC-EBKR-3B-6	CONTROL POWER KNIFE SWITCH [redacted]	RAB +21 COL 9 A&G	OPEN(1)		
ACC-EBKR-313A-10F	ACC JOCKEY PUMP A	RAB +21 COL 10A&H	ON		
ACC-EBKR-313B-10C	ACC JOCKEY PUMP B	RAB +21 COL 10A&K	ON		
ACC-EBKR-311A-10M	ACC PUMP A DISCHARGE LINE ISOLATION (ACC-110A)	RAB +21 COL 10A&H	ON		
ACC-EBKR-311B-12J	ACC PUMP B DISCHARGE LINE ISOLATION (ACC-110B)	RAB +21 COL 10A&K	ON		

(1) Closed after System Fill and Vent.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	086 A4.02	
Importance Rating	3.5	

K/A Statement

A4.02 - Ability to manually operate and/or monitor in the control room: Fire Detection Panels

Proposed Question: RO 65 Rev: 0

Per FP-001-020, Fire Emergency/Fire Report, if a fire detector located in the Control Room or ____(1)____ room alarms, the control room will direct the TGB Switchgear Operator to the TGB Switchgear and the Field Operator to the ____(2)____.

	(1)	(2)
A.	cable vault	+35 Relay Room
B.	cable vault	Control Room
C.	Isolation Panel	+35 Relay Room
D.	Isolation Panel	Control Room

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** FP-001-020 step 6.4.2 states that in the event of a fire alarm in the Control Room or the Cable vault rooms, the field operator will be directed to the +35 Relay Room.

B. Incorrect: FP-001-020 step 6.4.2 states that in the event of a fire alarm in the Control Room or the Cable vault rooms, the field operator will be directed to the +35 Relay Room. The Control Room is plausible, because the emergency communicator is directed to the Control Room.

C. Incorrect: FP-001-020 step 6.4.2 states that in the event of a fire alarm in the Control Room or the Cable vault rooms, the field operator will be directed to the +35 Relay Room. The isolation panel fire actions are located in a separate off-normal but do not require dispatching NAOs.

D. Incorrect: FP-001-020 step 6.4.2 states that in the event of a fire alarm in the Control Room or the Cable vault rooms, the field operator will be directed to the +35 Relay Room. The Control Room is plausible, because the emergency communicator is directed to the Control Room. The isolation panel fire actions are located in a separate off-normal but do not require dispatching NAOs.

Technical Reference(s): FP-001-020 revision 307 step 6.4.2
(Attach if not previously provided) OP-901-502 revision 27 steps 7.2-7.4
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO51 obj. 22 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

6.4 ACTIONS BY NUCLEAR PLANT OPERATORS (REACTOR OPERATORS) OR TSC LEAD COMMUNICATOR

NOTE

(1) A "FIRE" should be declared even without the actual observance of flames should the smoke or heat be of such degree that additional assistance may be necessary or the use of protective gear or SCBA equipment is considered.

(2) The Emergency Communicator is directed to report to the Control Room upon report of a fire or if an unplanned fire alarm occurs (in a plant equipment location) to support communications for Fire Brigade activation and notification of Hahnville Fire Department if needed as directed by the SM/CRS. If an Emergency Classification is declared, then the Emergency Communicator will perform E-Plan duties as directed by the Emergency Coordinator.

6.4.1 If an unplanned (other than maintenance or known equipment deficiency) fire alarm occurs or a report of a fire, then perform the following:

- Direct the Fire Brigade Leader to investigate (or a Fire Brigade Member if the Leader is not readily available).
- Direct the Emergency Communicator to report to the Control Room (only required for fire alarms associated with locations containing plant equipment or upon a fire report).

6.4.2 If a fire detector located in the Control Room (Rooms 304 or 305, Fire Zone RAB-1A) or Cable Vault (Room 260, Fire Zone RAB-1E) alarms (indicating a potential for a fire), then perform the following: **[P-26635]**

- Direct the Field Operator to report to the +35 Relay Room.
- Direct the Switchgear Operator to report to the TGB Switchgear.
- Verify the Control Room Staff (SM, CRS, STA, ATC, & BOP) is located within the Control Room.

6.4.3 If following investigation the location is determined to not contain a fire, then individuals may return to normal duties.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.4	
	Importance Rating	3.3	

K/A Statement

Conduct of Operations: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.
.
Proposed Question: RO 66 Rev: 0

A licensee has NOT stood the required number of proficiency watches prior to the end of the calendar quarter. Which one of the following correctly describes the 10CFR55 provision for the licensee to perform licensed duties during the next quarter?

A. Active status may be continued provided the remaining proficiency watches are stood during the first month of the following calendar quarter.

B. Active status may be continued provided 12 hours of under instruction watches are completed within the first month of the following calendar quarter.

C. The license must become inactive. 40 hours of under instruction watches shall be stood to return the license to active status.

D. The license must become inactive. Seven 8-hour or five 12-hour under instruction watches shall be stood to return the license to active status.

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: The license becomes inactive if proficiency watches are not stood. OI-024-000 states that the 40 hours under instruction should be completed within the first 30 days of the next quarter.
B. Incorrect: 12 hours of under instruction is the required number of hours need for an SRO to upgrade to active fuel handling supervisor status.
C. **CORRECT:** failure to complete the required number of proficiency watches will result in inactivation of the license, which will require 40 hours under instruction to reactivate.
D. Incorrect: 40 hours of under instruction must be performed to return an inactive license to active status.

Technical Reference(s): OI-024-000 Revision 307
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLO-OPS-TS03 obj. 4 (As available)

Question Source: Bank # 07944
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2006 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Minor editorial changes.

5.2 UPGRADING FROM INACTIVE TO ACTIVE STATUS

NOTE

Only one trainee can be assigned to a qualified watchstander for the purpose of completing an under instruction watch.

5.2.1 RO and SRO licensees must stand a minimum of 40 hours as a trainee on shift. A complete plant tour and all required shift turnover requirements must be completed during this 40 hours. The 40 hours should be started and completed within a 30 day period. **[P-13838]**

5.2.2 RO licensees must stand parallel watches as ATC or BOP trainee.

5.2.3 SRO licensee trainees must stand watches in the position to which the individual will be assigned. (i.e. Shift Manager to stand as SM and Control Room Supervisor to stand as CRS.) **[P-13838]**

5.2.4 For SRO licensees upgrading to Active SRO Status as Fuel Handling Supervisor only, a minimum of 12 hours as Fuel Handling Supervisor trainee while conducting core alterations or fuel movement is required. **[P-13838]**

5.2.5 Assistant Operations Manager (Shift) authorization must be obtained prior to standing trainee watches. This is documented on the “Authorization to Activate” line of Attachment 6.1

5.2.6 All conditions of Attachment 6.4 and all administrative requirements of Attachment 6.5 must be met for an individual to resume Active Status in a Reactor Operator or Senior Reactor Operator position.
[CR-WF3-1997-00831]

5.2.7 Use Attachment 6.1, Checklist for Upgrading from Inactive to Active Status, and Attachment 6.2, Plant Tour Checklist to record the completion of necessary trainee requirements.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.8	
	Importance Rating	3.4	

K/A Statement

Conduct of Operations: Ability to coordinate personnel activities outside the control room.

Proposed Question: RO 67 Rev: 0

Per OI-042-000, Watch Station Process, coordinating activities outside of the Control Room is listed as a responsibility for the ____(1)____ and the ____(2)____ .

	(1)	(2)
A.	Shift Manager	extra Nuclear Plant operator
B.	Shift Manager	At the Controls (ATC) operator
C.	Field Support Supervisor	extra Nuclear Plant operator
D.	Field Support Supervisor	At the Controls (ATC) operator

Proposed Answer: C.

Explanation: (Optional)

A. Incorrect: OI-042-000, Watch Station Process, section 5.1 lists the responsibilities of control room personnel. Coordination of activities outside of the control room is listed as responsibility of the FSS and extra NPO, not the SM.
B. Incorrect: OI-042-000, Watch Station Process, section 5.1 lists the responsibilities of control room personnel. Coordination of activities outside of the control room is listed as responsibility of the FSS and extra NPO, not the SM or the ATC.
C. **CORRECT:** OI-042-000, Watch Station Process, section 5.1 lists the responsibilities of control room personnel. Coordination of activities outside of the control room is listed as responsibility of the FSS and extra NPO.
D. Incorrect: OI-042-000, Watch Station Process, section 5.1 lists the responsibilities of control room personnel. Coordination of activities outside of the control room is listed as responsibility of the FSS and extra NPO, not the ATC.

Technical Reference(s): OI-042-000 revision 31
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPA00 obj. 3 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

5.1.2 PERSONNEL RESPONSIBILITIES

5.1.2.1 Control Room Staff

a. In order to ensure Licensed Operators on duty can perform actions associated with Security Events and Evacuation of a Control Room procedures, Control Room Staff assigned to support the minimum shift composition (SM, CRS, STA, ATC, and BOP) shall not leave the Protected Area unless officially relieved of their duties. In addition, these Licensed Operators (SM, CRS, STA, ATC, and BOP) shall obtain their designated Security Key Ring, a Flashlight, and an Operations Radio prior to leaving the Control Room Envelope. **[CR-WF3-2012-03815]**

5.1.2.2 Shift Manager

a. Communicate with the Woodlands Operator for the following:
   - Unit loading instructions
   - Unit operational problems
   - Inability to respond to demands or projected maneuvering schedules

b. In the event of receiving a 251 pager code, the Training Shift Manager performs the following:
   - Direct the Training Shift fire brigade members to report to Fire Locker 4 in the In-Processing Building.
   - Direct all other Training Shift members to report to the Skills Training Center.

5.1.2.3 Control Room Supervisor

a. Verify that complete and thorough shift turnovers take place.

5.1.2.4 Shift Manager or Control Room Supervisor

a. Activate the Fire Brigade/Hazardous Materials Response Team as necessary.

b. Authorize Radioactive Waste Discharge Permits.

c. Generate and approve Maintenance and Radiation Protection control documents.

d. Relieve or direct the relief of shift operating personnel responsibilities if deemed necessary due to psychological, emotional, or physical impairments.

e. Take action as required to respond to a Security Alert or an emergency situation and provide necessary support or assistance to Security Force personnel.

5.1.2.5 Field Support Supervisor/Shift Technical Advisor

a. Reports to the Shift Manager.

b. Peer to the Control Room Supervisor, yet reports to the CRS all field activities as the CRS remains the single point of contact for plant conditions. The SM/CRS will maintain responsibility for signing permission on surveillances.

c. Supervise and coordinate operational activities outside Control Room, on the appointed shift through all conditions of startup, power generation, shutdown.

d. Inform the Shift Manager or Control Room Supervisor of any abnormal conditions or changes to plant status and ensure initiation of procedurally required steps to initiate corrective actions.

e. Assist the Shift Manager in completing performance evaluations of personnel assigned at least annually.

f. Recommend to the Shift Manager changes in plant procedures, personnel, operation and equipment status such that overall plant efficiency and safety is improved.

g. Direct, as necessary, Operations Department personnel in their assignment to maintain plant safety and performance.

h. Coordinate, authorize/release, and brief tagouts, and other crew field activities.

i. Perform operability assessments Friday-Sunday, and other times as needed.

j. Perform PP&R and Times sheet duties for NAOs, as coordinated with the CRS & SM.

k. See EN-OP-115, Conduct of Operations, for additional responsibilities.

l. See UNT-7-011, Duties and Responsibilities of the Shift Technical Advisor, for additional responsibilities.

5.1.2.6 Nuclear Plant Operator

a. Maintain communications with Nuclear Auxiliary Operators and other station personnel performing functions which affect Control Room indications or controls.

b. Evaluate and acknowledge all computer and annunciator alarms received.

c. Perform, participate in, and/or coordinate surveillance <u>and</u> special testing activities in accordance with approved procedures.

d. Add initials, time, and date on the recorders on each night shift.

5.1.2.6.1 Reactor Operator “At-the-Controls” (ATC) Expectations

- Normally takes direction from the Control Room Supervisor
- Remain in the ATC area of the Control Room as identified in EN-OP-115-01, Control Room Conduct and Access Control.
- Maintains primary responsibility for monitoring control board indications making use of diverse and redundant instrumentation, for abnormalities in system lineups and operating parameters.
- Should avoid becoming involved in activities which may distract the ATC from primary responsibility.
- Maintain an active Senior Reactor Operator or Reactor Operator License.

5.1.2.6.1.1 Internet access is not permitted in the ATC area of the Control Room, except as specifically authorized by the Operations Manager.

5.1.2.6.1.2 The ATC Operator has the primary responsibility for performing Control Room Panel walk-downs in the "At the Controls” area for any abnormal condition or undesirable trend. This includes taking logs that do not require the ATC to leave the Surveillance Area.

5.1.2.6.1.3 Fire Detection panel and Protective relay panel CP-15 area are allowed areas for the ATC.

5.1.2.6.1.4 The ATC should not become involved in any activity that hampers their ability to monitor the plant.

5.1.2.6.1.5 The ATC is permitted to perform peer checks for the BOP and assist with surveillance activities.

5.1.2.6.1.6 The ATC will be relieved by another Licensed Operator during activities that may distract the ATC from the primary duty of monitoring the reactor:

- Leading reactivity or other briefs,
- Leaving the Surveillance Area for any reason.
- Performing detailed administrative duties

5.1.2.6.1.7 The ATC should spend most of his/her time in the area around CP-2 and CP-4 monitoring parameters necessary to support safe reactor operation.

5.1.2.6.1.8 The use of diverse plant indications are available and should be utilized especially during activities that may have the ATC away from the normal monitoring station at CP-2. Board indicators associated with primary plant power, temperature, pressure, etc. are available at CP-7 and CP-8 also, computer screens can be set up to aid in plant monitoring in these situations.

5.1.2.6.1.9 The ATC and BOP may be on the phone at the same time, but this should be minimized and oversight by one of the control room SROs will be maintained.

5.1.2.6.1.10 The ATC may have his back to the control room panels at various times as needed to perform other activities required of a licensed operator.

5.1.2.6.1.11 The ATC may sit at the NPO desk as required to utilize the computer for procedure activities, log documentation and other needed work. Monitoring of power and trends on can be done from this location and will continue to be the ATC responsibility.

Examples of allowed activities for the ATC (These are not all inclusive; questions should be addressed by the on-shift Shift Manger):

- The ATC may acknowledge the control room main fire panel alarms
- The ATC may change out chart papers
- The ATC can use the telephone to discuss plant information
- The ATC can get a book such as Tech Spec’s or cumulative tracking log if needed for reference.

5.1.2.6.2 Extra Nuclear Plant Operator (RO/SRO License)

- Function as a Non-Licensed operator as necessary.
- Take direction for control manipulations from the person having Control Room Command and Control.
- Coordinate activities of shift non-licensed personnel.
- Monitor and coordinate activities outside of the Control Room, as required.
- Maintain an active Senior Reactor Operator or Reactor Operator License.
- Conduct On the Job Training (OJT) and Task Performance Evaluation (TPE).
- Maintain communications with Nuclear Auxiliary Operators and other station personnel performing functions which affect Control Room indications or controls.
- Evaluate and acknowledge all computer and annunciator alarms received.
- Perform, participate in, and/or coordinate surveillance and special testing activities in accordance with approved procedures.
- Add initials, time, and date on the recorders on each night shift.
- Order cessation of any activity within the station which, if continued, will have an immediate and adverse impact upon unit operation or safety.
- Order cessation of any maintenance activity where the potential exists for inadvertently positioning components. Compensatory measures should be put in place prior to allowing the applicable activities to resume.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.44	
	Importance Rating	3.9	

K/A Statement

Conduct of Operations: Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Proposed Question: RO 68 Rev: 0

Given:

- Plant is in a refueling outage and in DEFUEL mode

The crew will declare the plant to be in Mode 6 when informed by the Refuel SRO that the first fuel bundle is located in the upender on the _____(1)_____ side and _____(2)_____.

	(1)	(2)
A.	fuel handling building	is capable of being transported to the containment side
B.	fuel handling building	the transfer carriage has been sent to the containment side
C.	containment building	the containment upender is vertical
D.	containment building	is grappled by the refueling machine

.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: OP-0100-014 states that Mode 6 conditions are reestablished when the first fuel bundle is placed in the upender on the Fuel Handling Building side and is capable of being transported to the Containment side.
B. Incorrect. OP-0100-014 states that Mode 6 conditions are reestablished when the first fuel bundle is placed in the upender on the Fuel Handling Building side and is capable of being transported to the Containment side.
C. Incorrect. OP-0100-014 states that Mode 6 conditions are reestablished when the first fuel bundle is placed in the upender on the Fuel Handling Building side and is capable of being transported to the Containment side.
D. Incorrect. OP-0100-014 states that Mode 6 conditions are reestablished when the first fuel bundle is placed in the upender on the Fuel Handling Building side and is capable of being transported to the Containment side.

Technical Reference(s): OP-100-014 Rev. 323 page 65
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-REQ04 obj. 4 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7,11
55.43

Comments:

Core Offload

Component/Condition	Affected Systems and Tech Specs	Required Action(s)
Core Off-Load (Administrative Requirements)	• 3.3.3.1 Radiation Monitoring: ♦ CCW Monitors A & B ♦ Containment Purge & Exhaust Isol • 3.3.3.7.1 Chlorine Detection • 3.3.3.7.3 Broad Range Gas Detection • 3.3.3.11 Explosive Gas Monitoring (when system is in service) • 3.4.9 (TRM) Structural Integrity of ASME 1, 2, & 3 components • 3.7.8 Snubbers (for systems required Operable in Modes 5 & 6) • 3.9.1 Boron Concentration (Mode 6) • 3.9.4 Containment Penetrations • 6.2.2 Unit Staff (Modes 5 & 6)	• Core Off-Load is established when the last fuel bundle is removed from the upender on the Fuel Handling Building side. Mode 6 conditions are reestablished when the first fuel bundle is placed in the upender on the Fuel Handling Building side and is capable of being transported to the Containment side. • Mode 6 Logs must be current prior to placing the first fuel bundle in the upender on the Fuel Handling Building side. • During Core Off-Load certain Technical Specifications will be applicable. If an Administrative Requirement is not maintained during Core Off-Load then a Condition Report should be generated in accordance with LI-102, Corrective Action Process. For Technical Specifications that are being maintained during Core Off-Load the applicable Action Statement must be followed. • During Core Off-Load conditions the term Operable may be replaced by Functional for the Administrative Requirements of Tech Specs contained in this section.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.1	
	Importance Rating	4.5	

K/A Statement

Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: RO 69 Rev: 0

Given:

- The crew is performing actions in OP-010-003, Plant Startup
- The CRS has directed the ATC to perform a VCT Makeup using the Dilute Makeup Mode per OP-002-005, Chemical and Volume Control

Per OI-042-000, Watch Station Process, a ____(1)____ should be performed prior to and immediately following the PMU addition and the evolution should be cross-checked by ______(2)______ .

	(1)	(2)
A.	shift update	verifying the pre-power dependent insertion limit annunciator is clear
B.	shift update	comparing what was placed in the batch counter to the totalizer reading
C.	reactivity brief	comparing what was placed in the batch counter to the totalizer reading
D.	reactivity brief	verifying the pre-power dependent insertion limit annunciator is clear

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Per OI-042-000, section 5.3.2, a shift update should be performed prior to and immediately following any reactivity manipulation and a cross-check of the reactivity manipulation is performed by comparing what was placed in the batch counter to the totalizer reading. The pre-power dependent insertion limit would come in if power rose too high for the present CEA configuration.
B. **CORRECT:** Per OI-042-000, section 5.3.2, a shift update should be performed prior to and immediately following any reactivity manipulation and a cross-check of the reactivity manipulation is performed by comparing what was placed in the batch counter to the totalizer reading.
C. Incorrect: Per OI-042-000, section 5.3.2, a shift update should be performed prior to and immediately following any reactivity manipulation and a cross-check of the reactivity manipulation is performed by comparing what was placed in the batch counter to the totalizer reading. A reactivity brief is required before the first reactivity manipulation of the shift.
D. Incorrect: Per OI-042-000, section 5.3.2, a shift update should be performed prior to and immediately following any reactivity manipulation and a cross-check of the reactivity manipulation is performed by comparing what was placed in the batch counter to the totalizer reading.

Technical Reference(s): OI-042-000 revision 31
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPA00 obj. 3 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments:

5.3.2 Reactivity Management

5.3.2.1 A tool for tracking positive and negative reactivity changes, as well as blended makeup to the Volume Control Tank is contained on Attachment 6.12, Reactivity Addition Sheet. During plant conditions where frequent PMU and/or Boric Acid additions are required, entries can be used that summarize several additions. Completed copies of Attachment 6.12 do not need to be retained and may be discarded.

5.3.2.2 Reactivity information should be discussed at the Reactivity Brief. A guideline is contained on Attachment 6.11, Reactivity Brief Guideline. The Reactivity Brief should be conducted after shift turnover is complete and prior to the first reactivity manipulation of the Shift.

5.3.2.3 All reactivity changes, including water and boric acid additions, should be logged in the Operations Station Logs.

5.3.2.4 Cross-checks of PMU/ACID additions should be performed by comparing what was placed into the batch counter(s) versus what the totalizer(s) read.

5.3.2.5 A Shift Update should be held prior to and immediately following any reactivity manipulation.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.21	
	Importance Rating	2.9	

K/A Statement

Equipment Control: Knowledge of pre- and post-maintenance operability requirements.

Proposed Question: RO 70 Rev: 0

Given:

- Plant is at 100% Power
- At 0100, the BOP operator bypasses the Channel C High LPD and Low DNBR trip bistables for a scheduled two hour I&C surveillance on Channel C Core Protection Calculator
- The I&C technician informs the CRS that the CPC has failed the surveillance and will require a card replacement

Operability will be tracked by ___(1)___ when the work begins. The Shift Manager will authorize operability following retest on ___(2)___.

	(1)	(2)
A.	the work package	OP-100-010, Att. 7.1, TS/TRM Entry Guidelines
B.	OP-100-010, Att. 7.1, TS/TRM Entry Guidelines	OP-100-010, Att. 7.2, EOS Checklist
C.	the work package	OP-100-010, Att. 7.2, EOS Checklist
D.	OP-100-010, Att. 7.1, TS/TRM Entry Guidelines	OP-100-010, Att. 7.1, TS/TRM Entry Guidelines

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: The work package is not used to track operability and OP-100-010 ATT 7.2 is used for when a component fails the surveillance
B. **CORRECT:** IAW OP-100-010 the initial work is performed using OP-100-010 Att. 7.1 TS/TRM Entry guidelines. When work components fail a surveillance, OP-100-010 ATT 7.2 EOS checklist is required.
C. Incorrect: The work package is not used to track operability and when a component fails, OP-100-010 ATT 7.2 EOS Checklist is required.
D. Incorrect: IAW OP-100-010 the initial work is performed using OP-100-010 Att. 7.1 TS/TRM Entry guidelines. . When work components fail a surveillance, OP-100-010 ATT 7.2 EOS checklist is required.

Technical Reference(s): OP-100-010 , 5.1 – 5.3 , rev 308
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPA00 Obj. 2 (As available)

Question Source: Bank # X Question #69
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2011 NRC Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

5.0 PROCEDURE

> **NOTE**
>
> Normally, EOS Checklists can be created using the EOS Database. When the database is used to generate EOS Checklists, then TS/TRMs listed in the plant's Component Database are available on the EOS CHECKLIST screen by clicking on the TS/TRM drop down box.

5.1 GENERAL INSTRUCTIONS

5.1.1 Using Attachment 7.1, TS/TRM Entry Guideline, evaluate the need for entering a TS/TRM for the following situations: **[P-23414]**

- Declaring Inoperable, equipment which performs or supports a specified safety function
- Plant conditions require an entry into TS/TRM action statement

5.1.2 If a TS/TRM Component/System is declared Inoperable, and either of the following is met, then also complete Attachment 7.2, EOS Checklist:

- Evolution, test or plant condition affectingTS/TRM Component/System operability will not be completed or restored within the shift
- Operability retest is required (excluding channel checks in accordance with OP-903-001, Technical Specification Surveillance Logs)

5.1.3 TRM LCO entries for fire assemblies and fire protection/detection systems do not need to be tracked by this procedure if both the following requirements are met: **[P-5763, P-10314]**

- Fire impairment is issued manually per FP-001-015 or electronically using the Waterford 3 Fire Impairment Database (refer to Attachment 7.6, Fire Impairment Review Checklist, when preparing Fire Impairments).
- Required compensatory actions are implemented by the Fire Impairment.

5.1.4 An EOS Checklist may be used to ensure retest requirements are satisfied for Components/Systems declared Inoperable, that do not meet the criteria of step 5.1.1. Examples of this include but are not limited to:

- Security Diesel Generator
- EBA Compressor

5.1.5 The EOS Tracking Program maintains an index of EOS Checklists.

5.1.6 After racking out / down 4.16KV / 6.9KV breakers, Attachment 7.3, 4.16 KV / 6.9 KV Breaker Checklist, should be listed on Attachment 7.2, EOS Checklist in Block 10 or as a line item in Block 13 in reference to the clearance or deviation that removed it from service. The signature in Block 14 of Attachment 7.2, should document proper completion of the checklist.

5.1.7 If a Technical Specification surveillance cannot be completed due to out of service equipment, then verify the incomplete surveillance is tracked by either creating a new EOS Checklist or by adding the surveillance to an existing EOS Checklist. **[P-15648]**

5.1.8 If a Work Order is emerged on out of service equipment, then verify the Work Order is tracked by either of the following:

- Create a new EOS Checklist
- Add the Work Order and any associated retests to an existing EOS Checklist

5.1.9 Prior to declaring a component operable or exiting a TS/TRM action statement, then utilize Attachment 7.5, TS/TRM Operability Guideline, as follows:

- The RO/SRO/STA reviews Attachment 7.5, TS/TRM Operability Guideline, to signify that the component / system is ready to be returned to OPERABLE status.
- The SM/CRS or authorizing SRO, review the applicable Attachment 7.5, TS/TRM Operability Guideline to signify that the component / system is ready to be returned to OPERABLE status.

5.2 GUIDELINES FOR PERFORMING ATTACHMENT 7.1, TS/TRM ENTRY GUIDELINE AND ATTACHMENT 7.5, TS/TRM OPERABILITY GUIDELINE

NOTE

Attachment 7.1, TS/TRM Entry Guideline, is a guideline for determining TS/TRM entries and declaring Components/Systems Inoperable. **[P-191]**

5.2.1 For declaring equipment Inoperable, use Attachment 7.1, TS/TRM Entry Guideline, and the impact statement when available, to brief appropriate shift personnel.

NOTE

A shift update should only be used when declaring equipment operable that had been declared Inoperable within the same shift (i.e. equipment inoperability was for scheduled maintenance).

5.2.2 For declaring equipment operable, use Attachment 7.5, TS/TRM Operability Guideline, to perform a shift update or brief with appropriate personnel.

5.2.3 The following provides general guidelines for completing Attachment 7.1.

- Line 1 Identify component Equipment Number, Description and System. **[P-20653]**
- Line 2 Identify the reason for removing the component from service.
- Line 3 RO/SRO/STA reviews the appropriate TS/TRMs for applicability, considering the listed TS/TRMs and questions.
- Line 4 RO/SRO/STA identifies TS/TRMs required to be entered for component being removed from service for the current Plant Mode.
- Line 5 RO/SRO/STA identifies applicable TS/TRMs not required to be entered, with justification and any notes needed to conduct a brief.
- Line 5a RO/SRO/STA documents his or her review by signing Guideline.
- Line 6 The SM/CRS performs all of the following:
  - Reviews and concurs with the TS/TRM requirements identified by RO/SRO/STA.
  - Verifies that the plant can support removing this component / system from service. **[P-189]**
  - Determines time Component / System was rendered Inoperable.
- Line 6a The SM/CRS or authorizing SRO documents his or her approval by signing Guideline.

- Line 7 Shift conducts Brief for TS/TRM Entry:
  - Include all available Control Room personnel
  - Identify and review all applicable TS/TRMs
  - Discuss action statements and compliance
  - If required, then discuss need for additional monitoring
- Line 8 Log all TS/TRM entries in Station Log. **[P-189, P-23414]**
- Line 9 If equipment being removed from service is not planned, then verify EOOS program is updated.
- Line 10 If the work requires an Operability retest other than OP-903-001 channel check or the work will be carried into the next shift, then verify Attachment 7.2, EOS Checklist, is also completed.

5.3 GUIDELINE FOR COMPLETING ATTACHMENT 7.2, EOS CHECKLIST

NOTE

The information for completing Attachment 7.2 may be entered in any order. The use of the term "signature" in this section includes electronic authorization using the EOS database for review, authorization and closure of EOS Checklists and closure of work documents on EOS Checklists.

- Block 1 Enter the EOS Checklist number.
- Block 2 Enter component Equipment Tag Number (UNID), System, and a description of the condition causing the Inoperable status. **[P-20653]**
- Block 3 Enter percent Reactor Power.
- Block 4 Enter current Plant Mode.
- Block 5 If mode changes are not allowed (T.S. 3.0.4 or TRM 3.0.4 is applicable), then indicate by checking NO. Otherwise, check YES. **[P-11571, P-23414]**
- Block 6 Indicate if a TS Addendum Logsheet is required.
- Block 7 Indicate if a Departmental Action Notice is issued, then document the number assigned, if applicable.
- Block 8 If the equipment is Inoperable due to a planned outage, then indicate by checking YES. Otherwise, check NO and notify the Shift Manager that a risk assessment is required to be performed in accordance with OI-037-000, Operations Risk Assessment Guideline.
- Block 9 List any Compensatory Actions taken or contingencies planned. Compensatory Actions should be assigned, whenever feasible, to maintain equipment available and heighten operator monitoring. Such actions might include the following:
    - Additional monitoring/tours
    - Staging tape at an Inoperable Control Room airlock door
    - Taking readings locally
    - Placing DCT fans in fast
    - Any other action taken to minimize the impact of the Inoperable equipment
    - If unavailability is expected to exceed 60 hours, then establish Risk Management Actions in accordance with OI-037-000, Operations' Risk Assessment Guideline.

NOTE

When equipment becomes Inoperable which is not scheduled, a Work Request is placed on the EOS for work tracking. Once converted to a Work Order this should be used to replace the Work Request.

- Block 10 Enter document number and document type for documents affecting operability of TS/TRM related components. Examples of items that may be listed include, but are not limited to:
  [P-20542, P-20652]
  - W/Os
  - W/Rs
  - Tasks
  - Temporary Modifications
  - Clearances
  - Breaker Checklists
  - Condition Reports
  - Valve / Breaker Lineups
  - Deviations
  - Technical Specification Logs if not taken within OP-903-001 required surveillance intervals (this may be listed in either Block 10 or 13)
  - Engineering Changes
- Block 11 Enter Equipment Tag Number (UNID) for each document.
- Block 12 If the item is a Work Order, then enter a brief description of the work to be performed. Otherwise, Block 12 may be N/A'd.
- Block 13 If an operability retest is required, then enter a brief description of the operability retest, otherwise enter NONE. ASME required exams (i.e. VT2) are regulatory required exams to be performed prior to or during restoration, and should be included in this block.
  [CR-WF3-2011-8222, P-166, P-209, P-10273, P-11910, P-20656, P-15174]

- Block 14 Verify the document listed has been completed to the point that it no longer affects operability of the component, then enter signature, date, and time. Examples of when the item may be released from the EOS Checklist include, but are not limited to: **[P-166]**
  - Satisfactory completion of the Work Activity including retests
  - Release of the clearance in accordance with OP-102, Protective and Caution Tagging
    - If component configuration control is transferred to an alternate method (e.g. deviation tracking sheet) and the component position affects operability of the component/system listed on the EOS Status Checklist, then the alternate configuration controlling document should be listed on the EOS Status Checklist.
  - Proper completion of Attachment 7.3, 4.16 KV / 6.9 KV Breaker Checklist
  - Proper closure of the Temporary Modification in accordance with, EN-DC-136, Temporary Modifications.
  - Verification that the document does not affect operability
  - In some cases the final maintenance testing required to close a Work Order must be deferred until certain plant conditions are established (e.g. the system is at Normal Operating Pressure for leak checks). In these cases the system may be declared operable for the purpose of mode changes provided. **[P-15648]**
    - The lead worker signs Work Document signifying the component has been restored and no additional work will be performed on that Work Document
    - Work Documentation supports declaring the system operable and clearing the EOS
    - Reasonable assurance is established via applicable methods, (UT, dye testing, visual inspection, etc.) the component is capable of performing its intended function
    - Operability retesting is complete
    - Any ASME-required testing (i.e. VT-2) is transferred to another EOS to allow tracking of the completion of these required exams when proper conditions exist.
  - The following personnel have authority to sign Block 14:
    - Any person qualified as Shift Technical Advisor
    - Licensed Reactor Operators
    - Licensed Senior Reactor Operators

- Block 15 — Enter TS/TRM numbers for all TS/TRMs being entered and all TS/TRMs that have a reasonable potential of being entered if plant conditions change.
- Block 16 — Enter a brief description of the Limiting condition for operation.
- Block 17 — Enter a brief description of the Required Action(s) to be taken.
- Block 18 — If the TS/TRM LCO action is being entered, then enter YES. If the TS/TRM LCO action is not being entered, then enter NO. **[P-23414]**
- Block 19 — If the TS/TRM applies only to specific equipment different from the equipment in Block 2, then that Equipment Tag Number may be entered here for clarification. **[P-20653]**
- Block 20 — If NO was checked in Block 18, then enter a Justification (e.g. not required in current mode, or redundant compressor is operable, etc…). Justification should always address the specified safety function(s).
  Block 20 may also be used for comments pertaining to the specific equipment listed in Block 19.
  Examples of comments that may be entered in Block 20 include, but are not limited to:
  - Exiting a TS/TRM for the specific equipment listed in Block 19 without closing the EOS Checklist
  - Entering a TS/TRM for the specific equipment listed in Block 19 after the EOS Checklist has been opened
- Block 21 — List the applicable modes for the LCO, and any plant conditions required in that mode.
- Block 22 — The RO/SRO/STA performs the following for EOS Opening:
  - Verifies all applicable LCOs are documented on the EOS Checklist.
  - Verifies the plant can support removing this component / system from service. **[P-189, P-23414]**
  - Documents his or her review by signing the Checklist.

NOTE

The CRS can perform / authorize the EOS Opening if the SM is absent from the Control Room and the Component / System outage is planned. If utilizing the authorizing SRO, then concurrence is required from the SM/CRS prior to signing the Checklist to declare TS or TRM equipment Inoperable. The authorizing SRO cannot sign both the review and approval Blocks.

- Block 23 The SM/CRS or authorizing SRO performs the following for EOS Opening:
  - Verifies the TS/TRM operability determination is correct
  - Verifies that the plant can support removing this component / system from service **[P-189]**
  - Declares the Component / System Inoperable by signing the Checklist and entering the date and time the Component/System was declared Inoperable
  - If utilizing authorizing SRO, then print name of SM/CRS contacted.
  - A copy may be printed from the EOS database and placed in the EOS Book as a backup to the database. This copy is not required to be maintained up to date when using the database.
- Block 24 The RO/SRO/STA reviews the EOS Checklist; utilizing Attachment 7.5, TS/TRM Operability Guideline, for proper completion, and signs to signify that the component / system is ready to be returned to OPERABLE status.

NOTE

(1) Documentation that verifies operability of the component/system need not be attached to the EOS Checklist. The signature in Block 14 should document verification of satisfactory item closure for each item listed on the EOS Checklist.

(2) The CRS can perform / authorize the EOS closure if the SM is absent from the Control Room and the Component / System outage was planned. If utilizing the authorizing SRO then concurrence is required from the SM/CRS prior to signing the Checklist to declare TS or TRM equipment operable. The authorizing SRO cannot sign both the review and approval Blocks.

- Block 25 The SM/CRS or authorizing SRO, performs the following for EOS Checklist closure:
  - Review the EOS Checklist for proper completion.
  - Review applicable Attachment 7.5, TS/TRM Operability Guideline.
  - Sign block 25 including date and time equipment was declared operable.
  - If utilizing authorizing SRO, then print name of SM/CRS contacted.
  - Print either a final copy from the EOS database showing all electronic signatures or a copy manually updated by hand and transmit to records. Any handwritten updates to former copies of the EOS Checklist maintained in the EOS Book should be compared to the final copy to be transmitted to records and diffferences resolved.
  - The EOS database should be updated as soon as possible if a manually updated copy of the EOS Checklist is transmitted to records.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.5	
	Importance Rating	2.9	

K/A Statement

Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO 71 Rev: 0

Which ONE of the following radiation monitors **(1)** is specified as the primary monitor with the designed sensitivity to measure small Primary to Secondary leakage **AND** which radiation monitor **(2)** is specified as qualified for use if the primary monitor fails?

	(1)	(2)
A	MS Line N-16 Monitor PRM-RE5501-1 (2)	Steam Generator 1 and 2 Blowdown PRM-IRE-0100X
B	MS Line N-16 Monitor PRM-RE5501-1 (2)	AE Discharge Monitor PRM-IRE-0004
C	AE Discharge Monitor PRM-IRE-0004	Steam Generator 1 and 2 Blowdown PRM-IRE-0100X
D	AE Discharge Monitor PRM-IRE-0004	MS Line N-16 Monitor PRM-RE5501-1 (2)

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: AE Discharge Radiation Monitor is considered the primary Radiation Monitor with the designed sensitivity to measure small Primary to Secondary Leakage. The Blowdown Rad Monitor is listed as available indication but is not the primary indication or backup.
B. Incorrect: AE Discharge is primary and MS Line N-16 is backup
C. Incorrect: Part 1 is correct. The Blowdown Rad Monitor is listed as available indication but is not the primary indication or backup.
D. **CORRECT:** AE Discharge Radiation Monitor is considered the primary Radiation Monitor with the designed sensitivity to measure small Primary to Secondary Leakage. The MS Line N16 Rad Monitors may be used as verification for the AE Discharge Radiation Monitor or as primary indication if the AE Discharge Radiation Monitor is out of service.

Technical Reference(s): OP-901-202 revision 11
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO02 obj. 3 (As available)

Question Source: Bank # 08934
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2009 NRC RO Exam

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

10 CFR Part 55 Content: 55.41 11
55.43

Comments:

B_2 INDICATIONS

NOTE

AE Discharge Radiation Monitor is considered the primary Radiation Monitor with the designed sensitivity to measure small Primary to Secondary Leakage. The MS Line N16 Rad Monitors may be used as verification for the AE Discharge Radiation Monitor or as primary indication if the AE Discharge Radiation Monitor is out of service. Low level leakage may not be evident on the MS Line N16 Radiation Monitors. A rise in the AE Discharge Radiation Monitor activity alone may warrant taking actions in this procedure.

1. Secondary activity rising ■2 GPD from the average steady state leakage value as indicated by ANY of the following:

   1.1 AE Discharge Radiation Monitor, PRM-IRE-0004 rising with a corresponding rise on EITHER of the following MS Line N16 Rad Monitors:

   - RE5501-1 (Leak Rate 1)

   OR

   - RE5501-2 (Leak Rate 2)

   1.2 PRM-IRE-0004 out of service with rising activity on EITHER of the following MS Line N16 Rad Monitors:

   - RE5501-1, (Leak Rate 1)

   OR

   - RE5501-2, (Leak Rate 2)

   1.3 Rising Secondary activity indicated on the following Radiation Monitors (listed in order of sensitivity):

   - PRM-IRE-0002, Condenser Vacuum Pump
   - PRM-IRE-0100X, Steam Generator 1 and 2 Blowdown
   - PRM-IRE-5500A, Main Steam Line 1
   - PRM-IRE-5500B, Main Steam Line 2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.15	
	Importance Rating	2.9	

K/A Statement

Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO 72 Rev: 0

Per OI-038-000, Emergency Operating Procedures Operations Expectations/Guidance, following a momentary loss of power or voltage dip, some radiation monitors needed for Emergency Plan may require ___________?

A. a restart of the sample pump

B. verification of proper setpoint

C. reset of their power supply

D. local purge operation

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: OI-038-000, step 5.3.4.1, states that some radiation monitors sample pumps needed for Emergency Plan may require restarting following a loss of power or voltage dips. Failure to restart sample pumps could delay the ED classification (W-3 specific issue)

B. Incorrect: OI-038-000, step 5.3.4.1, states that some radiation monitors sample pumps needed for Emergency Plan may require restarting following a loss of power or voltage dips. This distractor is plausible because it is an action that can be performed on a rad monitor per OP-004-001, Radiation Monitors

C. Incorrect: OI-038-000, step 5.3.4.1, states that some radiation monitors sample pumps needed for Emergency Plan may require restarting following a loss of power or voltage dips. This distractor is plausible because it is feasible that a voltage dip could trip the local supply breaker.

D. Incorrect: OI-038-000, step 5.3.4.1, states that some radiation monitors sample pumps needed for Emergency Plan may require restarting following a loss of power or voltage dips. This distractor is plausible because it is an action that can be performed on a rad monitor per OP-004-001, Radiation Monitors

Technical Reference(s): OI-038-000 step 5.3.4.1 revision 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 4 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 11
55.43 ____

Comments:

5.3 REACTOR TRIP RECOVERY

5.3.1 Reset ESFAS Actuation

- If an EFSAS has occurred, the operator should reset the actuation and secure the applicable components using the Standard Appendix or the offnormal procedure if additional guidance is necessary.
- This step should be completed prior to exiting the EOPs.

5.3.2 Sample the RCS

- The operator should notify chemistry to sample the RCS to comply with the required Technical Specification sample.
- This step should be completed prior to exiting the EOPs.

5.3.3 Calculate Shutdown Margin

- If emergency boration is in progress, it is necessary for the operator to calculate Shutdown Margin and secure the boration prior to exiting this procedure.

5.3.4 Radiation Monitor Restoration

5.3.4.1 Some radiation monitors' sample pumps needed for Emergency Plan may require restarting following a loss of power or voltage dips. Examples are Plant Stack WRGM, Plant Stack PIGs, Condenser WRGM and Fuel Handling Building PIGs. OP-004-001, Radiation Monitoring, limitation 3.2.1 provides a more comprehensive list of Radiation Monitors that will require their pumps to be restarted.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.5	
	Importance Rating	3.7	

K/A Statement

Emergency Procedures/Plan: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question: RO 73 Rev: 0

Per OP-100-017, Emergency Operation Procedure Implementation Guide:

Simultaneous performance of more than one Optimal Recovery Procedures (ORP) is ____(1)____ . Simultaneous performance of an Off-Normal Operating Procedure and an Optimal Recovery Procedure is allowed _____(2)_____ .

	(1)	(2)
A.	allowed	at the discretion of the CRS
B.	allowed	only if the ORP directs its use
C.	not allowed	at the discretion of the CRS
D.	not allowed	only if the ORP directs its use

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: OP-100-017 step 5.11.1 states simultaneous performance of more than one optimal recovery procedure is prohibited. Step 5.10.2 states, other procedures (Normal Operating Procedures, Alarm Response, Offnormals) may be used as directed by the CRS.

B. Incorrect: OP-100-017 step 5.11.1 states simultaneous performance of more than one optimal recovery procedure is prohibited. Part 2 is correct.

C. **CORRECT:** OP-100-017 step 5.11.1 states simultaneous performance of more than one optimal recovery procedure is prohibited. Step 5.10.2 states, other procedures (Normal Operating Procedures, Alarm Response, Offnormals) may be used as directed by the CRS.

D. Incorrect: Part 1 is correct: Step 5.11.1 states simultaneous performance of more than one optimal recovery procedure is prohibited.

Technical Reference(s): OP-100-017 revision 1
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 4 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

5.10 Reference to Other Procedures

5.10.1 The EOPs are designed to minimize the interface with other procedures. The EOPs include standard appendices that operators may need during recovery. This provides the control room staff with an easily located set of instructions while minimizing the simultaneous use of other procedures. Where standard appendices are needed, the required procedure will be referenced in the body of the EOP.

5.10.2 Other procedures (Normal Operating Procedures, Alarm Response, Offnormals) are usually not normally needed to supplement the EOPs, but may be used as directed by the CRS.

5.11 USE OF MULTIPLE EMERGENCY OPERATING PROCEDURES

5.11.1 Simultaneous performance of more than one optimal recovery procedure is prohibited. The FRP is used for multiple events (except as noted below).

5.11.2 Certain events do not require offsite power in order to adequately mitigate the effects of the event. For this reason, if a loss of offsite power occurs concurrently with any of the following events: LOCA, ESD, or SGTR then the diagnosed ORP should be entered. LOOP/LOFC should only be entered when it is the only event in progress.

5.11.3 The LOMF event requires offsite power to be available to implement the ORP.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.9	
	Importance Rating	3.8	

K/A Statement

Emergency Procedures/Plan: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: RO 74 Rev: 0

Given:

- An Excess Steam Demand is in progress
- The crew is performing required actions in OP-902-004, Excess Steam Demand
- Chemistry is unable to obtain an RCS Boron sample
- Reactor power is dropping

Emergency Boration may be terminated if which of the following conditions are met?

A. Pressurizer level is approaching the upper end of the control band and HPSI Throttle Criteria is being met.

B. Pressurizer level is approaching the upper end of the control band and a Shutdown Margin calculation is performed as soon as plant conditions permit.

C. Reactor Vessel Level Monitoring indicates the core is covered and HPSI Throttle Criteria is being met.

D. Reactor Vessel Level Monitoring indicates the core is covered and a Shutdown Margin calculation is performed as soon as plant conditions permit.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: OI-038-000 step 5.4.52 states that if an RCS Boron concentration is not attainable, emergency boration can be secured if the following conditions are met 1. Reactor power is dropping, 2. Pressurizer level is approaching the upper end of the control band and 3. HPSI Throttle Criteria is being met. This guidance precludes going water solid in the pressurizer. This guidance is generic such that it applies to all Emergency Operating Procedures.

B. Incorrect: Part 1 is correct. Performing a SDM as soon as conditions permit is part of the OI-038-000 step 5.4.52 guidance but is not part of the criteria.

C. Incorrect: Part 2 is correct. RVLMS indicating the core is covered is credible since the applicant may determine that inventory control is of the concern. Inventory control is maintained by ensuring that HPSI throttle criteria is met.

D. Incorrect: OI-038-000 step 5.4.52 states that if an RCS Boron concentration is not attainable, emergency boration can be secured if the following conditions are met 1. Reactor power is dropping, 2. Pressurizer level is approaching the upper end of the control band and 3. HPSI Throttle Criteria is being met. . Performing a SDM as soon as conditions permit is part of the OI-038-000 step 5.4.52 guidance but is not part of the criteria.

Technical Reference(s): OI-038-000 step 5.4.52 revision 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 4 (As available)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 10
55.43 ______

5.4.50 Maintain SG Level

- Overfeeding of Steam Generators may cause excessive RCS cooldown. The operator should not add feedwater to a dry steam generator if another steam generator still contains water. Re-establish feedwater only to the Steam Generator that is not dry. If both Steam Generators become dry, slowly refill only one Steam Generator to reinitiate core cooling.

5.4.51 Maintain Shutdown Margin

- None

5.4.52 Maintain Shutdown Margin During the Cooldown

- Shutdown Margin is required to be maintained during the use of the EOPs with the following exception:
    - During the SBO and FRP procedure the reactor is only required to be maintained shutdown. This requirement is a 1%ΔK/K.
    - If unable to obtain an RCS boron sample or time does not permit performing a Shutdown Margin calculation, then Emergency Boration may be terminated under the following conditions:
        - Pressurizer level is approaching the upper end of the control band
        - HPSI Throttle Criteria is met
        - Reactor power is stable or dropping

        A Shutdown Margin calculation should be performed as soon plant conditions permit.
- Shutdown Margin is maintained with all CEAs fully inserted with Tc greater than 400°F.

5.4.53 Maintain Success Paths

- None

5.4.54 Makeup to the CSP

- ACCW flow to EFW should not be established until the CSP contents have depleted. The lineup should commence at 25% CSP level, however, ACCW flow to the suction of EFW should be completed before a CSP level of 11% is reached to prevent cavitation of the EFW Pumps.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.16	
	Importance Rating	3.5	

K/A Statement

Emergency Procedures/Plan: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.

Proposed Question: RO 75 Rev: 0

Per OP-100-017, Emergency Operating Procedure Implementation Guide, which of the following criteria (if any) would allow the crew to remain in a selected Optimal recovery procedure if the safety function status checklist is not being met?

A. The crew must exit to the functional recovery procedure any time the safety function status checklist is not met.

B. Two events are in progress and a shift brief is performed which must include the Shift Manager.

C. A single event is in progress and transitioning to the functional recovery procedure will provide no additional guidance to correct the failed safety function.

D. The reason for not meeting the safety function is known and TSC concurrence must be obtained.

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: OP-100-017 step 5.6.2 states that if a single event is in progress and transitioning to the functional recovery procedure will provide no additional guidance to correct the failed safety function, the decision may be made to stay in the ORP.
B. Incorrect: OP-100-017 step 5.6.2 states that if a single event is in progress and transitioning to the functional recovery procedure will provide no additional guidance to correct the failed safety function, the decision may be made to stay in the ORP. This step states that a brief should be performed which includes the SM.
C. **CORRECT**: OP-100-017 step 5.6.2 states that if a single event is in progress and transitioning to the functional recovery procedure will provide no additional guidance to correct the failed safety function, the decision may be made to stay in the ORP.
D. Incorrect: OP-100-017 step 5.6.2 (first bullet) states that the reason for not meeting the safety function is known and operator action produces immediate response to toward satisfying the failed safety function. TSC concurrence is required in some steps in the EOPs but is not required in this situation.

Technical Reference(s): OP-100-017 steps 5.6.2 and 5.11.1
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 4 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments:

5.6 SAFETY FUNCTION STATUS CHECKLIST

5.6.1 Each ORP has its own Safety Function Status Check (SFSC) which must be used whenever the ORP is in use. The SFSC is used to check the status of safety functions. By satisfying any condition of the SFSC acceptance criteria (Condition 1, Condition 2, or Condition 3 if applicable), the operating staff is assured that the actions being taken are maintaining the plant in a safe condition. If SFSC criteria are not satisfied, then the operators will take corrective actions to satisfy the safety functions, implement another ORP, or exit to the FRP.

5.6.2 The operator may stay in this procedure under any of the following conditions:

- The reason for not meeting the safety function is known and operator action produces immediate response toward satisfying the failed safety function
- If a single event is in progress and transitioning to the FRP will provide no additional guidance to correct the failed safety function (A shift brief should be performed, including the SM, prior to making this decision)

5.6.3 When using the FRP, the status of each Safety Function is tracked using the Safety Function Status Sheet. This tracking page is used to identify the initial criteria met for each Safety Function and tracks the completion of each subprocedure.

5.11 USE OF MULTIPLE EMERGENCY OPERATING PROCEDURES

5.11.1 Simultaneous performance of more than one optimal recovery procedure is prohibited. The FRP is used for multiple events (except as noted below).

5.11.2 Certain events do not require offsite power in order to adequately mitigate the effects of the event. For this reason, if a loss of offsite power occurs concurrently with any of the following events: LOCA, ESD, or SGTR then the diagnosed ORP should be entered. LOOP/LOFC should only be entered when it is the only event in progress.

5.11.3 The LOMF event requires offsite power to be available to implement the ORP.

Examination Outline Cross-Reference:		RO	SRO
	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000007 2.4.4	
	Importance Rating		4.7

K/A Statement

2.4.4 - Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: SRO 1 Rev: 0

Given:

- Plant is at 15% power
- The crew is making preparations to synch the Main Generator to the grid
- Main Feedwater Pump ‘A’ is operating
- Startup Transformer ‘B’ trips on the Sudden Pressure Relay resulting in a reactor trip
- Pressurizer pressure is 1950 PSIA and rising
- Pressurizer level is 15% and stable
- One backup Charging Pump has started
- RCS Temperature is 541°F and rising
- EDG ‘B’ is running loaded
- NAO reports no precipitation

Which of the following describes the correct procedure entry and required action?

A. OP-902-001, Reactor Trip Recovery and perform Appendix 20, Operation of DCT Sump Pumps within 30 minutes

B. OP-902-001, Reactor Trip Recovery and start an additional Charging Pump

C. OP-902-003, Loss of Offsite Power/ Loss of Forced Circulation Recovery and perform Appendix 20, Operation of DCT Sump Pumps within 30 minutes

D. OP-902-003, Loss of Offsite Power/ Loss of Forced Circulation Recovery and start an additional Charging Pump

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Using OP-902-009 Appendix 1, these set of conditions will require entry into OP-902-001. Appendix 20 is required to be performed, but it is not required to be completed within 30 minutes as there is not a probable maximum precipitation event in progress.
B. **CORRECT** Using OP-902-009 Appendix 1, these set of conditions will require entry into OP-902-001. Three charging pumps should be running with the present conditions. The crew would be expected to manually start the third charging pump to restore PZR level to 33%.
C. Incorrect: One side of offsite power is still available; entry into OP-902-003 is not required unless both trains of offsite power are lost. Appendix 20 is required to be performed, but it is not required to be completed within 30 minutes as there is not a probable maximum precipitation event in progress. Plausible because this direction is contained in procedure OP-902-003.
D. Incorrect: One side of offsite power is still available; entry into OP-902-003 is not required unless both trains of offsite power are lost. Three charging pumps should be running with the present conditions. The crew would be expected to manually start the third charging pump to restore PZR level to 33%. Plausible because this direction is contained in OP-902-003.

Technical Reference(s): OP-902-001 revision 14
(Attach if not previously provided) OP-902-009 Appendix 1 revision 309
(including version/revision number) TGOP 902-001 revision 304

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PP030 obj. 1 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

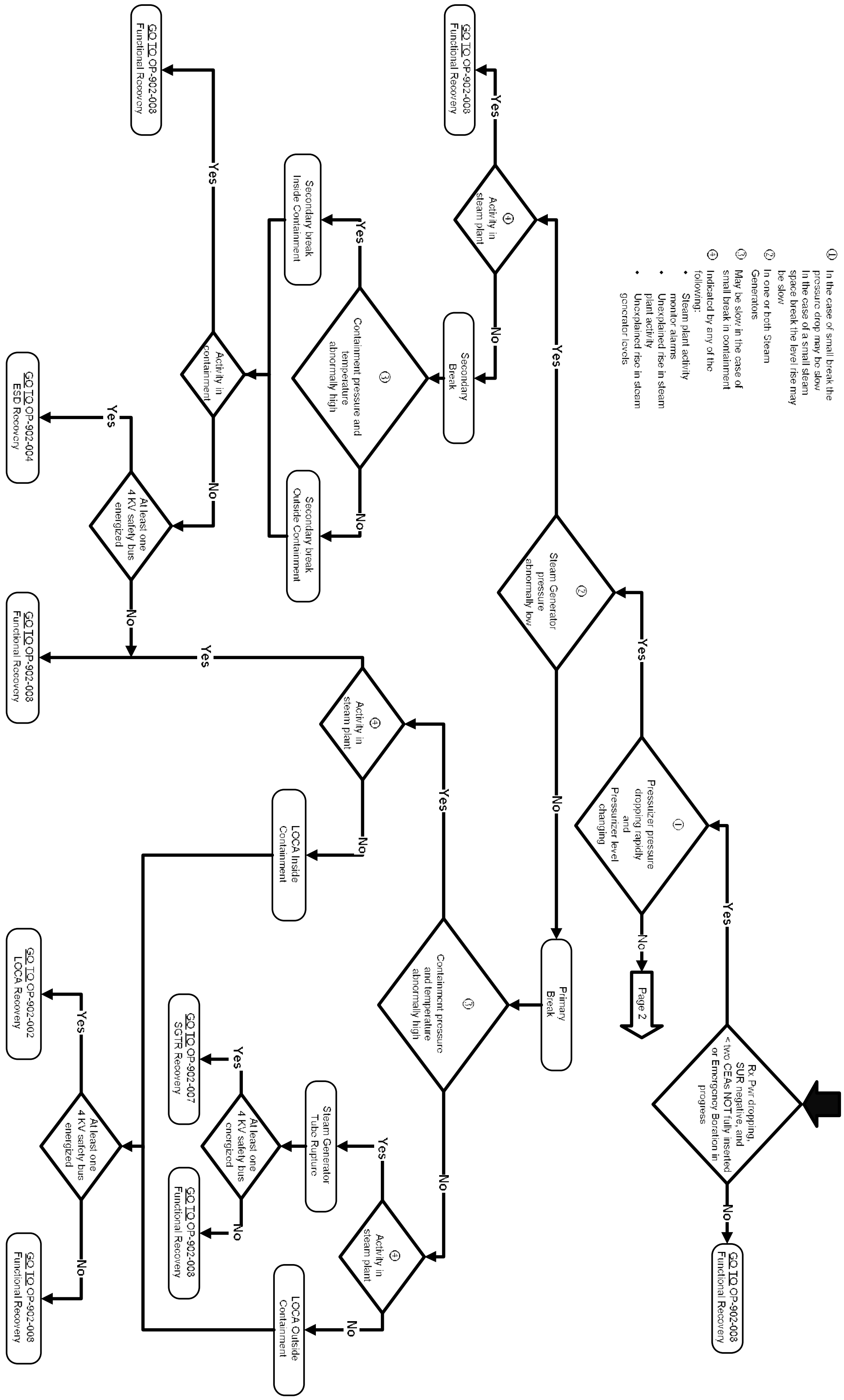
Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

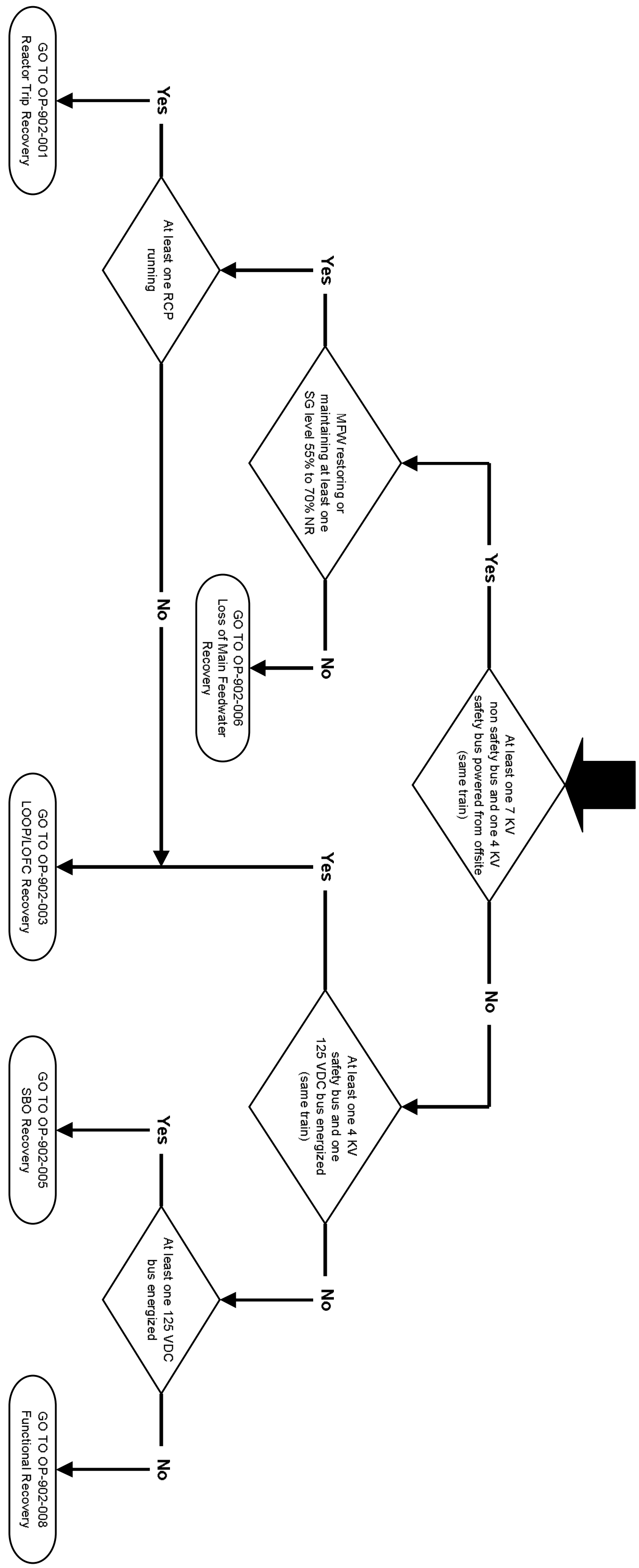
Comments:

Diagnostic Flow Chart



WATERFORD 3 SES STANDARD APPENDICES	OP-902-009 Revision 309 Page 5 of 195
	Appendix 1 Page 2 of 2

Diagnostic Flow Chart (Cont'd)



Operation of DCT Sump Pumps

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

This attachment should be performed following any power interruption to either the 3A or 3B safety buses (as directed from EOPs).

IF a Probable Maximum Precipitation (PMP) event is in progress **AND ANY** Dry Cooling Tower (DCT) Motor Driven Sump Pump is unavailable, **THEN** **BOTH** of the following shall be performed for the affected DCT sump to prevent flooding of associated 315A(B) Motor Control Center and Transformer within time frames as listed:

- One DCT Motor Driven Sump Pump is aligned for operation within 30 minutes of the PMP event.
- The DCT Portable Sump Pump (diesel driven) is aligned for operation within 3 hours of the PMP event.

____ 1.1 At MCC-314A, place the following switches to Bypass:

- DCT #1 Sump Pump A Radiation Monitor bypass switch
- DCT #2 Sump Pump A Radiation Monitor bypass switch

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
Confirm Diagnosis	
* 1. Confirm diagnosis of an uncomplicated reactor trip by checking Safety Function Status Check Acceptance Criteria are satisfied.	1.1 GO TO **ONE** of the following: • Appendix 1, "Diagnostic Flowchart" • OP-902-008, "Functional Recovery Procedure"
Announce the Event	
2. Announce an uncomplicated reactor trip has occurred using the plant page.	
Classify the Event	
* 3. Advise the Shift Manager to REFER TO EP-001-001, "Recognition & Classification of Emergency Condition" and implement the Emergency Plan.	
Restore Operation of DCT Sump Pumps	
* 4. **IF** power has been interrupted to either 3A or 3B safety buses, **THEN** perform Appendix 20, "Operation of DCT Sump Pumps".	

INSTRUCTIONS	CONTINGENCY ACTIONS
Implement Placekeeping	
5. REFER TO Section 6.0, "Placekeeper" and record the time of the reactor trip.	
Verify Pressurizer Level Control	
* 6. Check pressurizer level meets **BOTH** the following conditions: • Level is 7% to 60% • Trending to 33% to 60%	6.1 Control charging and letdown to restore pressurizer level 33% to 60%.
Verify Pressurizer Pressure Control	
* 7. Check pressurizer pressure meets **BOTH** the following conditions: • Pressure is 1750 psia to 2300 psia • Trending to 2125 psia to 2275 psia	7.1 Control pressurizer heaters and spray to restore pressurizer pressure 2125 psia to 2275 psia.
Verify RCS Temperature Control	
* 8. Check Steam Bypass Control System is controlling RCS T_C 530ºF to 550ºF.	8.1 Control Steam Bypass Control System or ADV to restore RCS T_C 530ºF to 550ºF.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - MODES 1, 2, AND 3

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection train,

b. One OPERABLE low-pressure safety injection train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water storage pool on a safety injection actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*#.

ACTION:

a. With one ECCS subsystem inoperable due to one low pressure safety injection train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than (a) and 100% of ECCS flow equivalent to a single OPERABLE ECCS subsystem available, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

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*With pressurizer pressure greater than or equal to 1750 psia.
#With RCS average temperature greater than or equal to 500°F.

c. With both LPSI trains inoperable due to less than 100% of ECCS flow equivalent to a single OPERABLE ECCS subsystem, restore at least one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
a. 2SI-V1556 (SI-506A)	a. Hot Leg Injection	a. SHUT
b. 2SI-V1557 (SI-502A)	b. Hot Leg Injection	b. SHUT
c. 2SI-V1558 (SI-502B)	c. Hot Leg Injection	c. SHUT
d. 2SI-V1559 (SI-506B)	d. Hot Leg Injection	d. SHUT

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

2. Verifying the ECCS piping is full of water.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the safety injection system sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying the action of the open permissive interlock (OPI) and isolation valve position alarms of the shutdown cooling system when the reactor coolant system pressure (actual or simulated) is between 392 psia and 422 psia.

2. A visual inspection of the safety injection system sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 380 cubic feet of granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 13.07 ± 0.03 grams of TSP from a TSP storage basket is submerged, without agitation, in 4 ± 0.1 liters of 120 ± 10°F water borated to 3011 ± 30 ppm, the pH of the mixed solution is raised to greater than or equal to 7 within 3 hours.

e. At least once per 18 months by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a. High pressure safety injection pump.

b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the low pressure safety injection pumps stop, the safety injection system sump isolation valves open.

f. By verifying that each of the following pumps required to be OPERABLE performs as indicated on recirculation flow when tested pursuant to the Inservice Testing Program:

1. High pressure safety injection pump differential pressure greater than or equal to 1429 psid.

2. Low pressure safety injection pump differential pressure greater than or equal to 168 psid.

SURVEILLANCE REQUIREMENTS (Continued)

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves by verifying that each ECCS throttle valve opens to the proper throttled position each time the valve is cycled:

HPSI System Valve Number				LPSI System Valve Number	
a.	SI-225A	e.	SI-227A	a.	SI-138A
b.	SI-225B	f.	SI-227B	b.	SI-138B
c.	SI-226A	g.	SI-228A	c.	SI-139A
d.	SI-226B	h.	SI-228B	d.	SI-139B

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow characteristics:

HPSI System - Single Pump (Cold leg injection mode)

The sum of the injection lines flow rates, excluding the highest flow rate, is greater than or equal to 675 gpm.

HPSI SYSTEM - Single Pump (Hot/cold leg injection mode)

With the system operating in the hot/cold leg injection mode, the hot leg flow must be greater than or equal to 436 gpm and within ± 10% of the cold leg flow.

LPSI System - Single Pump

Flow for each pump is greater than or equal to 4810 with the total developed head greater than or equal to 268 feet but less than or equal to 292 feet.

SURVEILLANCE REQUIREMENTS (Continued)

i. Each time HPSI Pump A/B is placed in or taken out of service in place of HPSI Pump A or B, the pump being placed in service shall be demonstrated OPERABLE by:

1. Verifying that each valve in the flow path is in its correct position; and

2. Verifying the pump starts manually and upon receipt of a SIAS test signal; and

3. Performing Surveillance Requirement 4.5.2f.1., if not previously accomplished within the required frequency.

j. Following any maintenance which drains portions of the system, by verifying the ECCS piping is full of water.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000009 2.2.40	
	Importance Rating		4.7

K/A Statement

2.2.40 - Ability to apply Technical Specifications for a system.

Proposed Question: SRO 2 Rev: 0

Given:

- The plant is at 100% power
- The AB bus is aligned to the B Train
- HPSI Pumps A and B are aligned for service
- LPSI Pump A is tagged out
- Technical Specification 3.5.2.a has been entered

HPSI Pump B breaker develops a malfunction that causes the breaker to be unable to close. Regarding ECCS equipment available to mitigate a SBLOCA, which of the following is true in accordance with Technical Specifications?

A. TS 3.5.2.b is applicable and may be exited once the AB HPSI pump is aligned to replace the B HPSI pump.

B. TS 3.5.2.b is applicable and may **ONLY** be exited when one ECCS pump is restored.

C. TS 3.0.3 is applicable and may be exited once the AB HPSI pump is aligned to replace the B HPSI pump.

D. TS 3.0.3 is applicable and may **ONLY** be exited when one ECCS pump is restored.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: TS 3.5.2.b will be entered because two ECCS subsystems are unavailable and 100% flow equivalency is available for both LPSI and HPSI trains. Per the TS 3.5.2 basis, the operable LPSI and HPSI Trains do not need to be the same train. Aligning the AB HPSI pump to replace the B HPSI pump will allow the B ECCS train to become operable allowing exit of 3.5.2.b

B. Incorrect: TS 3.5.2.b will be entered because two ECCS subsystems are unavailable and 100% flow equivalency is available for both LPSI and HPSI trains. Per the TS 3.5.2 basis, the operable LPSI and HPSI Trains do not need to be the same train. Aligning the AB HPSI pump to replace the B HPSI pump will allow the B ECCS train to become operable allowing exit of 3.5.2.b. Plausible because restoring 1 ECCS pump will allow exit of TS 3.5.2.b, but this is not the only method as described in choice A.

C. Incorrect: Per the TS 3.5.2 basis, the operable LPSI and HPSI Trains do not need to be the same train. The applicant concludes that the requirement is for the same train to be operable, he could determine that TS 3.0.3 is required. 3.0.3 may be exited if AB pump was aligned to the B train.

D. Incorrect: Per the TS 3.5.2 basis, the operable LPSI and HPSI Trains do not need to be the same train. The applicant concludes that the requirement is for the same train to be operable, he could determine that TS 3.0.3 is required. 3.0.3 may be exited if AB pump was aligned to the B train.

Technical Reference(s): TS 3.5.2
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: TS 3.5.2

Learning Objective: WLP-OPS-SI00 obj. 7 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41
55.43 2

3/4.5.2 ECCS SUBSYSTEMS - MODES 1, 2, AND 3

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection train,

b. One OPERABLE low-pressure safety injection train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water storage pool on a safety injection actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*#.

ACTION:

a. With one ECCS subsystem inoperable due to one low pressure safety injection train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than (a) and 100% of ECCS flow equivalent to a single OPERABLE ECCS subsystem available, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

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*With pressurizer pressure greater than or equal to 1750 psia.
#With RCS average temperature greater than or equal to 500°F.

c. With both LPSI trains inoperable due to less than 100% of ECCS flow equivalent to a single OPERABLE ECCS subsystem, restore at least one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

Each subsystem includes the piping, instruments, and controls to ensure the availability of an OPERABLE flowpath capable of taking suction from the RWSP on a SIAS and automatically transferring suction to the containment sump upon a recirculation actuation signal (RAS). The flowpath for each subsystem must maintain its designed independence to ensure that no single failure can disable both ECCS subsystems.

An ECCS subsystem is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they are not capable of performing their automatic design function, or if supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent trains. Due to the redundancy of trains and the diversity of trains, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of these ACTIONs is to maintain a combination of OPERABLE equipment such that 100% of the ECCS flow equivalent to a single OPERABLE subsystem remains available.

100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem exists when the equivalent of one HPSI train, one LPSI train, and a suction flow path as described in the LCO are OPERABLE. The OPERABLE components may be in opposite subsystems. The HPSI component of the 100% ECCS flow equivalent may be composed of any combination of OPERABLE HPSI components such that flow is available to all four RCS loops. The LPSI component of the 100% ECCS flow equivalent may be composed of any combination of OPERABLE LPSI components such that flow is available to any two RCS loops. This allows increased flexibility in plant operations when components in opposite subsystems are inoperable.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

3.5.2, ACTION (a) addresses the specific condition where the only affected ECCS subsystem is a single LPSI train. A LPSI train consists of a pump, and two injection flow paths, including motor-operated valves operated by a common AC power source. The availability of at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem is implicit in the definition of ACTION (a).

If LCO 3.5.2 requirements are not met due to the condition described in ACTION (a), then the inoperable LPSI train components must be returned to OPERABLE status within seven (7) days of discovery. This seven (7) day Allowed Outage Time is based on the findings of deterministic and probabilistic analysis CE NPSD-995, "CEOG Joint Applications Report for Low Pressure Safety Injection System AOT Extension". Seven (7) days is a reasonable amount of time to perform many corrective and preventative maintenance items on the affected LPSI train. CE NPSD-995 concluded that the overall risk impact of the seven (7) day Allowed Outage Time was either risk-beneficial or risk-neutral.

ACTION (b) addresses other scenarios where the availability of at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem exists but the full requirements of LCO 3.5.2 are not met. If conditions of ACTION (b) were to exist, then inoperable components must be restored within 72 hours of discovery. The 72 hour Allowed Outage Time is based on an NRC reliability study (NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975) and is a reasonable amount of time to effect many repairs.

ACTION (c) addresses the condition in which 100% ECCS flow is unavailable due to two inoperable LPSI trains and requires restoration of at least one LPSI train to OPERABLE status within one hour or the plant placed in HOT STANDBY in 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

In the event less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem exists due to other conditions, LCO 3.0.3 is entered and the plant must be brought to a MODE (MODE 3 with pressurizer pressure less than 1750 psia and RCS average temperature less than 500°F) in which the LCO does not apply.

ECCS SUBSYSTEMS (Continued)

When in mode 3 and with RCS temperature greater than or equal to 500°F two OPERABLE ECCS subsystems are required to ensure sufficient emergency core cooling capability is available to prevent the core from becoming critical during an uncontrolled cooldown (i.e., a steam line break) from greater than 500°F.

With the RCS temperature below 500°F and the RCS pressure below 1750 psia, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0. The requirement to dissolve a representative sample of TSP in a sample of water borated to be representative of post-LOCA sump conditions provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures. A boron concentration of 3011 ppm boron is postulated to be representative of the highest post-LOCA sump boron concentration. Post LOCA sump pH will remain between 7.0 and 8.1 for the maximum (3011 ppm) and minimum (1504 ppm) boron concentrations calculated using the maximum and minimum post-LOCA sump volumes and conservatively assumed maximum and minimum source boron concentrations.

- **(DRN 02-1635, Ch. 16; DRN 03-445, Ch. 26)**

With the exception of systems in operation, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will prevent water hammer, pump cavitation, and pumping noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SIAS or during SDC. The LPSI system has been evaluated for voids in the discharge piping. The piping system has been qualified for the hydraulic transient. In addition, the reactor has been qualified for an intrusion of a small gas bubble. Therefore, from a design basis standpoint, for injection capacity and prevention of water hammer, pump cavitation, and pumping noncondensible gas the LPSI system will be considered operable and full of water with the existence of voids in the system discharge legs. The 31 day frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the adequacy of the procedural controls governing system operation.

- **(DRN 02-1635, Ch. 16; DRN 03-445, Ch. 26)**

ECCS SUBSYSTEMS (Continued)

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

The requirement to verify the minimum pump differential pressure on recirculation flow ensures that the pump performance curve has not degraded below that used to show that the pump exceeds the design flow condition assumed in the safety analysis and is consistent with the requirements of ASME Section XI.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000011 EA2.04	
	Importance Rating		3.9

K/A Statement

EA2.04 - Ability to determine or interpret the following as they apply to a Large Break LOCA: Significance of PZR readings

Proposed Question: SRO 3 Rev: 0

Given:

- A LOCA has occurred
- The crew is performing the actions of OP-902-002, LOCA Recovery Procedure
- Pressurizer level indicates 10% and rising
- Steam Generator #1 level is 51% NR and rising
- Steam Generator #2 level is 50% NR and rising
- Both Steam Generator levels are being controlled with EFW in manual
- CET temperature is 430 degrees F
- RCS pressure is 524 psia
- QSPDS Reactor Vessel level 5 indicates NOT VOIDED
- Containment temperature is 210 degrees F

Which of the following must the CRS direct prior to the crew throttling HPSI flow?

A. Raise vessel plenum level until Vessel Plenum level indicates 100%.

B. Raise Steam Generator levels to greater than 55% NR.

C. Raise subcooling to ≥ 28° F.

D. Raise pressurizer level to greater than 23%.

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: Vessel level is satisfied when level 5 indicates not voided or plenum level is greater than 80%.The plenum is not required to be full.
B. Incorrect: Steam Generator levels must be maintained or being restored to 55% NR. The operator has control of steam generator level in manual and level is capable of being restored.
C. Incorrect: The RCS is 42 degrees subcooled, based on CET temperature and pressurizer pressure which meets the requirement of 28 degrees subcooled.
D. **CORRECT:** Pressurizer level must be greater than 7% to meet HPSI throttle criteria. However, harsh environment conditions exist in containment due to containment temperature >200 degrees F. Therefore minimum pressurizer level is 23% and not meeting HPSI throttle criteria. The guidance for harsh conditions in containment is located in the administrative procedure OI-038-000, EOP Operations Expectations/Guidance procedure and must be applied to this question.

Technical Reference(s): OP-902-002 step 23 revision 18
(Attach if not previously provided) OI-038-000 revision 7
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE02 obj. 19 (As available)
WLP-OPS-PPE01 obj. 4

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41
55.43 5

INSTRUCTIONS | CONTINGENCY ACTIONS

HPSI Throttle Criteria

* 23. **IF** HPSI pumps are operating, **AND ALL** of the following conditions are satisfied:

- RCS subcooling is greater than or equal to 28ºF
- Pressurizer level is greater than 7% **[23%]** and controlled
- Verify **ALL** steam generators capable of steaming are being maintained or restored to within the following level:
    - 55% to 70% NR **[60-80% NR]** using MFW or EFW in auto or manual
- RVLMS indicates level higher than Hot Leg by at least one of the following:
    - QSPDS REACTOR VESSEL LEVEL 5 **NOT** voided
    - VESSEL LEVEL PLENUM greater than or equal to 80%

THEN throttle HPSI flow or stop **ONE** HPSI pump at a time.

5.4.29 HPSI Pump Minimum Flow Criteria

- None

5.4.30 HPSI Pump Restart Criteria

- If any throttle criteria is degrading rapidly, the HPSI pump may be started or valves may be opened in anticipation of exceeding the criteria.

5.4.31 HPSI Throttle Criteria

- If RCS pressure is < 1000 psia, use other parameters in addition to the subcooled margin indicators to verify subcooling.
- Water solid operations of the pressurizer should be avoided unless minimum subcooling (28°F) cannot be maintained in the RCS. If the RCS is solid, closely monitor any makeup or draining and any system heatup or cooldown, to avoid any unfavorable rapid pressure excursions.
- If Emergency Boration is in progress to meet Reactivity Control Safety Function, at least one charging pump must remain in operation unless this creates a challenge to the pressurizer safety valves.
- When a void exists in the reactor vessel and RCPs are operating, it is not possible to obtain an accurate reactor vessel liquid level indication due to the effect of the RCP induced pressure head on the RVLMS. However, inventory trending may still be discerned.
- If the operator chooses to throttle SI flow control valves and the HPSI pumps remain running, consideration should be given to the amount of time the HPSI pumps are operated on recirc.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000026 AA2.03	
	Importance Rating		2.9

K/A Statement

AA2.03 - Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition

Proposed Question: SRO 4 Rev: 0

Given:

- Plant is operating at 100% with the following Annunciators in alarm:
  - o CCW Surge Tank Level LO-LO on Panel M & N
  - o CCW FLOW LO red alarms for all Reactor Coolant Pumps
  - o Controlled Bleedoff Temperature Hi alarms for all Reactor Coolant Pumps
- CCW Surge Tank A level is 0%.
- CCW Surge Tank B level lowered to 15% but is being restored via the CCW Makeup Pumps

The crew has entered OP-901-510, CCW System Malfunction. To mitigate this event, the CRS will complete subsection ____(1)____. The crew will be required to _____(2)_____.

	(1)	(2)
A.	E_3 High System Temperature and transition to E_1 System Leakage	isolate the leak and then restore flow to the AB loop
B.	E_3 High System Temperature and transition to E_1 System Leakage	restore flow to the AB loop and then isolate the leak
C.	E_1 System Leakage only	restore flow to the AB loop and then isolate the leak
D.	E_1 System Leakage only	isolate the leak and then restore flow to the AB loop

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: Despite a RCP CBO high temperature annunciator locked in, System leakage is the appropriate subsection to enter. Based on the annunciators and surge tank levels, the AB loop has isolated and there is no flow to the RCPs. Step 1 of System Leakage has the crew to restore flow to the RCPs because they must be tripped within 3 minutes on the loss of CCW. The leak is isolated further into the procedure. E3 will provide no useful guidance for the CCW leak.

B. Incorrect: Despite a RCP CBO high temperature annunciators locked in, System leakage is the appropriate subsection to enter. Based on the annunciators surge tank levels, the AB loop has isolated and there is no flow to the RCPs. Step 1 of System Leakage has the crew to restore flow to the RCPs because they must be tripped within 3 minutes on the loss of CCW. The leak is isolated further into the procedure. E3 will provide no useful guidance for the CCW leak

C. **CORRECT**: Part 1 is correct. Based on the annunciators surge tank levels, the AB loop has isolated and there is no flow to the RCPs. Step 1 of System Leakage has the crew to restore flow to the RCPs because they must be tripped within 3 minutes. The leak is isolated further into the procedure.

D. Incorrect: Part 1 is correct. Based on the annunciators surge tank levels, the AB loop has isolated and there is no flow to the RCPs. Step 1 of System Leakage has the crew to restore flow to the RCPs because they must be tripped within 3 minutes. The leak is isolated further into the procedure.

Technical Reference(s): OP-901-510 revision 301
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO50 obj. 6 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

B$_2$ INDICATIONS

- CCW Surge Tank level dropping
- CCW System pressure abnormally low or dropping
- CCW System or component flows abnormal
- Amber trip/trouble light on affected CCW Pump control switch
- Dry Cooling Tower bypasses and isolates automatically
- CCW Safety Headers automatically split
- CCW AB Header automatically isolates
- CCW makeup valves open causing CCW Surge Tank to overflow
- CCW Makeup Pumps automatically starting
- Rising/high CCW temperature
- Rising/high temperature on components supplied by CCW
- Leakage indicated by local inspection

C AUTOMATIC ACTIONS

1. At 70% dropping CCW Surge Tank level, CMU-226, WATER STORAGE MAKEUP CCW SURGE TANK, Opens.

2. At 52% dropping CCW A(B) Surge Tank Level, CMU-538A(B), CCW MAKEUP VALVE A(B), Opens.

NOTE 3

1. WITH CCW Makeup Pumps aligned for Automatic Operation, the CCW Makeup Pumps will start on a low CCW Surge Tank Level (20%) AND automatically stop three minutes later OR on a high level (52%).
2. IF the CCW Makeup Pump(s) are automatically stopped by the three minute timer AND CCW Surge Tank Level is NOT ≥ 24%, THEN further Operation of the CCW Makeup Pump(s) will require placing the Control Switch to START.

3. At 20% dropping CCW A(B) Surge Tank level, the following actions occur:

• CMU-0004A(B)	CCW MAKEUP PUMP A(B)	Starts
• CC-134A(B)	CCW A(B) DRY COOLING TOWER BYPASS	Opens
• CC-135A(B)	CCW A(B) DRY COOLING TOWER ISOLATION	Closes

C AUTOMATIC ACTIONS (Continued)

NOTE 4

1. CCW Suct & Disch Header Tie Valves AB to A remain Open when CCW ASSIGNMENT Switch is aligned to A.
2. CCW Suct & Disch Header Tie Valves AB to B remain Open when CCW ASSIGNMENT Switch is aligned to B.

4. At 16% dropping CCW A(B) Surge Tank level, the following valves Close, to split the A AND B CCW trains:
   - CC-126A/CC-114A — CCW SUCT & DISCH HEADER TIE VALVES AB TO A
   - CC-127A/CC115A — CCW SUCT & DISCH HEADER TIE VALVES AB TO A
   - CC-126B/CC-114B — CCW SUCT & DISCH HEADER TIE VALVES AB TO B
   - CC-127B/CC-115B — CCW SUCT & DISCH HEADER TIE VALVES AB TO B
   - CC-200A/CC-727 — CCW SUCT & DISCH HEADER TIE VALVES A TO AB
   - CC-200B/CC-563 — CCW SUCT & DISCH HEADER TIE VALVES B TO AB

5. IF loss of CCW Pump(s) occurs due to loss of Vital Bus, THEN applicable pump will start 7 seconds after bus is energized by the Emergency Diesel Generator.

6. WHEN CCW temperature ≥ 92°F, THEN CCW Dry Cooling Tower Fans in Auto start sequentially in Slow speed at 60 second intervals.

7. WHEN ALL CCW Dry Cooling Tower Fans in AUTO are operating in Slow speed AND CCW temperature ≥ 92°F, THEN CCW Dry Cooling Tower Fans shift sequentially to Fast speed at 60 second intervals.

8. WHEN CCW temperature > 100°F, then applicable Auxiliary Component Cooling Water Pump Starts AND ALL CCW Dry Cooling Tower Fans in Auto shift to Fast speed.

C AUTOMATIC ACTIONS (Continued)

9. WHEN CCW temperature > 102°F, THEN Essential Chillers shift from Dry Tower mode to Wet Tower mode.

10. High Reactor Coolant Pump CCW Return temperature (155°F) actuates automatic isolation of applicable RCP Seal Cooler by Closing the following valves:

- CC-679A/CC-6651A 1A RCP Seal Cooler
- CC-679B/CC-6651B 1B RCP Seal Cooler
- CC-680A/CC-666A 2A RCP Seal Cooler
- CC-680B/CC-666B 2B RCP Seal Cooler

END

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

PLACEKEEPER
START DONE

1. IF ANY of the following occur, THEN GO TO Subsection E_1, System Leakage: ☐ ☐
   - CCW Surge Tank level dropping ☐
   - CCW Dry Cooling Towers isolated due to low CCW Surge Tank level ☐
   - CMU-226, WATER STORAGE MAKEUP CCW SURGE TANK, cycling frequently ☐
   - CCW header isolates due to low CCW Surge Tank level ☐
   - Local observation of CCW leak reported to Control Room ☐

2. IF ANY of the following occur, THEN GO TO Subsection E_2, Loss of CCW Pump(s): ☐ ☐
   - CCW system OR component flows low ☐
   - Amber trip/trouble light on CCW PUMP A(B)(AB) Control Switch ☐

3. IF EITHER of the following indications occur, THEN GO TO Subsection E_3, High System Temperature: ☐ ☐
   - CCW temperature rising/high ☐
   - High temperatures on components cooled by CCW ☐

E_1 SYSTEM LEAKAGE

CAUTION 1

IF COMPONENT COOLING WATER IS LOST TO REACTOR COOLANT PUMP SEAL(S) FOR > 10 MINUTES, THEN RESTORING COMPONENT COOLING WATER TO REACTOR COOLANT PUMP(S) MAY RESULT IN SEAL FAILURE.

	PLACEKEEPER START	PLACEKEEPER DONE
1. IF CCW flow is lost to AB Header AND Surge Tank level is restored on EITHER side, THEN align CCW flow to AB Header by Opening the valves associated with the restored side:	☐	☐
1.1 Train A:		☐
• CC-200A/CC-727 CCW SUCT & DISCH HEADER TIE VALVES A TO AB		
OR		
1.2 Train B:		☐
• CC-200B/CC-563 CCW SUCT & DISCH HEADER TIE VALVES B TO AB		
1.3 Restore CCW flow to the SFPHX by opening CC-620, Fuel Pool Heat Exch's Temperature Control	☐	☐

	PLACEKEEPER START	PLACEKEEPER DONE
2. IF CCW flow is lost to the AB Header AND CAN NOT be restored within 3 minutes, THEN perform the following:	☐	☐
2.1 Trip the Reactor.		☐
2.2 Stop ALL Reactor Coolant Pumps.		☐
2.3 Implement OP-902-000, STANDARD POST TRIP ACTIONS, concurrently with this procedure.		☐
2.4 IF RCS temperature > 130°F, THEN perform the following:	☐	☐
• Close CVC-101, LETDOWN STOP VALVE		☐
• Operate Charging Pumps as necessary to maintain Pressurizer level in accordance with OP-902-000, STANDARD POST TRIP ACTIONS	☐	Cont

PLACEKEEPER
START DONE

2.5 Close the following valves:

- PSL-105 RCS SAMPLE ISOL HOT LEG (IN)
- PSL-203 RCS SAMPLE ISOL PZR SURGE (IN)
- PSL-303 RCS SAMPLE ISOL PZR ISOL VLV (IN)
- SSL-8004A SAMPLING ISOLATION SG 1
- SSL-8004B SAMPLING ISOLATION SG 2
- SSL-301B SAMPLING ISOLATION MAIN STM LINE 2
- PSL-107 RCS SAMPLE ISOL HOT LEG (OUT)
- PSL-204 RCS SAMPLE ISOL PZR SURGE (OUT)
- PSL-304 RCS SAMPLE ISOL PZR STEAM (OUT)
- SSL-8006A SAMPLING ISOLATION SG 1
- SSL-8006B SAMPLING ISOLATION SG 2
- SSL-301A SAMPLING ISOLATION MAIN STM LINE 1

2.6 Perform OP-901-513, SPENT FUEL POOL COOLING MALFUNCTION, concurrently with this procedure.
[INPO IER 11-2 Recommendation 4]

2.7 Locally Secure Operating Boric Acid Concentrators in accordance with OP-007-001, BORON MANAGEMENT SYSTEM.

2.8 Locally place BOTH Waste Gas Compressor Control Switches to Off.

NOTE 3

1. Manual Override of RCP Seal Cooler Isolation Valves is accomplished by positioning associated Control Switch to CLOSE, THEN to OPEN.
2. IF after 100 seconds CCW Return temperature is NOT $<$ 145°F, THEN the applicable RCP Seal Cooler(s) will Isolate.

	PLACEKEEPER START	DONE
3. IF CCW to a Reactor Coolant Pump Seal Cooler(s) Isolate, THEN within 3 minutes restore Component Cooling Water flow to applicable Reactor Coolant Pump(s) by opening the applicable RCP CCW Isolation Valve(s):	☐	☐
• CC-679A/CC-6651A RCP 1A SEAL COOLER		☐
• CC-679B/CC-6651B RCP 1B SEAL COOLER		☐
• CC-680A/CC-666A RCP 2A SEAL COOLER		☐
• CC-680B/CC-666B RCP 2B SEAL COOLER		☐
4. IF CCW is lost to in-service Shutdown Cooling train, THEN implement OP-901-131, SHUTDOWN COOLING MALFUNCTION, AND perform concurrently with this procedure.	☐	☐
5. IF CCW Surge Tank level is dropping, THEN verify the Automatic Actions occur as necessary.	☐	☐
6. Make the following plant page TWICE:	☐	☐

"ATTENTION STATION PERSONNEL. ATTENTION STATION PERSONNEL. A COMPONENT COOLING WATER LEAK IS IN PROGRESS. ANYONE DISCOVERING A SIGNIFICANT WATER LEAK CONTACT THE CONTROL ROOM AT 3100."

CAUTION 7

1. IN MODES 1 – 3, A CSP LEVEL OF ≥ 92% IS REQUIRED TO COMPLY WITH TECHNICAL SPECIFICATION 3.7.1.3. **[EC-I91-003, ER-W3-98-0876]**
2. IN MODE 4, A CSP LEVEL OF ≥ 11% IS REQUIRED TO COMPLY WITH TECHNICAL SPECIFCIATION 3.7.1.3.
3. WITH THE CSP INOPERABLE, RESTORE THE CSP TO OPERABLE STATUS WITHIN 4 HOURS, IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.7.1.3.

		PLACEKEEPER START	DONE
7.	Attempt to maintain CCW Surge Tank level ≥ 52% by ANY combination of the following, IF necessary:	☐	☐
	• Operate CMU-538A, COMPONENT COOLING WATER MAKEUP VALVE A	☐	Cont
	• Operate CMU-0004A, COMPONENT COOLING WATER MAKEUP PUMP A	☐	Cont
	• Operate CMU-538B, COMPONENT COOLING WATER MAKEUP VALVE B	☐	Cont
	• Operate CMU-0004B, COMPONENT COOLING WATER MAKEUP PUMP B	☐	Cont
	• Operate CMU-226, WATER STORAGE MAKEUP CCW SURGE TANK.	☐	Cont
8.	IF CCW Surge Tank level is being maintained ≥ 52% by automatic makeup, THEN attempt to locate AND isolate leak.	☐	☐
9.	IF ALL the following conditions are indicated, THEN trip the affected Component Cooling Water Pump:	☐	☐
	• CCW Loops are split		☐
	• CCW Surge Tank level lost		☐
	• CCW Pump cavitation		☐

	PLACEKEEPER START	DONE
10. IF CCW Dry Cooling Tower has Isolated, THEN verify associated ACCW Pump is Operating.	☐	☐

NOTE 11

IF the NNS Loop is Isolated AND CCW Surge Tank Level returns to normal AND makeup is no longer required, THEN the leak is in the NNS Header.

	START	DONE
11. Determine if leak is in the NNS Loop by performing the following:	☐	☐
11.1 Close the following valves:		
• CC-501 NNS LOOP SUPPLY ISOL		☐
• CC-562 NNS LOOP RETURN ISOL		☐
11.2 IF CCW Surge Tank level continues to cycle OR Makeup System is operating to maintain level, THEN Open the following valves:	☐	☐
• CC-501 NNS LOOP SUPPLY ISOL		☐
• CC-562 NNS LOOP RETURN ISOL		☐

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000038 2.2.38	
	Importance Rating		4.5

K/A Statement

G2.2.38 - Knowledge of conditions and limitations in the facility license.

Proposed Question: SRO 5 Rev: 0

Given:

- Plant at 100% power and steady state
- N-16 Radiation Monitor for Steam Generator 2 reads 65 GPD
- PSLR reads 85 GPD
- Chemistry reports primary to secondary leakage in Steam Generator 2 is consistent with PSLR

Technical Specification 3.4.5.2, Operational Leakage, is _____(1)______.

The Nuclear Energy Institute recommended a limit of 150 GPD for primary to secondary leakage; however, the limit in Technical Specification 3.4.5.2 is more restrictive ____(2)____.

	(1)	(2)
A.	not met	to ensure the magnitude of leakage does not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems
B.	met	to ensure the magnitude of leakage does not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems
C.	not met	because the proximity of the east ADV to the east control room air intake could result in unacceptable radiological consequences
D.	met	because the proximity of the east ADV to the east control room air intake could result in unacceptable radiological consequences

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: Part 1 is correct - T.S. 3.4.5.2 Action a. With any PRESSURE BOUNDARY LEAKAGE, or primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours. T.S. LCO 3.4.5.2 LCO c requires entry for 75 gallons per day primary to secondary leakage through any one steam generator (SG), which 85 GPD exceeds. The description in Part 2 is the bases for IDENTIFIED LEAKAGE limits, not primary to secondary leakage.

B. Incorrect: Primary to secondary leakage is specifically excluded from TS 3.4.5.2, Action b. The description in Part 2 is the bases for IDENTIFIED LEAKAGE limits, not primary to secondary leakage.

C. **CORRECT**: T.S. 3.4.5.2 Action a. is correct based on given leakage greater than 75 gallons per day primary to secondary leakage through SG #2. The bases is correct based on not exceeding 5 REM TEDE per person for the duration of the accident.

D. Incorrect: Primary to secondary leakage is specifically excluded from TS 3.4.5.2, Action b. The bases is correct based on not exceeding 5 REM TEDE per person for the duration of the accident.

Technical Reference(s): Technical Specification 3.4.5.2, Amendment 204
(Attach if not previously provided) and Basis
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-RCS00 Obj. 9 (As available)
WLP-OPS-PPO20 Obj. 5

Question Source: Bank # X Question #8
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2012 SRO NRC Exam

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

10 CFR Part 55 Content: 55.41 ____
55.43 2

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 75 gallons per day primary to secondary leakage through any one steam generator (SG),

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE, primary to secondary leakage, and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

NOTE: Not required to be performed until 12 hours after establishment of steady state operation.

4.4.5.2.1 Reactor Coolant System leakages, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

4.4.5.2.2 Primary to secondary leakage shall be verified to be $\leq$ 75 gallons per day through any one SG at least once per 72 hours.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.2.3 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1, Section A and Section B, shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve,

d. Following valve actuation for valves in Section B due to automatic or manual action or flow through the valve:

    1. Within 24 hours by verifying valve closure, and

    2. Within 31 days by verifying leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

4.4.5.2.4 Each Reactor Coolant System pressure isolation valve power-operated valve specified in Table 3.4-1, Section C, shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SECTION A

SI-329A	SIT Check
SI-329B	"
SI-330A	"
SI-330B	"
SI-336A	Cold Leg Injection Check
SI-336B	"
SI-335A	"
SI-335B	"
SI-510A	Hot Leg Injection Check
SI-512A	"
SI-510B	"
SI-512B	"
SI-241	HPSI Check
SI-242	"
SI-243	"
SI-244	"

SECTION B

SI-142A	LPSI Check
SI-142B	"
SI-143A	"
SI-143B	"

SECTION C POWER-OPERATED VALVES

SI-401A	SDC Suction Isolation
SI-401B	"
SI-405A	"
SI-405B	"
SI-4052A	SDC Suction Bypass Isolation
SI-4052B	SDC Suction Bypass Isolation

(a) Maximum Allowable Leakage (each valve):

1. SI-4052A(B) leakage limit is less than or equal to 0.375 gpm.

2. Except as noted below, leakage rates greater than 1.0 gpm are unacceptable.

3. For SI-401A(B) and SI-405A(B), leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. For SI-401A(B) and SI-405A(B), leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

5. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 200 psid.

3/4.4.5.2 OPERATIONAL LEAKAGE

> (EC-3173 Ch. 53)

MODE in the Applicability of the associated LCO if any of the following conditions are satisfied: (1) the SR has been performed within the surveillance interval (i.e. it is current) and is known not to be failed or (2) the SR is required to be met, but not performed, in the MODE to be entered and is known not to be failed. The initial surveillance performance will be completed within 12 hours once the plant is at stable operating pressure following the establishment of steady state conditions. Other instruments such as those contained in TS 3/4.4.5.1 can be utilized to determine whether RCS operational leakage limits are being exceeded prior to initial performance.

Once the plant establishes steady state operation, 12 hours is allowed for completing the SR. If the SR was not performed within this 12 hour interval, there would then be a failure to perform the SR within the specified interval, and the provisions of 4.0.3 would apply. Should the 72 hour interval be exceeded while steady state operation has not been established, this NOTE allows 12 hours after steady state operation has been established to perform the SR. The SR is still considered to be performed within the surveillance interval. Therefore, if the Surveillance was not performed within the 72 hour (plus the extension allowed by 4.0.2) interval, but steady state operation was not established, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of 4.0.4 occurs when changing MODES, even with the 72 hour surveillance interval not met, provided operation does not exceed 12 hours with the establishment of steady state operation.

< (EC-3173 Ch. 53)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

> (DRN 04-1243, Ch. 38; 06-916, Ch. 48)

The primary to secondary leakage limit of 75 gallons per day through any one SG is based on the operational leakage performance criterion in NEI 97-06. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, “The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day.” The NEI 97-06 limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion (since it is less than 150 gpd through any one SG) in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

< (DRN 04-1243, Ch. 38; 06-916, Ch. 48)

OPERATIONAL LEAKAGE (Continued)

>(DRN 04-1243, Ch. 38)

Steam generator tube cracks having primary-to-secondary leakage less than 150 gpd per steam generator during operation will have an acceptable margin of safety to withstand loads imposed during normal operation and postulated accidents (Reference NEI 97-06). Due to the proximity of the east atmospheric dump valve to the east control room intake, the primary-to-secondary leakage limit required to achieve acceptable radiological consequences, for accidents that rely on reactor coolant system cooldown using the steam generators, is limiting. Therefore, 75 gpd per steam generator is imposed as the primary-to-secondary operational leakage limit.

<(DRN 04-1243, Ch. 38)

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

>(LBDCR 13-003, Ch. 74)

3/4.4.6 DELETED

<(LBDCR 13-003, Ch. 74)

3/4.4.7 SPECIFIC ACTIVITY

>(DRN 03-173, Ch. 18; 05-131, Ch. 39)

The Code of Federal Regulations, 10 CFR 50.67 specifies the maximum total effective dose equivalent an individual offsite can receive during a design basis accident. The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The specific activity limits ensure that these doses are held within the appropriate 10 CFR 50.67 requirements (small fraction, well within, or within) during analyzed transients and accidents.

<(DRN 05-131, Ch. 39)

Operation with iodine specific activity levels greater than the LCO limit is permissible for up to 48 hours, provided the activity levels do not exceed 60 uCi/gm. A 48 hour limit was established because of the low probability of an accident occurring during this period. The dose consequences of an accident during this 48 hour period would not exceed the full 10 CFR 50.67 limits.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

<(DRN 03-173, Ch. 18)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000077 AA2.05	
	Importance Rating		3.8

K/A Statement

AA2.08 - Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operational status of offsite circuit

Proposed Question: SRO 6 Rev: 0

Given:

- Plant is in Mode 1
- The Shift Manager receives a call from the grid operator informing him that Waterford's post-trip grid voltage would be 222 kV, which is below the required value of TRM 3.8.1.1 for post-trip emergency loads
- Current Line A and B grid voltages do **NOT** meet the requirements of OP-903-066, Electrical Breaker Alignment Check

Based on this information, the crew should ____(1)____ because ____(2)_____.

	(1)	(2)
A.	declare both AC off-site circuits inoperable and enter both Tech Spec 3.8.1.1 and TRM 3.8.1.1	sufficient power may not exist to supply safety related equipment needed for the mitigation of a design basis accident
B.	declare both AC off-site circuits inoperable and enter both Tech Spec 3.8.1.1 and TRM 3.8.1.1	post trip excessive current from this reduced voltage will actuate the SUT protective relays
C.	enter only TRM 3.8.1.1	sufficient power may not exist to supply safety related equipment needed for the mitigation of a design basis accident
D.	enter only TRM 3.8.1.1	post trip excessive current from this reduced voltage will actuate the SUT protective relays

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: TS entry is required due to offsite circuit not meeting the requirements of OP-903-066. In this instance the offsite circuit is inoperable per Technical Specifications. The basis is correct.

B. INCORRECT: TS entry is required due to offsite circuit not meeting the requirements of OP-903-066. In this instance the offsite circuit is inoperable per Technical Specifications. The basis is incorrect because this is not the reason for entering the spec but is plausible because lower voltage raises current.

C. INCORRECT: TS entry is required due to offsite circuit not meeting the requirements of OP-903-066. In this instance the offsite circuit is inoperable per Technical Specifications. The basis is correct.

D. INCORRECT: TS entry is required due to offsite circuit not meeting the requirements of OP-903-066. In this instance the offsite circuit is inoperable per Technical Specifications. The basis is incorrect because this is not the reason for entering the spec but is plausible because lower voltage raises current.

Technical Reference(s): TRM (pg 4h) and Tech Spec 3.8.1.1 & Bases
OP-901-314 E_1 Step 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO30 Obj: 5 (As available)

Question Source: Bank # ______ Question #6
Modified Bank # X (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2009 NRC SRO Exam

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

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments: Changes made include stem altered to indicate that the voltages do not meet OP-903-066 requirements which makes A correct. Changed part 2 of all distractors. Part 2 of A and C are now the same and a new reason was given for the basis for entering the specs in choices B and D

→(DRN 05-1013, Am. 103)

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 Offsite A.C. circuit voltage, as measured at the Waterford 3 Switchyard, shall be such that predicted post-trip offsite A.C. circuit voltage, as determined by the Grid Operator, will be sufficient to supply post-trip emergency loads.

APPLICABILITY: MODE 1

ACTIONS:

a. When notified by the Grid Operator that predicted post-trip offsite A.C. circuit voltage will be less than 223kV, initiate a Condition Report and complete an offsite A.C. circuit operability determination within 12 hours.

SURVEILLANCE REQUIREMENTS:

None

←(DRN 05-1013, Am. 103)

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. Diesel oil feed tanks containing a minimum volume of 339 gallons of fuel, and

2. A separate diesel generator fuel oil storage tank, and

3. A separate fuel transfer pump.

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

ACTION:

a. With one offsite circuit of 3.8.1.1a inoperable, demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter. Restore the offsite A.C. circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1b inoperable:

(1) Demonstrate the OPERABILITY of the remaining A.C. circuits by performing Surveillance Requirements 4.8.1.1.1a (separately for each offsite A.C. circuit) within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator (unless it has been successfully tested in the last 24 hours) by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated.

(2) Restore the diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, unless the following condition exists:

ACTION: (Continued)

(a) The requirement for restoration to OPERABLE status within 72 hours may be extended to 10 days if a temporary emergency diesel generator is verified available, and

(b) If at any time the temporary emergency diesel generator availability cannot be met, either restore the temporary emergency diesel generator to available status within 72 hours (not to exceed 10 days from the time the permanent plant EDG originally became inoperable), or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one offsite A.C. circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; and, if the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours (unless it is already operating) unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite A.C. circuit or diesel generator) to OPERABLE status in accordance with the provisions of ACTION statement a or b, as appropriate, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2a.4 performed under this ACTION statement satisfies the diesel generator test requirement of ACTION statement a or b.

d. With one diesel generator inoperable, in addition to ACTION b. or c. above, verify that:

(1) All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

(2) When in MODE 1, 2, or 3, the steam-driven emergency feed pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

e. With two of the above required offsite A.C. circuits inoperable, restore one of the inoperable offsite A.C. circuits to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite A.C. circuit, follow ACTION statement a with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit. A successful test of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2a.4 performed under this ACTION statement satisfies the diesel generator test requirement of ACTION statement a.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator, follow ACTION statement b with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable diesel generator.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months by transferring manually and automatically unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE*:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the diesel oil feed tank,

2. Deleted,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the diesel oil feed tank,

4. Verifying the diesel starts**. The generator voltage and frequency shall be at least 3920 volts and 58.8 Hz in $\leq$ 10 seconds after the start signal. The steady state voltage and frequency shall be maintained at 4160 + 420, -240 volts and 60 $\pm$ 1.2 Hz. The diesel generator shall be started for this test by using one of the following signals:

    a) Manual.

    b) Simulated loss-of-offsite power by itself.

    c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

    d) An ESF actuation test signal by itself.

*All planned starts for the purpose of surveillance in this section may be preceded by a prelube period as recommended by the manufacturer.

**A modified diesel generator start involving idling and gradual acceleration to synchronous speed may be used for this surveillance requirement as recommended by the manufacturer. When modified start procedures are not used, the time, speed, voltage, and frequency tolerances of this surveillance requirement must be met.

5. Verifying the generator is synchronized, loaded to an indicated 4000-4400 Kw* in accordance with the manufacturer's recommendation and operates for at least an additional 60 minutes#, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the diesel oil feed tanks.

c. Deleted

*This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variation due to changing bus loads shall not invalidate the test.

#This surveillance requirement shall be preceded by and immediately follow without shutdown a successful performance of 4.8.1.1.2a.4 or 4.8.1.1.2d.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 184 days a diesel generator fast start test shall be performed in accordance with TS 4.8.1.1.2a.4. Performance of the 184 day fast start test satisfies the 31 day testing requirements specified in TS 4.8.1.1.2a.4.

e. At least once per 18 months by:

1. Verifying the generator capability to reject a load of greater than or equal to 498 kW while maintaining voltage at 4160 +420, -240 volts and frequency at 60 +4.5, -1.2 Hz.

2. Verifying the generator capability to reject a load of an indicated 4000-4400 kW without tripping. The generator voltage shall not exceed 5023 volts during and following the load rejection.

3. During shutdown, simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses and the permanently connected loads within 10 seconds after the auto-start signal, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 +420, -240 volts and 60 +1.2, -0.3 Hz during this test.

4. Verifying that on an SIAS actuation test signal (without loss-of-offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady-state generator voltage and frequency shall be 4160 +420, -240 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

5. During shutdown, simulating a loss-of-offsite power in conjunction with an SIAS actuation test signal, and

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses and the permanently connected loads within 10 seconds after the auto-start signal, energizes the auto-connected emergency loads through the load sequencer and operates for greater than or equal to 5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 +420, -240 volts and 60 +1.2, -0.3 Hz during this test.

   c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

6. Verifying the diesel generator operates for an interval of not less than 24 hours. During 2 hours of this test, the diesel generator shall be loaded to an indicated 4700 to 4900 Kw* and during 22 hours of this test, the diesel generator shall be loaded to an indicated 4000 to 4400 Kw.* The generator voltage and frequency shall be 4160 +420, -240 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be 4160 ± 420 volts and 60 +1.2, -0.3 Hz during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.a.4.**

7. During shutdown, verifying that the auto-connected loads and permanently connected loads to each diesel generator do not exceed the 2000-hour rating of 4400 kW.

*This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variation due to changing bus loads shall not invalidate the test.

**If Surveillance Requirement 4.8.1.1.2.a.4 is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at an indicated 4000-4400 kw* for 2 hours or until internal operating temperatures have stabilized. Within 5 minutes of securing the diesel generator, perform Surveillance Requirement 4.8.1.1.2.a.4.

8. During shutdown, verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

9. During shutdown, verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

10. Verifying that each fuel transfer pump transfers fuel to its associated diesel oil feed tank by taking suction from the opposite train fuel oil storage tank via the installed cross connect.

11. During shutdown, verifying that the automatic load sequence timer is OPERABLE with the time of each load block within ±10% of the sequenced load block time.

12. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

   a) turning gear engaged
   b) emergency stop
   c) loss of D.C. control power
   d) governor fuel oil linkage tripped

f. Deleted

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 600 rpm (60 ± 1.2 Hz) in less than or equal to 10 seconds.

h. Deleted

SURVEILLANCE REQUIREMENTS (Continued)

i. Deleted

4.8.1.1.3 Reports - (Not Used)

THIS PAGE NOT USED

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C SOURCES, and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion l7 of Appendix A to 10 CFR Part 50.

→(DRN 04-1243, Ch. 38; EC-1735, Ch. 55; EC-10725, Ch. 56)

←(DRN 04-1243, Ch. 38; EC-1735, Ch. 55; EC-10725, Ch. 56)

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. When one diesel generator is inoperable to perform either preplanned maintenance (both preventive and corrective) or unplanned corrective maintenance work, the allowed-outage-time (AOT) can be extended from 72 hours to 10 days, if a temporary emergency diesel generator (TEDG) is verified available and aligned for backup operation to the permanent plant EDG removed from service. The TEDG will be available prior to removing the permanent plant EDG from service for the extended preplanned maintenance work or prior to exceeding the 72-hour AOT for the extended unplanned corrective maintenance work. A Configuration Risk Management Program (CRMP) is implemented to assess risk of this activity when applying this ACTION. The TEDG availability is verified by: (1) starting the TEDG and verifying proper operation; (2) verifying 24 hour onsite fuel supply; and (3) ensuring the TEDG is aligned to supply power through a 4.16 kV non-safety bus to the 4.16kV safety bus. A status check for TEDG availability will also be performed at least once every 72 hours following the initial TEDG availability verification. The status check shall consists of: (1) verifying the TEDG equipment is mechanically and electrically ready for manual operation; (2) verifying 24 hour onsite fuel supply; and (3) ensuring the TEDG is aligned to supply power through a 4.16 kV non-safety bus to the 4.16 kV safety bus. If the TEDG becomes unavailable during the 10 day AOT and cannot be restored to available status, the EDG AOT reverts back to 72-hours. The 72 hours begins with the discovery of the TEDG unavailability, not to exceed a total of 10 days from the time the EDG originally became inoperable. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

>(DRN 03-375, Ch. 19)

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that(1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. With the minimum AC and DC power sources and associated distribution systems inoperable the ACTION requires the immediate suspension of various activities including operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SHUTDOWN MARGIN or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling concentration. This may result in an overall reduction in boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including increases when operating with a positive moderator temperature coefficient, must also be evaluated to ensure they do not result in a loss of required SHUTDOWN MARGIN. Suspension of these activities does not preclude completion of actions to establish a safe conservative condition.

<(DRN 03-375, Ch. 19)

>(EC-38571, Ch. 71)

The fuel handling accident (UFSAR Section 15.7.3.4) analysis assumes protection against load movements with or over irradiated fuel assemblies that could cause fuel assembly damage. Examples of load movements include movement of new fuel assemblies, irradiated fuel assemblies, and the dummy fuel assembly. The load movements do not include the movement over assemblies in a transfer cask using a single-failure-proof handling system. The load movements do not include the movement of the spent fuel machine or refuel machine without loads attached. It also does not include load movements in containment when the reactor vessel head or Upper Guide Structure is still installed. Load movements also exclude suspended loads weighing less than 1000 lbm (e.g. Westinghouse analysis CN-NFPE-09-57 describes no fuel failure for loads weighing less than 1000 lbm based upon the 2000 lbm analysis for drops distributed over two assemblies).

<(EC-38571, Ch. 71)

>(EC-10752, Ch. 56)

LCO 3.8.1.3

ACTION a

>(EC-15945, Ch. 61)

This ACTION ensures that each diesel generator fuel oil storage tank (FOST) contains fuel oil of a sufficient volume to operate each diesel generator for a period of 7 days. An administrative limit of greater than 40,033 gallons assures at least 39,300 usable gallons are stored in the tank accounting for volumetric shrink and instrumentation uncertainty. This useable volume is sufficient to operate the diesel generator for 7 days based on the time-dependent loads of the diesel generator following a loss of offsite power and a design bases accident and includes the capacity to power the engineered safety features in conformance with Regulatory

<(EC-10725, Ch. 56; EC-15945, Ch. 61)

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

>(EC-10725, Ch. 56; EC-15945, Ch. 61)

Guide 1.137 October 1979. The minimum onsite stored fuel oil is sufficient to operate the diesel generator for a period longer than the time to replenish the onsite supply from the outside sources discussed in FSAR 9.5.4.2.

An additional provision is included in the ACTION which allows the diesel generators to remain operable when their 7 day fuel oil supply is not available provided that at least a 6 day supply of fuel oil is available. This provision is acceptable on the basis that replacement fuel oil is onsite within the first 48 hours after falling below the 7 day supply. An administrative limit of greater than 37,696 gallons assures at least 37,000 usable gallons are stored in the tank,

<(EC-10725, Ch. 56; EC-15945, Ch. 61)

LCO 3.8.1.3 (Continued)

ACTION a (Continued)

accounting for volumetric shrink and instrumentation uncertainty. This useable volume is sufficient to operate the diesel generator for 5 days based on the full continuous load (4400kW) of the diesel generator and is sufficient to operable the diesel generator for greater than 6 days based on the time dependent loads of the diesel generator following a loss of offsite power and a design basis accident.

ACTION b

>(EC-15945, Ch. 61)

This ACTION is entered as a result of a failure to meet the acceptance criterion of particulate limits. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between surveillance frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7-day Completion Time allows for further evaluation, re-sampling, and re-analysis of the DG fuel oil.

<(EC-15945, Ch. 61)

ACTION c

With the new fuel oil properties defined in the Bases for SR 4.8.1.1.2.c not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a diesel generator start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the diesel generator would still be capable of performing its intended function.

ACTION d

>(EC-15945, Ch. 61)

This ACTION is entered as a result of the failure to meet any of the other ACTIONS.

<(EC-10725, Ch. 56; EC-15945, Ch. 61)

E1 RESPONSE TO DEGRADED GRID (CONT'D)

PLACEKEEPER

	START	DONE	N/A
5. Perform required notifications in accordance with OI-035-000, Notification Matrix.	☐	☐	

NOTE

The Gretna Transmission Operations Center (TOC) may be contacted using any of the following phone numbers or methods:

- Direct line (on the SM desk and on the NPO desk)
- 8-577-4454
- (504) 366-7237
- Fax line – (504) 374-4495

	START	DONE	N/A
6. Direct the POC to obtain the following from the Gretna TOC for aid in assessment of the degradation of the offsite power grid.	☐	Continuous ⬇	
• The Supervisory Control and Data Acquisition (SCADA) Activity Log		☐	
• Relay flag data for post-event review		☐	
• Applicable digital fault recorder and/or relay plots, as available		☐	
7. With the assistance of site Engineering and other site technical resources, evaluate the degradation of the offsite power grid and Operability based upon the following offsite electrical grid limits as listed in ENS-DC-199, Off Site Power Supply Design Requirements:	☐	☐	
• 230KV Bus Voltage Limits: 223.1 – 241.5 KV		☐	
• Accident loads 230 KV System: 71.33 MW and 34.21 MVAR		☐	
• MVAR Limits: -75 – +400 MVAR		☐	
• Frequency Limits: 58.5 – 61.8 Hz		☐	

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000028 AA2.02	
	Importance Rating		3.8

K/A Statement

AA2.02 – Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunction: PZR level as a function of power level or T-ave. including interpretation or malfunction

Proposed Question: SRO 7 Rev: 0

Given:

- Plant is stable at 100% power
- RCS Temperature loop1 Hot Leg (RC-ITI-0111X) has failed low
- The crew has entered OP-901-110, Pressurizer Level Control Malfunction

As a result of the failed temperature instrument, Letdown flow will ____(1)____. The CRS will implement subsection _____(2)_____ of OP-901-110.

	(1)	(2)
A.	rise	E1 Pressurizer Level Control Channel Malfunction
B.	lower	E1 Pressurizer Level Control Channel Malfunction
C.	rise	E2 Pressurizer Level Setpoint Malfunction
D.	lower	E2 Pressurizer Level Setpoint Malfunction

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: The first part is correct. The hot leg indicator is in effect a pressurizer level control channel, but the E1 subsection of OP-901-110 provides guidance for failure of the pressurizer level instruments. E2 of OP-901-110 provides the guidance for PZR level setpoint malfunctions.
B. Incorrect: The hot leg indicator that has failed low is an input to Tave that is fed to the Reactor Regulating System. Pressurizer level setpoint is a function of Tave. Therefore, the Pzr level setpoint drops causing letdown flow to rise. The hot leg indicator is in effect a pressurizer level control channel, but the E1 subsection of OP-901-110 provides guidance for failure of the pressurizer level instruments. E2 of OP-901-110 provides the guidance for PZR level setpoint malfunctions
C. **CORRECT**: The hot leg indicator that has failed low is an input to Tave that is fed to the Reactor Regulating System. Pressurizer level setpoint is a function of Tave. Therefore, the Pzr level setpoint drops causing letdown flow to rise. The applicant must recognize that the Thot instrument will affect pressurizer level setpoint and enter subsection E2.
D. Incorrect: The hot leg indicator that has failed low is an input to Tave that is fed to the Reactor Regulating System. Pressurizer level setpoint is a function of Tave. Therefore, the Pzr level setpoint drops causing letdown flow to rise. The second part is correct.

Technical Reference(s): OP-901-110 Revision 7 page 6 and 9
(Attach if not previously provided) SD-RR figure 2
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 obj. 1 (As available)
WLP-OPS-RR00 obj. 2

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

Comments:

C AUTOMATIC ACTIONS

1. IF Pressurizer level rises above program level, THEN the following should occur:

   1.1 Letdown flow rises to a maximum of 126 GPM.

   1.2 IF level rises 4.0■ above program level, THEN the following should occur.

   - Backup Charging Pumps receive backup stop signal
   - IF Pressurizer pressure ■2275 PSIA, THEN Backup Heaters energize

2. IF Pressurizer level drops below program level, THEN the following should occur:

   2.1 Letdown flow drops to a minimum of 28 GPM.

   2.2 IF level drops 2.5■ below program level, THEN first backup Charging Pump starts.

   2.3 IF level drops 3.9■ below program level, THEN second backup Charging Pump starts.

   2.4 IF level drops 6.0■ below program level, THEN BOTH backup Charging Pumps receive backup start signal.

   2.5 IF level drops to 28■, THEN ALL Pressurizer Heaters de-energize.

END

E SUBSEQUENT OPERATOR ACTIONS:

E_0 GENERAL

1. Stop Turbine load changes.

2. IF malfunction is due to failure of Letdown Flow Control valve, THEN GO TO OP-901-112, CHARGING/LETDOWN MALFUNCTION.

3. IF malfunction is due to failure of Pressurizer Level Control Channel (incorrect readings on EITHER RC-ILI-0110X OR RC-ILI-0110Y), THEN GO TO Subsection E_1, Pressurizer Level Control Channel Malfunction.

4. IF malfunction is due to failure of Pressurizer Level Setpoint (RC-ILIC-0110), THEN GO TO Subsection E_2, Pressurizer Level Setpoint Malfunction.

5. IF malfunction is due to failure of Pressurizer Level Controller (RC-ILIC-0110), THEN GO TO Subsection E_3, Pressurizer Level Controller Malfunction.

END

E_2 PRESSURIZER LEVEL SETPOINT MALFUNCTION (CONT'D)

NOTE

(1) Selecting the non-faulted channel may cause automatic actions to occur if actual level is not at program level.

(2) If a Tc or Th instrument fails, refer to OP-901-501, PMC or Core Operating Limit Supervisory System Malfunction, E2, Loss of COLSS (PMC Available).

9. Check Reactor Regulating System (RRS) Hot Leg indicators (RC-ITI-0111-X AND RC-ITI-0121-X) for abnormal readings AND perform the following:

9.1 IF RCS Temperature Loop 1 Hot Leg (RC-ITI-0111-X) indicates abnormally high OR low, THEN select LOOP 2 for TAVE LOOP SELECTOR on BOTH RRS local cabinets (CP-12A AND CP-12B).

9.2 IF RCS Temperature Loop 2 Hot Leg (RC-ITI-0121-X) indicates abnormally high OR low, THEN select LOOP 1 for TAVE LOOP SELECTOR on BOTH RRS local cabinets (CP-12A AND CP-12B).

10. Check Reactor Regulating System (RRS) Cold Leg indicators (RC-ITI-0111-Y AND RC-ITI-0121-Y) for abnormal readings AND perform the following:

10.1 IF RCS Temperature Loop 1 Cold Leg (RC-ITI-0111-Y) indicates abnormally high OR low, THEN perform EITHER of the following:

- Select presently non-selected position (ALT OR NORM) on TCOLD LOOP 1 selector switch located behind CP-2, Reactor Control (Preferred Method).

OR

- Select LOOP 2 on BOTH RRS local cabinets (CP-12A AND CP-12B)

10.2 IF RCS Temperature Loop 2 Cold Leg (RC-ITI-0121-Y) indicates abnormally high OR low, THEN perform EITHER of the following:

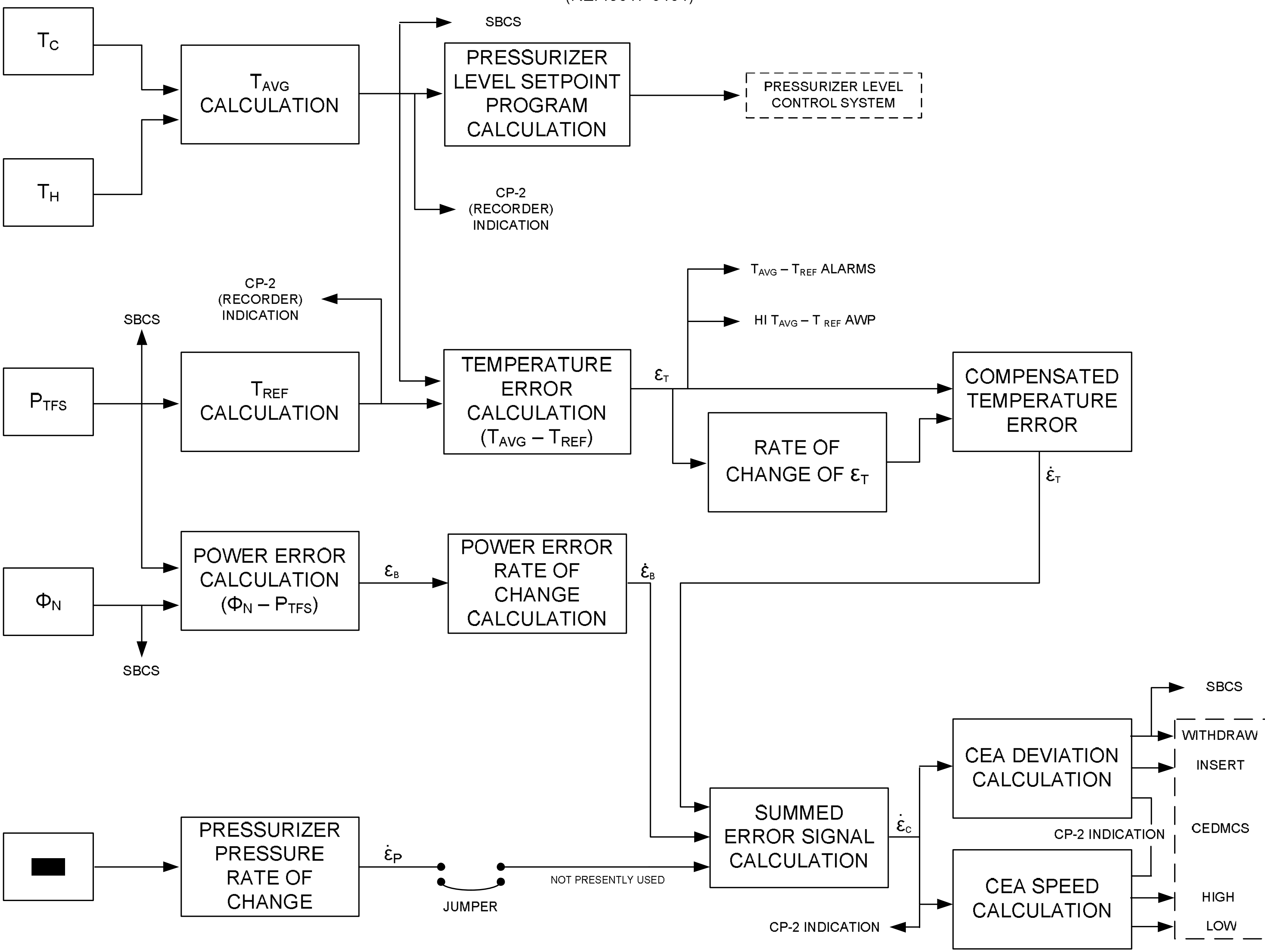
- Select presently non-selected position (ALT OR NORM) on TCOLD LOOP 2 selector switch located behind CP-2, Reactor Control (Preferred Method).

OR

- Select LOOP 1 on BOTH RRS local cabinets (CP-12A AND CP-12B).

FIG. 02 REACTOR REGULATING SYSTEM BASIC FUNCTIONAL DIAGRAM

(REF.5817-8101)



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000060 2.4.50	
	Importance Rating		4.0

K/A Statement

2.4.50 –Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: SRO 8 Rev: 0

Given:

- Plant is operating at 100%
- The crew is discharging Gas Decay Tanks
- The following annunciators is received:
    - Waste Gas Disch Rad High Dryer/Mon Trouble (Cabinet G, E-10)
- The ATC operator informs the CRS that the discharge activity exceeds the Gaseous Release Permit set point

The CRS will direct the crew to verify the ____(1)____ high alarm set point set in accordance with the release permit. The annunciator response procedure will refer the CRS to ____(2)_____ and close GWM-309, Waste Gas Discharge Flow Control.

	(1)	(2)
A.	Plant Stack PIG Monitor A <u>or</u> B (PRM-IRE-0100.1 <u>or</u> 0100.2)	OP-007-003, Gaseous Waste Management
B.	Plant Stack PIG Monitor A or B (PRM-IRE-0100.1 or 0100.2)	OP-901-413, Waste Gas Discharge High Radiation
C.	Gaseous Waste Management Monitor (PRM-IRE-0648)	OP-901-413, Waste Gas Discharge High Radiation
D.	Gaseous Waste Management Monitor (PRM-IRE-0648)	OP-007-003, Gaseous Waste Management

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: Plant Stack PIG Monitors are listed as an indication of Waste Gas Discharge Activity high, but is not the rad monitor adjusted for the release permit. OP-500-007 Att. 4.50 refers the crew to OP-901-413. Plausible because OP-007-003 has limitations on flow and activity.
B. Incorrect: Plant Stack PIG Monitors are listed as an indication of Waste Gas Discharge Activity high, but is not the rad monitor adjusted per the release permit. OP-500-007 Att. 4.50 refers the crew to OP-901-413.
C. **CORRECT:** Gaseous Waste Management Monitor is the rad monitor that is set in accordance with the release set point. OP-500-007 Att. 4.50 refers the crew to OP-901-413. The annunciator response procedure also directs verification of the correct setpoint on PRM-IRE-0648.
D. Incorrect: Gaseous Waste Management Monitor is the rad monitor that is set in accordance with the release set point. OP-500-007 Att. 4.50 refers the crew to OP-901-413. Plausible because OP-007-003 has limitations on flow and activity.

Technical Reference(s): OP-500-007 Attachment 4.50 revision 16
(Attach if not previously provided) OP-901-413
(including version/revision number) OP-007-003 caution before step 6.4.9 rev. 304

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PP040 obj. 1, 3 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ______
55.43 4

Comments:

CAUTION

1. IF DISCHARGE ACTIVITY EXCEEDS THE GASEOUS RELEASE PERMIT SETPOINT THEN OP-901-413, WASTE GAS DISCHARGE HIGH RADIATION, SHALL BE ENTERED.
2. AT LEAST **ONE** RAB EXHAUST FAN SHALL BE OPERATING WHILE DISCHARGING.
3. THE RELEASE SHOULD BE TERMINATED IF METEOROLOGICAL CONDITIONS ARE OUTSIDE THE PERMISSIBLE LIMITS.
4. THE RELEASE SHALL BE TERMINATED IF THE IN-SERVICE GDT PRESSURE BEGINS TO DECREASE UNTIL ADDITIONS TO THE GAS SURGE HEADER HAVE BEEN SECURED. IF NO NEW GASES HAVE BEEN INTRODUCED AND A WR HAS BEEN WRITTEN ON THE LEAKING INLET VALVE, THEN DISCHARGING CAN CONTINUE.
5. IF DISCHARGING **ALL THREE** GDTs SIMULTANEOUSLY, THEN ADDITIONS TO THE GAS SURGE TANK FROM THE VCT, GAS SURGE HEADER OR THE CONTAINMENT VENT HEADER SHOULD **NOT** BE MADE DURING GDT RELEASES DUE TO POTENTIAL LEAKAGE OF GDT INLET VALVES. DISCHARGING ALL GDTS SIMULTANEOUSLY IS THE PREFERRED METHOD. **[CR-98-1291]**

6.4.9 If the Gaseous Waste Discharge Radiation Monitor, PRM-IRE-0648, is operable and sample flow has risen to >2 scfm as seen locally and documented on Attachment 11.4, then continue to throttle Open Waste Gas Discharge GWM-IFIT-0648 Outlet Isolation Valve, GWM-311, to establish desired flow within limit indicated on Gaseous Release Permit.

6.4.10 If Gaseous Waste Discharge Radiation Monitor, PRM-IRE-0648, is not operable, then commence discharging by Throttling Open Waste Gas Discharge GWM-IFIT-0648 Outlet Isolation Valve, GWM-311, to establish the desired flow rate within the limit indicated on the Gaseous Release Permit.

6.4.11 Record the 0-hour data readings on the Gaseous Release Permit.

C AUTOMATIC ACTIONS

1. Waste Gas Discharge Flow Control GWM-309 closes to terminate gaseous waste release.

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

PLACEKEEPER
START DONE

1. Verify Waste Gas Discharge Flow Control, GWM-309 Closed. ☐
2. Verify alarm valid by checking Waste Gas Discharge Radiation Monitor chart recorder GWM-IFRR-0648. ☐
3. Isolate Gas Decay Tank being discharged by placing Charge/Release Control Switch on LCP-42 to OFF and verify outlet valve closes. ☐ ☐
4. Check Plant Stack Radiation Monitors PRM-IUR-0100.1AS and PRM-IUR-0100.2BS to determine length and severity of discharge. ☐ ☐
5. Notify Radiation Protection of release problem. ☐
6. Request Chemistry Department to sample tank being discharged for isotopic analysis. ☐ ☐
7. Verify Gaseous Waste Management Radiation Monitor high alarm setpoint (CP-6) was set in accordance with Release Permit. ☐ ☐
8. Verify Waste Gas Discharge Flow Rate Indication GWM-IFRR-0648 was less than or equal to maximum flow rate specified by release permit. ☐ ☐
9. Verify no inadvertent Decay Tank release occurred: ☐ ☐
   - 9.1 Check that Gas Decay Tanks not being discharged do not show pressure decrease. ☐
   - 9.2 Verify Gaseous Waste Management System valve alignment was correct for Decay Tank intended to be discharged. ☐ ☐
10. Verify Gas Surge Tank To Plant Vent Isolation Valve, GWM-304, Locked Closed. ☐
11. Advise Shift Manager to refer to EP-001-001, Recognition And Classification Of Emergency Conditions. ☐

4.50 WASTE GAS DISCH RAD HIGH DRYER/MON TRBL (E-10) REV 16

INITIATING DEVICE	SETPOINT
PRMIRE0648 K1 Relay (Fail Alarm)	De-energized
PRMIRE0648 K2 Relay (High Alarm)	Variable
GWMIPS0648, GWM Rad Monitor Dryer Outlet Pressure	6 PSIG (R: 3 PSIG)

POSSIBLE EFFECTS

1. Termination of Gas Decay Tank Discharge (GWM-309 closes).

CONTROL ROOM INDICATIONS

- RM-11, Grid EFL, GWM RAB -4, Gaseous Waste System Noble Gas Monitor (PRMIRE0648)
- GWMIFRR0648, Waste Gas Flow and Rad Recorder
- Waste Gas Discharge Flow Control Valve, GWM-309, position.
- PMC Point ID A41300, Gas to Stack Flow

LOCAL INDICATIONS

- PRMIRE0648, GWM Rad Monitor local indication
- GWMIFI6712, Waste Gas Decay Tanks to Plant Vent Flow Indicator, at LCP-42A
- Waste Gas Discharge Flow Control Valve, GWM-309, local indication
- Power Supplies:
  - ▪ PRMEBKR014AB 10, Waste Gas Discharge Rad Detector
  - LWMEBKR45AB 6, Waste Mgmnt Disch Flow and Rad Recorder

POSSIBLE CAUSES

1. HI activity

RECOMMENDED ACTIONS

1.1 Refer to OP-901-413, Waste Gas Discharge High Radiation.

1.2 Verify discharge auto isolates.

1.3 Direct Chemistry to sample GDT to determine validity of activity levels.

1.4 Verify correct setpoint in PRMIRE0648, Waste Gas Rad Monitor.

WASTE GAS DISCH RAD HIGH DRYER/MON TRBL

POSSIBLE CAUSES	RECOMMENDED ACTIONS
2. Waste Gas Rad Monitor failure.	2.1 Discharge GDT in accordance with OP-007-003, Gaseous Waste Management, with radiation monitor out of service.
3. GWM Rad Monitor Dryer outlet press High.	3.1 Verify discharge lineup in accordance with OP-007-003, Gaseous Waste Management.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	CE/A13 2.4.35	
	Importance Rating		4.0

K/A Statement

2.4.35 - Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Proposed Question: SRO 9 Rev: 0

Given:

- Plant is in Mode 3 after a loss of all RCPs resulted in a Reactor Trip
- The crew is performing OP-902-003, Loss of Offsite Power/Loss of Forced Circulation Recovery
- Ability to manually steam the Steam Generators has been lost in the control room
- Natural Circulation has not been established
- The CRS has directed the NAO to operate the ADVs locally to enhance natural circulation

The CRS directs the NAO to operate the ADVs locally to establish natural circulation. The ADVs may be operated locally by ____(1)____. In accordance with OP-902-003, proper feeding and steaming will be verified by observing Steam Generator pressure approximately equal to the saturation pressure for its associated loop _____(2)_____.

	(1)	(2)
A.	manual handwheel or local pneumatic	Thot
B.	manual handwheel or local pneumatic	Tcold
C.	manual handwheel only	Thot
D.	manual handwheel only	Tcold

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Per OP-902-003 step 15.1, Natural Circulation if not confirmed will direct proper feeding and steaming by observing steam generator pressure is approximately equal to associated loop Tcold. Part 1 is correct.
B. **CORRECT**: Atmospheric Dump valves may be operated in 2 different manners either by local pneumatic or local manual handwheel. Per OP-902-003 step 15.1, Natural Circulation if not confirmed will direct proper feeding and steaming by observing steam generator pressure is approximately equal to associated loop Tcold.
C. Incorrect: Atmospheric Dump valves may be operated in 2 different manners either by local pneumatic or local manual handwheel. Per OP-902-003 step 15.1, Natural Circulation if not confirmed will direct proper feeding and steaming by observing steam generator pressure is approximately equal to associated loop Tcold.
D. Incorrect Atmospheric Dump valves may be operated in 2 different manners either by local pneumatic or local manual handwheel. Part 2 is correct.

Technical Reference(s): OP-902-003 revision 8
(Attach if not previously provided) OP-005-004 revision 26
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 obj. 4 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS
Check Single Phase Natural Circulation	
* 15. **IF NO** RCPs are operating, **THEN** check natural circulation flow in at least one loop by **ALL** of the following: • RCS subcooling greater than or equal to 28°F based on representative CET temperature • Loop ΔT less than 58°F • Hot and cold leg temperature constant or lowering • T_H and representative CET temperature ΔT is less than 10°F	15.1 Verify proper steam generator feeding and steaming by verifying operating loop steam generator pressure is approximately equal to saturation pressure for the existing Tc.
Restore Electrical Power	
* 16. **WHEN** Electrical power is available, **AND** resources permit, **THEN** REFER TO Appendix 12, "Electrical Restoration", and restore electrical busses as applicable.	

8.7 LOCAL OPERATION OF ATMOSPHERIC DUMP VALVE

NOTE

Refer to T.S. 3.7.1.7 prior to performing this section.

CAUTION

OPENING ADV MAY CAUSE AN RCS COOLDOWN, RESULTING IN A CHANGE IN REACTIVITY AND STEAM GENERATOR LEVELS.

8.7.1 Establish communications with the control room.

8.7.2 If desired to operate MS-116A(B) locally using pneumatic operation, then perform the following:

8.7.2.1 Record the pressure at the outlet of the transducer: ________ PSIG.

8.7.2.2 Adjust the pressure at Air Regulator outlet on the front of the panel to the pressure noted in step 8.7.2.1.

8.7.2.3 Turn the Pneumatic Permissive Valve above the Transducer to Manual.

8.7.2.4 Turn the Pneumatic Permissive Valve above the Air Regulator to Manual.

NOTE

The ADV will be closed when pressure at outlet of Air Regulator is <4.5 PSIG and full open when pressure at outlet of Air Regulator is >15 PSIG.

8.7.2.5 Adjust the Air Regulator pressure to obtain desired valve position by performing any of the following:

- Open MS-116 A(B) by turning air regulator adjusting screw in the clockwise direction to raise air pressure.
- Close MS-116 A(B) by turning the air regulator adjusting screw in the counterclockwise direction to lower air pressure.

8.7.3 When local pneumatic operation of MS-116A(B) is no longer desired, then perform the following:

8.7.3.1 Verify the applicable Atmospheric Dump Valve M/A station on CP-8 is in Manual with minimum output.

8.7.3.2 Turn the Pneumatic Permissive Valve above the Air Regulator from Manual to Auto.

8.7.3.3 Raise the pressure of the Air Regulator on the front of the panel until pressure no longer rises. Check that outlet pressure on the Air Regulator on the front of the panel is between 20-25 PSIG.

8.7.3.4 Adjust Air Regulator outlet pressure on the front of the panel until the outlet pressure just begins to drop. Check that Air Regulator outlet pressure on the front of the panel is between 20-25 PSIG.

CAUTION

IF THE PRESSURE NOTED IN STEP 8.7.3.5.1 IS GREATER THAN 4.5 PSIG, THEN THE ADV MAY OPEN WHEN RETURNED TO REMOTE OPERATION.

8.7.3.5 Verify pressure at outlet of Transducer is ■ 4.5 PSIG.

8.7.3.5.1 If the pressure is greater than 4.5 PSIG, then verify the applicable Atmospheric Dump Valve M/A station on CP-8 is in Manual with minimum output.

8.7.3.5.2 If the applicable Atmospheric Dump Valve M/A station is in Manual with minimum output and transducer pressure is ■ 4.5 PSIG, then I&C Maintenance should be contacted for assistance prior to proceeding.

NOTE

Operation of the Air Regulator may loosen the lock nut on the Air Regulator on the front of the panel. The lock nut should be checked for tightness. A 1/2" open end wrench is required to tighten the lock nut. Do not over tighten.

8.7.3.6 Turn the Pneumatic Permissive Valve above the Transducer from Manual to Auto.

8.7.4 If desired to operate MS-116A(B) using local handwheel, then perform the following:

8.7.4.1 Close Local NG/IA Isolation to the positioner for applicable MS-116A(B) SG 1(2) MS Atm Dump Vlv:

- NG MVAAA8271 C NG/IA Isolation to MS-116A

  or

- NG MVAAA9281 D NG/IA Isolation to MS-116B

8.7.4.2 Open Filter Petcock Drain and bleed off Air/N2 pressure by opening Filter Petcock Drain.

8.7.4.3 Open MS-116A(B) SG 1(2) MS Atm Dump Vlv Local Valve Positioner Equalizing Valve.

8.7.4.4 Engage local handwheel and Open MS-116A(B) SG 1(2) MS Atm Dump Vlv to desired position by performing the following:

8.7.4.4.1 Unscrew clevis from top of the Manual Override Shaft.

8.7.4.4.2 Turn handwheel to expose actuator shaft above manual override shaft.

8.7.4.4.3 Slide clevis onto actuator shaft.

8.7.4.4.4 Turn handwheel to open MS-116A(B) to desired position.

8.7.5 When local handwheel operation of MS-116A(B) is no longer desired, then perform the following:

8.7.5.1 Verify the applicable Atmospheric Dump Valve M/A station on CP-8 is in Manual with minimum output.

8.7.5.2 Using the local handwheel, Close SG 1(2) MS Atm Dump Vlv, MS-116A(B).

8.7.5.3 Turn local handwheel until pressure is relieved from clevis and remove clevis from actuator shaft.

8.7.5.4 Turn handwheel until clevis can be installed onto manual override shaft and screw clevis onto manual override shaft.

8.7.5.5 Close Filter Petcock Drain.

8.7.5.6 Open Local NG/IA Isolation to the positioner for applicable MS-116A(B) SG 1(2) MS Atm Dump Vlv:

- NG-8271C NG/IA Isolation to MS-116A

  or

- NG-9281D NG/IA Isolation to MS-116B

8.7.5.7 Close MS-116A(B) SG 1(2) MS Atm Dump Vlv Positioner Equalizing Valve.

8.7.5.8 Place Atmospheric Dump Valve M/A station in desired position as directed by SM/CRS.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	CE/A11 AA2.2	
	Importance Rating		3.4

K/A Statement

AA2.2 - Ability to determine and interpret the following as they apply to the (RCS Overcooling) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: SRO 10 Rev: 0

Given:

- Plant was operating at 100% power with EDG A danger tagged
- An Excess Steam Demand occurred
- SIAS, CIAS, and MSIS have been initiated
- Pressurizer level is 0%
- After entering OP-902-004, Excess Steam Demand Recovery, the following conditions change:
  - Representative CET temperature and RCS pressure start to rise.
  - A Loss of Off Site Power occurs

All components respond as designed to the event.

Based on these conditions, the CRS will ____(1)____ and ____(2)____

	(1)	(2)
A.	remain in OP-902-004	direct the ATC to stabilize RCS pressure **above** HPSI Pump shutoff head.
B.	remain in OP-902-004	direct the ATC to stabilize RCS pressure **below** HPSI Pump shutoff head.
C.	exit OP-902-004 and enter OP-902-008, Functional Recovery	direct the ATC to stabilize RCS pressure **above** HPSI Pump shutoff head.
D.	exit OP-902-004 and enter OP-902-008, Functional Recovery	direct the ATC to stabilize RCS pressure **below** HPSI Pump shutoff head.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**: Required to stay in OP-902-004 to address stabilization of RCS pressure ABOVE the HPSI pump shutoff head. The safety function status checklist in OP-902-004 is still being met with one safety bus powered.

B. INCORRECT: Required to stay in OP-902-004 to address stabilization of RCS pressure ABOVE the HPSI pump shutoff head.

C. INCORRECT: OP-902-004 address the required actions. The safety function status checklist in OP-902-004 is still being met with one safety bus powered. The step to stabilize RCS pressure is contained in OP-902-008, but entry into OP-902-008 is not required.

D. INCORRECT: OP-902-004 address the required actions. The safety function status checklist in OP-902-004 is still being met with one safety bus powered. The step to stabilize RCS pressure is contained in OP-902-008, but entry into OP-902-008 is not required.

Technical Reference(s): OP-902-004 revision 14
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE04 obj. 7 (As available)

Question Source: Bank # X Question #4
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2009 SRO Exam

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Actions to stabilize RCS temperature following an excess steam demand event should be initiated when **BOTH** of the following parameters are met:

- CET temperatures rise
- Pressurizer pressure rise

Stabilize RCS Temperature

* 16. Verify RCS temperature is stabilized by performing the following:

a. Place the ADV for the least affected steam generator to manual and fully open the ADV.

b. Manually initiate EFAS for the least affected steam generator.

c. Place the EFW Flow Control Valve to manual and commence feeding the least affected steam generator.

d. **IF** RCS pressure is greater than or equal to 1500 psia, **THEN** stabilize RCS pressure at a value not to exceed 1600 psid between the RCS and the lowest SG pressure.

d.1 **IF** RCS pressure is less than 1500 psia, **THEN** stabilize RCS pressure at greater than HPSI shutoff head (1500-1600 psia).

SAFETY FUNCTION:

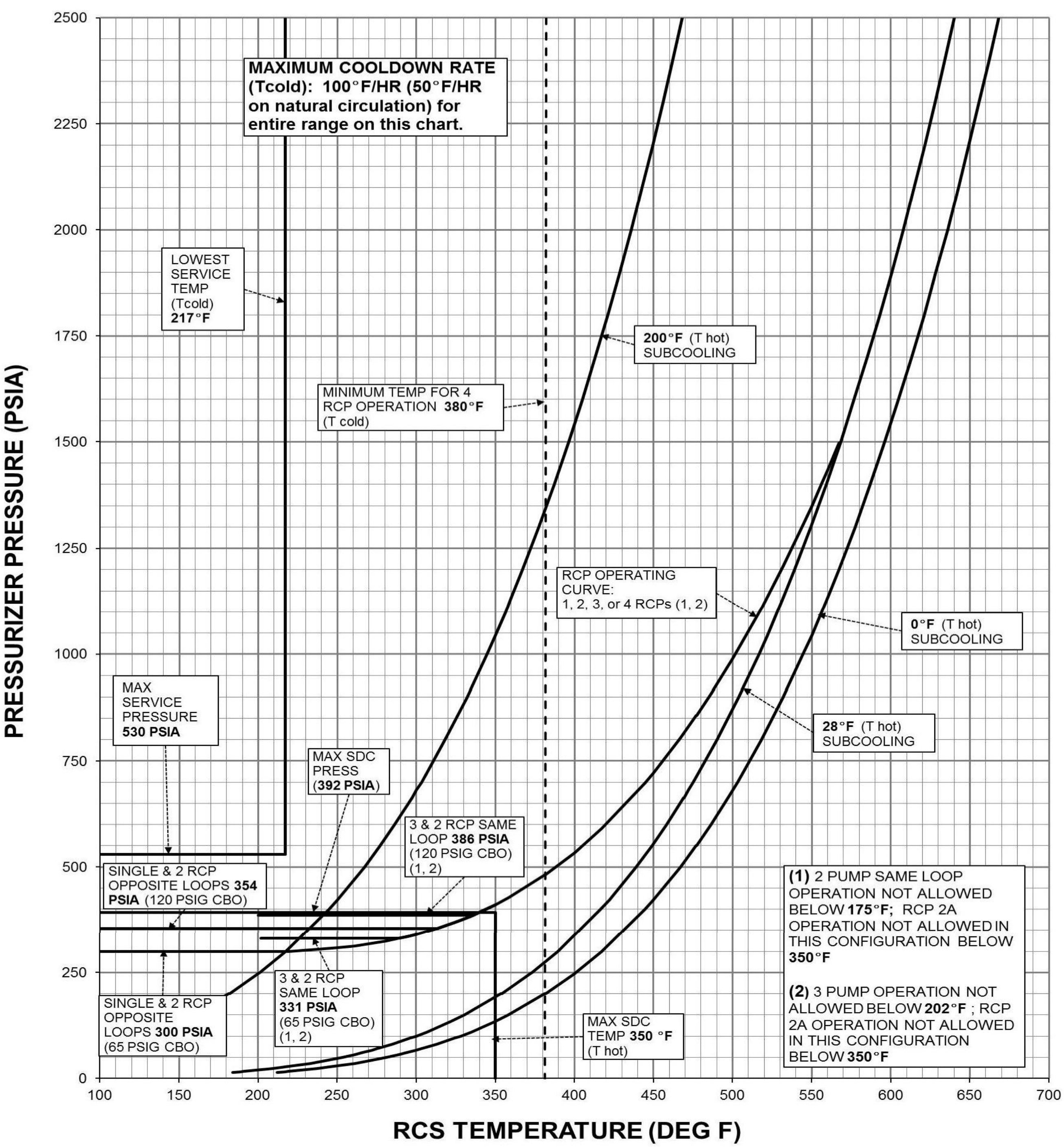
2. Maintenance of Vital Auxiliaries (AC and DC Electrical Power)

PARAMETER	CRITERIA	CRITERIA SATISFIED
Condition 1		
a. Bus A3S, A-DC and at least one associated vital AC Instrument Channel	Energized	_____ _____ _____ _____
Condition 2		
a. Bus B3S, B-DC and at least one associated vital AC Instrument Channel	Energized	_____ _____ _____ _____

Figures

Attachment 2-A: RCS Pressure and Temperature Limits

RCS PRESSURE AND TEMPERATURE LIMITS (NON-HARSH)

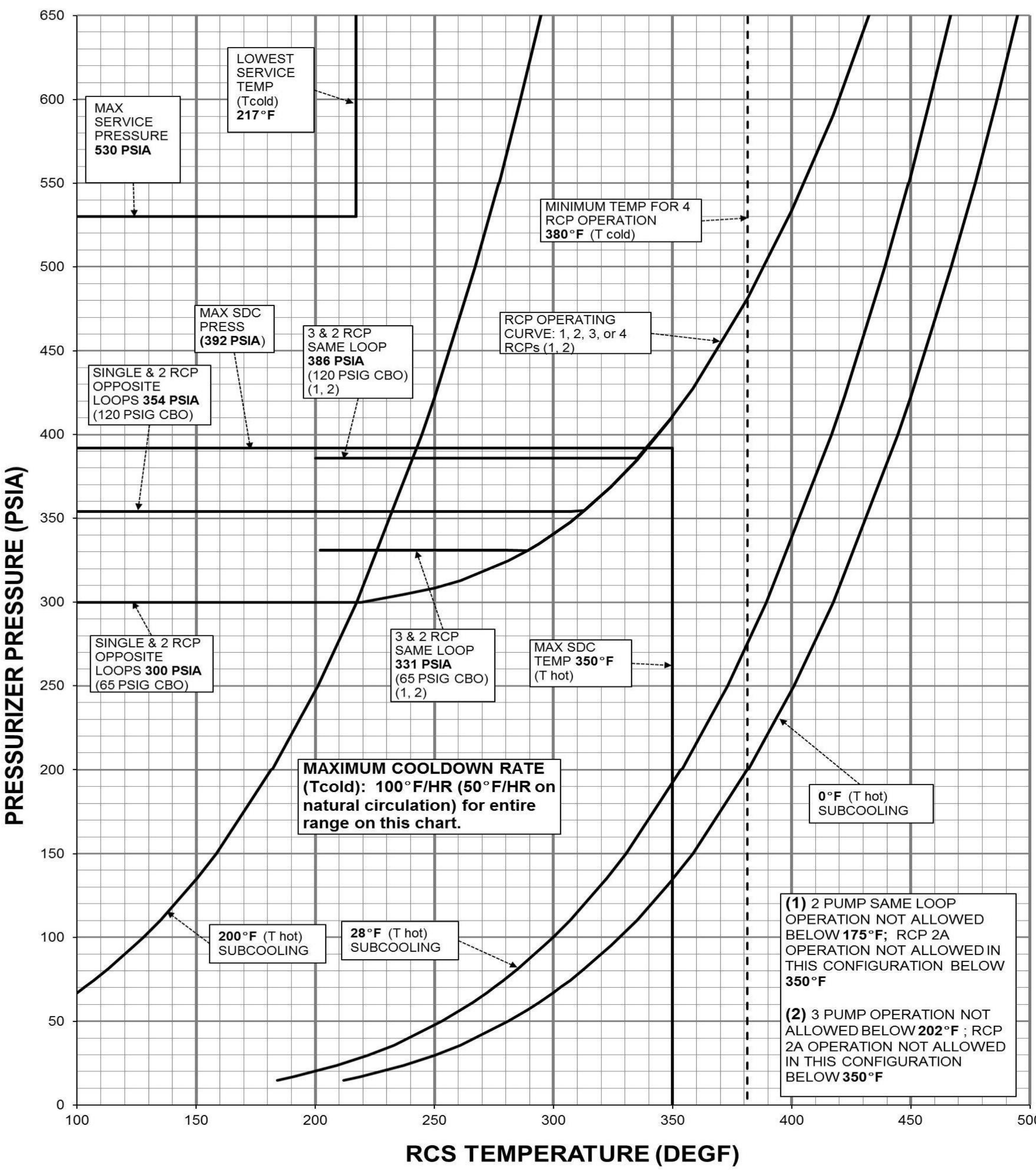


End of Attachment 2-A

Figures

Attachment 2-B: RCS Pressure and Temperature Limits

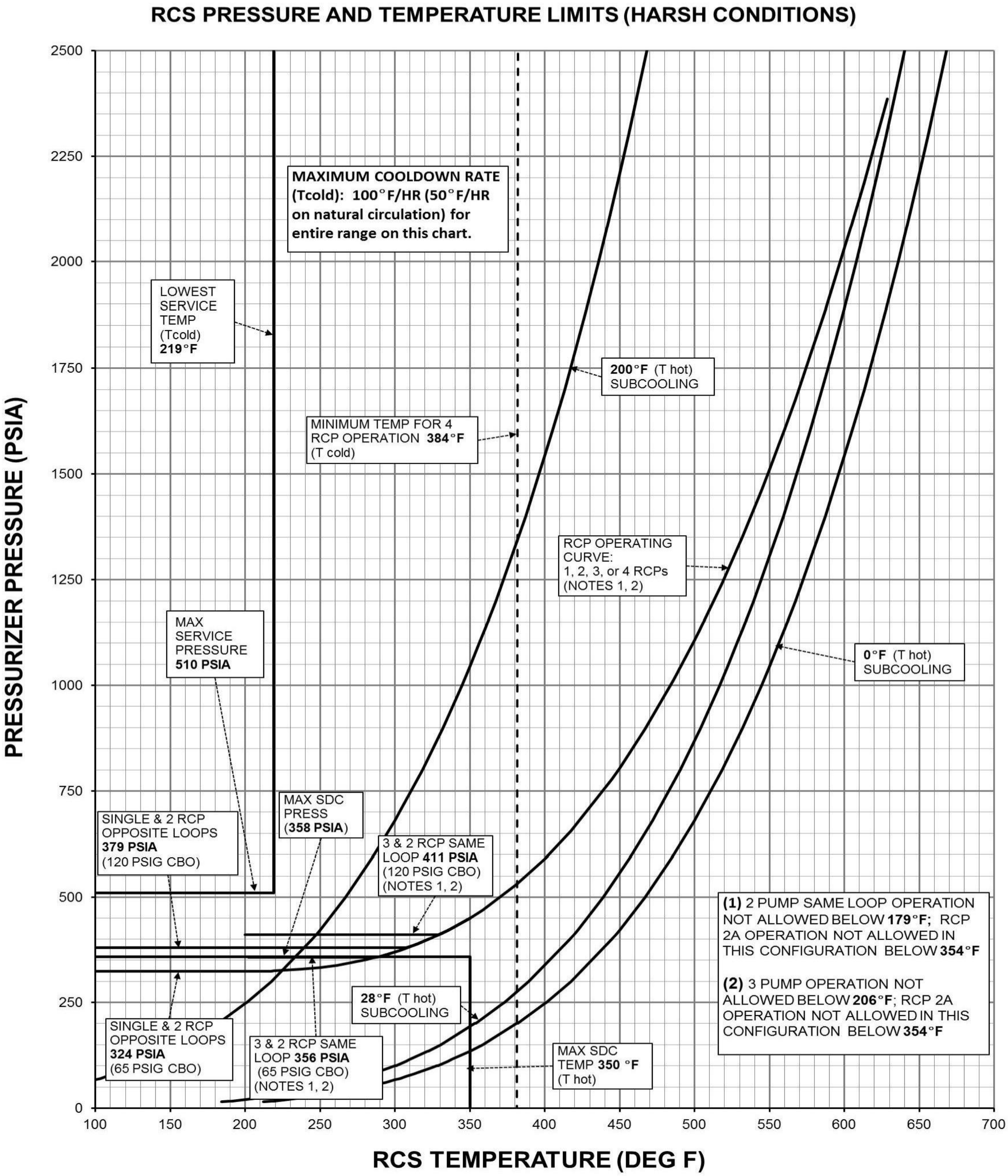
RCS PRESSURE AND TEMPERATURE LIMITS (NON-HARSH) Expanded View



End of Attachment 2-B

Figures

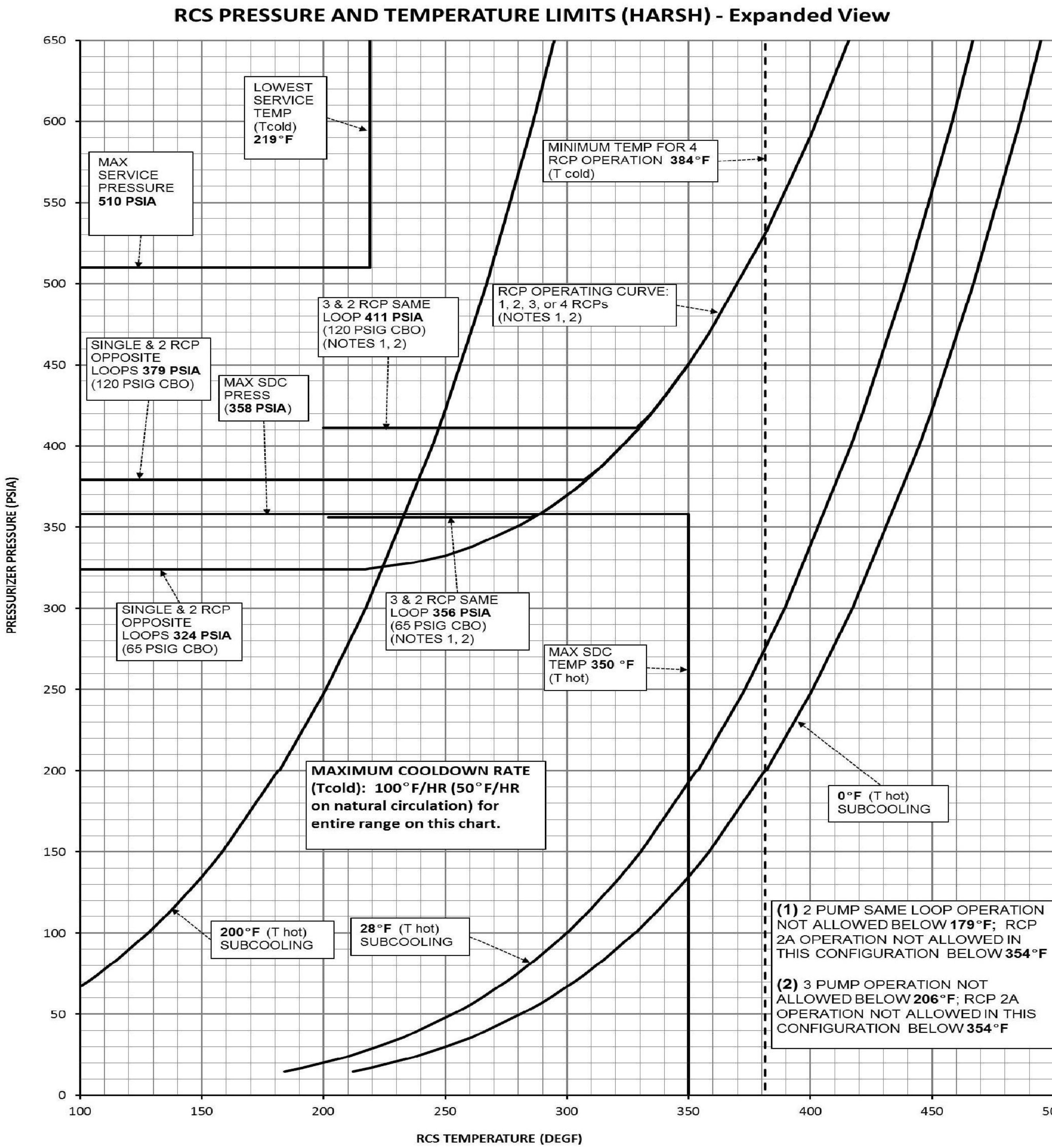
Attachment 2-C: RCS Pressure and Temperature Limits [Harsh Environment]



End of Attachment 2-C

Figures

Attachment 2-D: RCS Pressure and Temperature Limits [Harsh Environment]



End of Attachment 2-D

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	010 G2.1.25	
	Importance Rating		4.2

K/A Statement

2.1.25 –Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: SRO 11 Rev: 0

Given:

- A SGTR event is in progress with S/G #2 faulted
- The crew has completed the rapid cooldown of the RCS
- RCS T_{HOT} is 518 °F
- RCS T_{COLD} is 504 °F
- RCS pressure is 1450 PSIA
- S/G #2 pressure is 650 PSIA
- Reactor Coolant Pumps 1B and 2B are operating
- OP-902-007, Steam Generator Tube Rupture Recovery directs the reduction of RCS pressure.

Based on plant conditions, the CRS should direct the ATC to reduce RCS pressure, with a MINIMUM pressure of

A. 700 PSIA

B. 940 PSIA

C. 1100 PSIA

D. 1200 PSIA

Proposed Answer: C

Explanation: (Optional)

A. INCORRECT: Within 50 PSIA of the affected S/G is a target pressure, but plant conditions do not support lowering this far with RCPs running.

B. INCORRECT: Lowering RCS pressure < 945 PSIA is a target pressure, but plant conditions do not support lowering this far with RCPs running.

C. **CORRECT**: This is the lowest pressure that supports 2 RCPs running with T_{COLD} at 504 °F. The Tech Guide for OP-902-007 states that if continued RCP operation is possible, the requirement to maintain NPSH takes precedence over the strategy of equalizing primary pressure and secondary pressure. The applicant must know the Tech Guide guidance that keeping continued operation of the RCPs takes precedence over reducing RCS pressure below RCP operating limits. The applicant will use figure 2A of OP-902-009.

D. INCORRECT: This pressure will support continued operation of the Reactor Coolant Pumps, but is not the minimum pressure that RCS pressure could be reduced. The applicant would get this answer if Thot and not Tcold was applied to figure 2-A

Technical Reference(s): OP-902-007 Step 12 revision 15
(Attach if not previously provided) OP-902-009, Att 2-A revision 309
(including version/revision number) TGOP-902-007 step 12 revision 305

Proposed references to be provided to applicants during examination: OP-902-009, Att 2-A and Att 2-C

Learning Objective: WLP-OPS-PPE04 obj. 3 (As available)

Question Source: Bank # X Question #12
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2009 NRC SRO Exam

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

10 CFR Part 55 Content: 55.41
55.43 5

INSTRUCTIONS	CONTINGENCY ACTIONS
Cooldown RCS to Less Than 520°F T_H	
11. Commence a rapid RCS cooldown to less than 520°F T_H using the steam bypass valves.	11.1 Commence a rapid RCS cooldown to less than 520°F T_H using **BOTH** atmospheric dump valves.
Maintain RCS Pressure	
* 12. Depressurize the RCS: a. Maintain pressurizer pressure within **ALL** of the following criteria: • Within Appendix 2A-D, "RCS Pressure and Temperature Limits" • Less than 930 psia • Within 50 psi of the most affected steam generator pressure • **IF** RCPs are operating, greater than the minimum RCP NPSH of Appendix 2A-D, "RCS Pressure and Temperature Limits" b. Operate main or auxiliary pressurizer spray. c. **IF** HPSI throttle criteria are met, **THEN** perform **ANY** of the following: • Control charging and letdown flow • Throttle HPSI flow	

Step Number 12 Maintain RCS Pressure

Objective

The intent of this step is to establish control of RCS pressure. The general goals associated with RCS pressure control are:

- Providing subcooling to support the core heat removal process
- Avoiding overpressure situations for PTS and RT NDT considerations
- Minimizing the pressure differential between the steam generator and the RCS to minimize the leakage
- Deliberately creating a primary to secondary differential pressure to establish backflow to control SG level rise or reduce SG pressure/temperature
- Controlling RCS pressure below the atmospheric dump valve or main steam safety valve lift pressure to prevent uncontrolled release of radioactivity to the environment

Instructions

Maintaining the RCS pressure below the lift setpoint of the ADV, within the PT limits and approximately equal to the isolated steam generator pressure [±50 psi] will minimize the loss of primary fluid to the secondary side and the possibility of overfilling the isolated SG. This action will minimize the potential for release of radiation to the environment by minimizing RCS to steam generator leakage.

The RCS pressure limit is reduced to 930 psia from 945 psia. The 930 psia is based on instrument uncertainty for a containment temperature of 200°F while the 945 psia was based on instrument uncertainty for a containment temperature of 120°F. This change is consistent in applicable EOPs.

Maintaining RCS pressure approximately equal to or less than the affected SG pressure allows for the backflow of secondary water into the RCS which provides several operational benefits. These benefits include:

- SG level can be maintained within the indicating range by controlling SG pressure
- The probability of filling the main steam piping with water is greatly reduced
- Use of the blowdown system for SG level control can be minimized, thus minimizing contamination of the secondary
- Depressurization of the isolated SG can be performed without steaming to the condenser or to the atmosphere
- Less secondary makeup water is required for the RCS cooldown

Step Number 12 Maintain RCS Pressure (cont)

Boron dilution of the RCS will occur due to unborated secondary water flowing through the tube rupture into the RCS. However, under most circumstances, this dilution will not threaten the maintenance of adequate shutdown margin.

A important point that the SGTR event strategy is to maintain or restore forced circulation as soon as possible to minimize the adverse affects of RCS dilution. On the other hand, maintaining adequate NPSH for RCP operation may cause the operator to hold RCS pressure above secondary pressure until the RCS is cooled down beyond the SG isolation temperature. If continued RCP operation is possible, the requirement to maintain NPSH takes precedence over the strategy of equalizing primary pressure and secondary pressure.

During the forced circulation cooldown process, the isolated steam generator may cool faster in the lower regions. The cooling of the isolated SG steam space will significantly lag in the cooldown and cause the fluid in the lower regions to be subcooled. If the tube rupture is located in this subcooled region, as it is most likely to be, then the primary fluid can be at the same pressure as the secondary fluid and still be subcooled. However the continued depressurization of the primary during the cooldown will now be limited by the ability to depressurize the isolated SG.

During natural circulation cooldown conditions the isolated steam generator will not cool unless there is a transfer of mass in the isolated SG. This complicates RCS pressure control during the cooldown. It is desirable to cool the RCS such that the tube bundle region of the affected SG remains subcooled. Voiding in the tube bundle region can be expected and may result in the region becoming a pressurizing source for the RCS. Maintaining the presence of subcooled liquid in affected loop will be a complicated process under natural circulation conditions. Forced circulation conditions are much more desirable and if possible should be maintained or restored. During natural circulation conditions, the cooldown and depressurization of the RCS will be limited to the operator's ability to control the conditions of the isolated steam generator.

Pressurizer pressure may be reduced by any of the following:

- Operation of Pressurizer sprays and heaters.
- Control of charging pumps, letdown and HPSI pumps (if HPSI Throttle criteria are met).

Step Number 12 Maintain RCS Pressure (cont)

Contingency Actions

None

Justification for Deviations

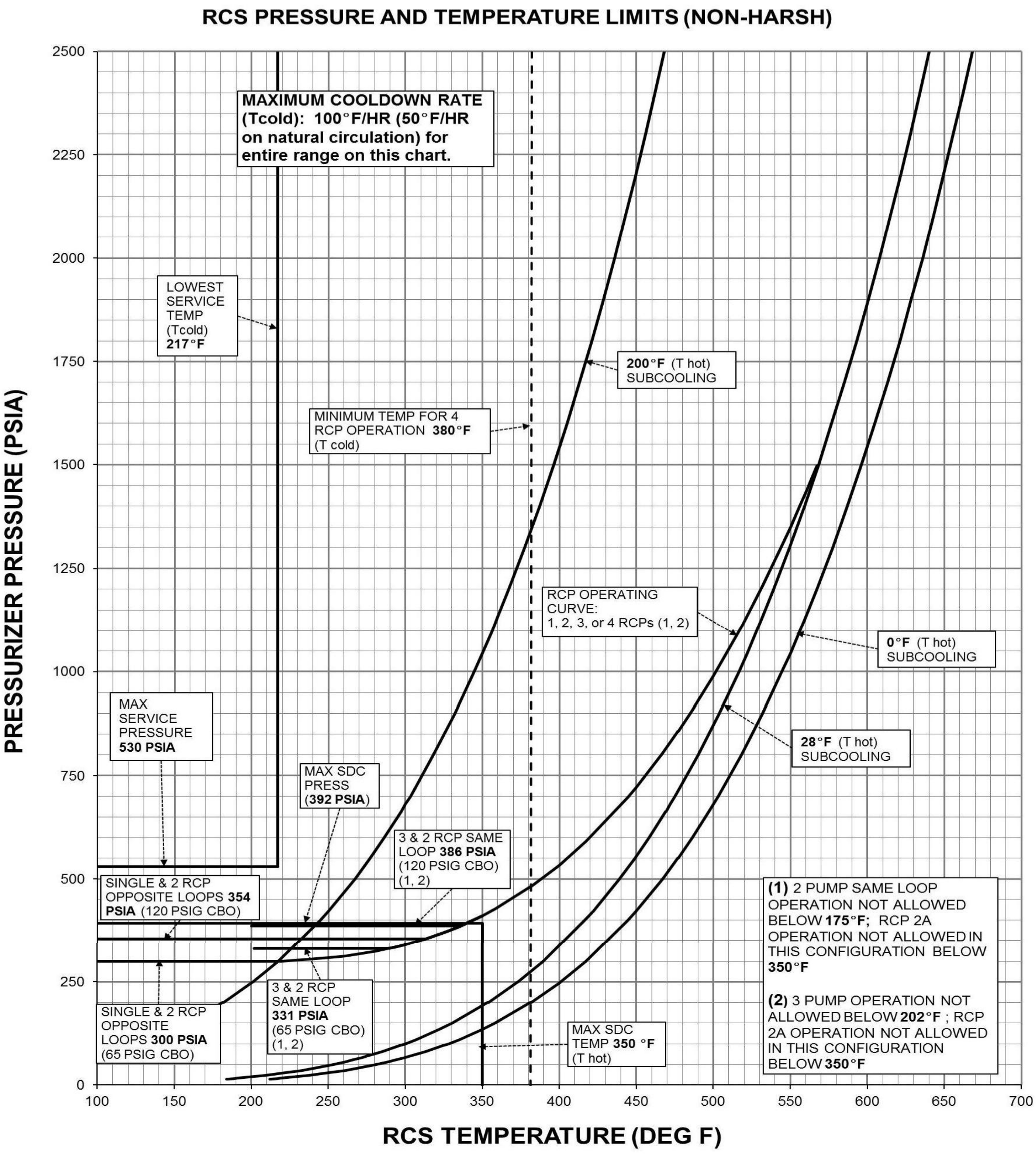
Waterford does not include Contingency Action 9.1 since PORVs are not part of the plant design.

References

1. ECS98-001 P.16
2. ECS98-001 P.29
3. EC 41278

Figures

Attachment 2-A: RCS Pressure and Temperature Limits



End of Attachment 2-A

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	005 2.4.45	
	Importance Rating		4.3

K/A Statement

2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: SRO 12 Rev: 0

Given:

- RCS temperature is 340°F
- RCS pressure is 380 PSIA
- SDC Train A is in service
- SDC Train B is in standby

The following annunciators are received:

- SIAS Train A Logic Initiated (Cabinet K, G-19)
- SIAS Train B Logic Initiated (Cabinet K, G-20)
- LOOP 1 SDC RELIEF VLV ACTIVE (Cabinet N, A-17)
- LOOP 2 SDC RELIEF VLV ACTIVE (Cabinet M, A-7)

The CRS will direct the BOP to FIRST ____(1)____ in accordance with the guidance in ____(2)____.

	(1)	(2)
A.	secure HPSI Pumps	OP-901-131, Shutdown Cooling Malfunction
B.	secure LPSI Pump B	OP-901-131, Shutdown Cooling Malfunction
C.	secure LPSI Pump B	OP-901-504, Inadvertent ESFAS Actuation
D.	secure HPSI Pumps	OP-901-504, Inadvertent ESFAS Actuation

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: HPSI pumps must be secured to prevent lifting LTOP reliefs. Pressure will rise to the shutoff head of HPSI pumps if they are not secured. The guidance is located in OP-901-504 and not in OP-901-131, Shutdown Cooling Malfunction, although the CRS may enter both off-normals.
B. Incorrect: OP-901-504 directs the crew to secure LPSI pump B but not till after HPSI pumps are secured because prevention of lifting LTOPs is a higher priority. The guidance is located in OP-901-504 and not in OP-901-131, Shutdown Cooling Malfunction, although the CRS may enter both off-normals.
C. Incorrect: OP-901-504 directs the crew to secure LPSI pump B but not till after HPSI pumps are secured because prevention of lifting LTOPs is a higher priority. Part 2 is correct.
D. **CORRECT**: HPSI pumps must be secured to prevent lifting LTOP reliefs. Pressure will rise to the shutoff head of HPSI pumps if they are not secured. The guidance is located in OP-901-504, Inadvertent ESFAS actuation. With the annunciators given, it can be determined that an inadvertent SIAS has occurred.

Technical Reference(s): OP-901-504 revision 8
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO50 obj. 3 (As available)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

Comments:

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

	PLACEKEEPER START	DONE	N/A
1. IF in Modes 3, 4, OR 5 AND any of the following occur, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS. • Unexpected Pressurizer Pressure drop AND Pressurizer Level changing • Unexpected Containment Pressure rise • Unexpected Steam Generator Pressure drop • Unexpected Steam Generator Level drop.		☐	☐
2. IF an inadvertent SIAS/CIAS occurs, THEN perform the following:		☐	☐
2.1 IF a Reactor Trip occurs, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS.		☐	☐
2.2 GO TO Subsection E_1 Inadvertent SIAS/CIAS.		☐	
3. IF an inadvertent CSAS occurs, THEN perform the following:		☐	☐
3.1 IF a Reactor Trip occurs, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS AND perform concurrently with this procedure.		☐	☐
3.2 GO TO Subsection E_2, Inadvertent CSAS.		☐	
4. IF an inadvertent RAS occurs, THEN GO TO Subsection E_3 Inadvertent RAS.		☐	☐
5. IF an inadvertent MSIS occurs, THEN perform the following:		☐	☐
5.1 IF a Reactor Trip occurs, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS AND perform concurrently with this procedure.		☐	☐
5.2 GO TO Subsection E_4 Inadvertent MSIS.		☐	

E$_1$ INADVERTENT SIAS/CIAS

PLACEKEEPER		
START	DONE	N/A

NOTE

If a Safety Injection actuation occurs with RCS pressure less than HPSI Pump shutoff head, the following transient response may be expected:

- Pressurizer level will rise and approach solid conditions due to HPSI and Charging Pump flows
- RCS temperatures will initially drop due to cold water being injected
- RCS pressure will rise to shutoff head of HPSI pumps
- Pressurizer water temperature will drop due to Pressurizer in-surge and loss of Pressurizer Heaters.

Step	Action	START	DONE	N/A
1.	Stop <u>ALL</u> operating HPSI Pumps by placing control switches to OFF.		☐	
2.	Operate Charging Pumps as necessary to maintain required Pressurizer Level.	☐	Continuous	
3.	Stop LPSI Pumps that are <u>NOT</u> being used for SDC <u>AND</u> place control switch(es) in OFF.		☐	

NOTE

Instrument Air Compressor Auto-Start feature is inhibited on SIAS. The idle compressor may be started locally.

Step	Action	START	DONE	N/A
4.	Open IA 909, CNTMT ISOLATION INSTRUMENT AIR.		☐	
5.	Start <u>TWO</u> CEDM Cooling Fans in accordance with OP-008-004, CONTROL ELEMENT DRIVE MECHANISM COOLING SYSTEM.			

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	012 A2.01	
	Importance Rating		3.6

K/A Statement

A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty bistable operation

Proposed Question: SRO 13 Rev: 0

The following plant conditions exist:

- The plant is in Mode 1
- Channel ‘B’ Wide Range Pressure transmitter (RC-IPI-0102B) has failed low
- Appropriate bistables have been bypassed on Channel ‘B’
- Subsequently, Channel ‘C’ Lo Pressurizer Pressure bistable has failed in the actuated state.

If the CRS directs bypassing the Channel ‘C’ Lo Pressurizer Pressure bistable, then:

A. Channel ‘B’ Lo Pressurizer pressure remains bypassed and Channel ‘C’ Lo Pressurizer pressure remains bypassed. No additional actions are required.

B. Channel ‘B’ Lo Pressurizer pressure remains bypassed, Channel ‘C’ does not bypass. Channel ‘C’ must be placed in trip to comply with Tech Spec 3.3.1.

C. A Reactor Trip and SIAS occurs. OP-901-504, Inadvertent ESFAS Actuation directs that Instrument Air will be restored to containment by opening IA-909, CNTMT ISOLATION INSTRUMENT AIR.

D. A Reactor Trip and SIAS occurs. OP-901-504, Inadvertent ESFAS Actuation directs that emergency boration may only be secured after Shutdown Margin is verified through Chemistry analysis.

Proposed Answer: C

Explanation: (Optional)

A. INCORRECT: Neither bistable will be bypassed since the wide range instrument feeds the low pressure bistable.
B. Incorrect: Neither bistable will be bypassed since the wide range instrument feeds the low pressure bistable. Channel C would be required to be placed in the tripped condition to comply with Tech Specs, but part 1 is incorrect.
C. **CORRECT**: The low pressure bistables actuate both the reactor trip RPS function and the ESFAS function. Instrument air isolates to containment and OP-901-504, Inadvertent ESFAS Signal directs restoring instrument air to containment.
D. Incorrect: Part 1 is correct. Emergency Boration does occur from the SIAS, but OP-901-504 does not require chemistry analysis to confirm SDM prior to securing Emergency Boration.

Technical Reference(s): OP-901-504, Inadvertant ESFAS Actuation Revision 8
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPS00 obj. 3 and 10 (As available)

Question Source: Bank # X
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2009 SRO NRC Exam

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

The lamps flash to remind the operator to reset the tripped function during testing. To reset the TCBs following a trip, the LK/UNLK switches must be placed in the UNLK position to enable the RPS reset pushbuttons by completing a series path between the reset pushbuttons and the closing coils of the associated TCB via LK/UNLK switch contacts. Then the eight RPS pushbuttons must be depressed to reset (close) their associated TCBs.

RPS Trip Matrices

Figure 8 shows all 6 RPS trip matrices. De-energizing any of the 6 RPS matrices will result in a reactor trip. RPS matrix AB is shown in Figure 9. The other 5 matrices are similar to matrix AB. Each matrix consists of bistable relay contacts connected in the form of a ladder. The contacts in Figure 9 are shown in energized, non-tripped state. The closed K101 contacts on the left side of Figure 9 are operated by channel A bistable relays. The closed K101 contacts on the right side of Figure 9 are operated by channel B bistable relays. When the K101 contacts for both channels are open for any parameter, the matrix relays will drop out. Auctioneered DC power supplies from each channel are connected in parallel to one end of the ladder. Four matrix output relays, are connected in parallel at the other end. With this configuration, a failure of one of the power supplies will not result in a complete matrix trip; however, two relays will trip and cause two K relays to de-energize, tripping half the reactor trip breakers (which is not enough to cause a reactor trip). This situation can occur on loss of a single 120 VAC SUPS failure. When a logic matrix does trip, its four matrix output relays will de-energize. The matrix output relays open contacts in four trip paths and de-energize the four K relays to initiate a reactor trip. The K501-K515 contacts are trip channel bypass contacts. Trip channel bypass relays energize to close these contacts. The trip channel bypass circuit is shown on Figure 11.

Indicating fuses for each power supply are located in the back of CP-10. The fuse lamp will be lit if the fuse is blown. R1 lamps are also located in the back of CP-10. The lights are energized when the power supply is energized.

Trip Channel Bypasses

All trip bistables have Trip Channel Bypass capability to remove them from service for maintenance or testing. When one channel's bistable for a particular trip function is in Trip Channel Bypass, the trip logic is converted to 2/3 by relying on the three remaining channels. This bypass is both initiated and removed manually by toggle action pushbuttons (shown on Figure 11). There is an electrical interlock which allows only one channel for a given trip function to be bypassed at a time. Attempting to place two bistables in Trip Channel Bypass for a given trip function will result in both bistables defaulting to an unbypassed condition. Trip Channel Bypass is accomplished exclusively on the BCP portion of CP-10 via pushbuttons located behind a locked panel door beneath the trip indicators. Each trip parameter has its own toggle action, latchdown pushbutton. When a channel is placed in Trip Channel Bypass, amber

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

	PLACEKEEPER		
	START	DONE	N/A
1. IF in Modes 3, 4, OR 5 AND any of the following occur, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS. • Unexpected Pressurizer Pressure drop AND Pressurizer Level changing • Unexpected Containment Pressure rise • Unexpected Steam Generator Pressure drop • Unexpected Steam Generator Level drop.		☐	☐
2. IF an inadvertent SIAS/CIAS occurs, THEN perform the following:		☐	☐
2.1 IF a Reactor Trip occurs, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS.		☐	☐
2.2 GO TO Subsection E_1 Inadvertent SIAS/CIAS.		☐	
3. IF an inadvertent CSAS occurs, THEN perform the following:		☐	☐
3.1 IF a Reactor Trip occurs, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS AND perform concurrently with this procedure.		☐	☐
3.2 GO TO Subsection E_2, Inadvertent CSAS.		☐	
4. IF an inadvertent RAS occurs, THEN GO TO Subsection E_3 Inadvertent RAS.		☐	☐
5. IF an inadvertent MSIS occurs, THEN perform the following:		☐	☐
5.1 IF a Reactor Trip occurs, THEN GO TO OP-902-000, STANDARD POST TRIP ACTIONS AND perform concurrently with this procedure.		☐	☐
5.2 GO TO Subsection E_4 Inadvertent MSIS.		☐	

E_1 INADVERTENT SIAS/CIAS

PLACEKEEPER		
START	DONE	N/A

NOTE

If a Safety Injection actuation occurs with RCS pressure less than HPSI Pump shutoff head, the following transient response may be expected:

- Pressurizer level will rise and approach solid conditions due to HPSI and Charging Pump flows
- RCS temperatures will initially drop due to cold water being injected
- RCS pressure will rise to shutoff head of HPSI pumps
- Pressurizer water temperature will drop due to Pressurizer in-surge and loss of Pressurizer Heaters.

Step		START	DONE	N/A
1.	Stop ALL operating HPSI Pumps by placing control switches to OFF.		☐	
2.	Operate Charging Pumps as necessary to maintain required Pressurizer Level.	☐	Continuous	
3.	Stop LPSI Pumps that are NOT being used for SDC AND place control switch(es) in OFF.		☐	

NOTE

Instrument Air Compressor Auto-Start feature is inhibited on SIAS. The idle compressor may be started locally.

Step		START	DONE	N/A
4.	Open IA 909, CNTMT ISOLATION INSTRUMENT AIR.		☐	
5.	Start TWO CEDM Cooling Fans in accordance with OP-008-004, CONTROL ELEMENT DRIVE MECHANISM COOLING SYSTEM.			

E$_1$ INADVERTENT SIAS/CIAS (CONT'D)

PLACEKEEPER		
START	DONE	N/A

CAUTION

CEDM FANS SHOULD BE STARTED WITHIN 30 MINUTES OF INADVERTENT SIAS/CIAS ACTUATION TO PREVENT DAMAGE TO CEDM COILS.

Step		Action	START	DONE	N/A
6.		Restore Pressurizer Pressure control as follows:	☐	☐	
	6.1	Close SSD-3A-8, SST A32 FEEDER breaker.		☐	
	6.2	Close SSD-3B-9, SST B32 FEEDER breaker.		☐	
	6.3	Operate Pressurizer Heaters as required to control Pressurizer Pressure.	☐	Continuous	
7.		Investigate cause of inadvertent SIAS/CIAS <u>AND</u> initiate corrective action.	☐	☐	
8.		Advise SM/CRS to initiate a Condition Report in accordance with EN-LI-102, CORRECTIVE ACTION PROCESS.		☐	
9.		<u>IF</u> Pressurizer Pressure is greater than variable low Pzr Pressure trip setpoint, <u>THEN</u> <u>GO</u> <u>TO</u> Step 14.		☐	☐

NOTE

Steps 10 through 13 reset Low PZR Pressure Trip Setpoints.

Step		Action	START	DONE	N/A
10.		<u>IF</u> Pressurizer Pressure is less than variable Low Pzr Pressure trip setpoint, <u>THEN</u> perform the following:	☐	☐	☐
	10.1	Depress TEST POWER SUPPLY push button at CP-10 <u>AND</u> verify indicator Illuminated.		☐	
	10.2	Select channel to be reset with CHANNEL TEST Switch.		☐	

E$_1$ INADVERTENT SIAS/CIAS (CONT'D)

	PLACEKEEPER START	DONE	N/A
10.3 Place BISTABLE SELECT switch to Bistable 6.		☐	
10.4 Place METER INPUT SELECT switch to INPUT <u>AND</u> note DVM reading.		☐	
10.5 Depress <u>AND</u> hold TEST push button on Bistable Control Panel.		☐	
10.6 Raise Test Pot output until DVM reads 4.0 VDC.		☐	
10.7 Depress lamp RESET push button as necessary to clear Trip <u>AND</u> Pretrip lights.		☐	
10.8 Lower Low PZR Pressure setpoint to reset value as follows:		☐	
10.8.1 Lower Test Pot output until LO PZR PRESS Pretrip occurs.		☐	
10.8.2 Depress LOW PZR PRESS SETPOINT RESET push button.		☐	
10.8.3 Depress lamp RESET push button <u>AND</u> verify Pretrip clears.		☐	
10.8.4 Wait 10 seconds.		☐	
10.8.5 Repeat Steps 10.8.1 through 10.8.4 until <u>EITHER</u> of the following exists: • DVM reads less than the reading noted in Step 10.4 with Low PZR Pressure Trip <u>AND</u> Pretrip clear <u>OR</u> • DVM reads 1.133 VDC (100 PSIA).		☐	

E1 INADVERTENT SIAS/CIAS (CONT'D)

	PLACEKEEPER	
START	DONE	N/A

NOTE

One Low PZR Pressure Trip channel may be bypassed to allow SIAS Initiation and Actuation relays to be reset after setpoints have been reset on two channels.

Step	START	DONE	N/A
10.9 Release the TEST push button.		☐	
10.10 Place BISTABLE SELECT switch to OFF.		☐	
10.11 IF PZR Pressure is < 400 PSIA, THEN place RPS/ESFAS PZR PRESS BYPASS switch to BYPASS.		☐	☐
10.12 Place CHANNEL TEST Switch to OFF OR a non-channel position.		☐	
11. Repeat Step 10 for the remaining tripped Low PZR Pressure Trip Channels.	☐	☐	
12. Depress TEST POWER SUPPLY push button AND verify indicator NOT Illuminated.		☐	
13. Place CHANNEL TEST Switch in OFF position.		☐	

CAUTION

RESETTING SAFETY INJECTION ACTUATION OPENS CVC 183, VCT DISCH VALVE, AND STOPS BAM PUMPS.

Step	START	DONE	N/A
14. Reset SIAS AND CIAS Initiation relays on ALL FOUR channels as follows:	☐	☐	
14.1 On CP-10 place Reset Permissive switch to UNLK position.		☐	
14.2 Depress SIAS AND CIAS Reset push buttons.		☐	

E1 INADVERTENT SIAS/CIAS (CONT'D)

PLACEKEEPER: START | DONE | N/A

14.3 Verify initiation relay indicator Illuminated on ENGINEERED SAFETY FEATURES SYSTEM mimic on CP-10:

- Channel A (Red Lamp)
- Channel B (Yellow Lamp)
- Channel C (Green Lamp)
- Channel D (Blue Lamp).

14.4 Place Reset Permissive switch to LK position.

15. Reset SIAS AND CIAS actuation logic on BOTH trains as follows:

15.1 On CP-33 depress BOTH ESFAS RESET push buttons for each ESFAS signal listed below:

- SIAS
- CIAS.

15.2 Verify the following Trip Path Indicators illuminated on ENGINEERED SAFETY FEATURES SYSTEM mimic at CP-10:

- SIAS A 1,3 AND 2,4
- SIAS B 1,3 AND 2,4
- CIAS A 1,3 AND 2,4
- CIAS B 1,3 AND 2,4.

E$_1$ INADVERTENT SIAS/CIAS (CONT'D)

	PLACEKEEPER START	DONE	N/A
16. Verify following alarms have Cleared:	☐	☐	
• SIAS TRAIN A LOGIC INITIATED (CABINET K, G-19)		☐	
• SIAS TRAIN B LOGIC INITIATED (CABINET K, G-20)		☐	
• CIAS TRAIN A LOGIC INITIATED (CABINET K, E-19)		☐	
• CIAS TRAIN B LOGIC INITIATED (CABINET K, E-20).		☐	
17. Secure Emergency Boration as follows:	☐	☐	
17.1 Verify Open CVC 183, VCT DISCH VALVE.		☐	
17.2 Verify Closed CVC 507, RWSP TO CHARGING PUMPS.		☐	
17.3 Verify BORIC ACID PUMPS A <u>AND</u> B are Stopped.		☐	
17.4 Verify the following valves are Closed:	☐	☐	
• BAM 133 EMERGENCY BORATION VALVE		☐	
• BAM 113A GRAVITY FEED VALVES TANK A		☐	
• BAM 113B GRAVITY FEED VALVES TANK B.		☐	
17.5 Verify the following valves Open:	☐	☐	
• BAM 126A PUMP A RECIRC VALVE		☐	
• BAM 126B PUMP B RECIRC VALVE.		☐	
18. Restore Charging <u>AND</u> Letdown in accordance with OP-002-005, CHEMICAL AND VOLUME CONTROL.	☐	☐	

Examination Outline Cross-Reference:			
	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013 A2.05	
	Importance Rating		4.2

K/A Statement

A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc control power

Proposed Question: SRO 14 Rev: 0

Given:

- A failure has occurred that has resulted in a loss of DC control power to LPSI Pump B breaker only.
- The control room received a LPSI PUMP B TRIP/TROUBLE annunciator.
- Subsequently, a Large Break LOCA occurs
- Containment pressure is 22.8 PSIA and steady
- RCS pressure is 350 PSIA and lowering
- Low Pressure Safety Injection (LPSI) Pump A tripped on overcurrent

In response to this event, the SRO should ____(1)___ and ____(2)____ to address the Inventory Control safety function.

	(1)	(2)
A.	continue in OP-902-002, Loss of Coolant Accident Recovery	realign CS Pump ‘A’ to replace LPSI Pump ‘A’
B.	continue in OP-902-002, Loss of Coolant Accident Recovery	verify LPSI flow and HPSI flow are meeting their flow curves
C.	go to OP-902-008, Functional Recovery	realign CS Pump ‘A’ to replace LPSI Pump ‘A’
D.	go to OP-902-008, Functional Recovery	verify LPSI flow and HPSI flow are meeting their flow curves

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: On a loss of control power to LPSI Pump B breaker, the breaker will not close on a SIAS. In this scenario, there will be no LPSI pumps available and the crew will have to go to OP-902-008 due to not meeting the SFSC in OP-902-002. The step that refers the operator to the appendix for realigning CS Pump A to replace LPSI pump A is only found in OP-902-008. Part 2 is plausible because ORPs do refer to appendices but not this particular appendix.
B. Incorrect: On a loss of control power to LPSI Pump B breaker, the breaker will not close on a SIAS. In this scenario, there will be no LPSI pumps available and the crew will have to go to OP-902-008 due to not meeting the SFSC in OP-902-002. The step that refers the operator to the appendix for realigning CS Pump A to replace LPSI pump A is only found in OP-902-008. If the applicant concludes that LPSI Pump B will start due to a loss of control power, he could determine that the LPSI curves are still being met and should be verified.
C. **CORRECT**: On a loss of control power to LPSI Pump B breaker, the breaker will not close on a SIAS. In this scenario, there will be no LPSI pumps available and the crew will have to go to OP-902-008 due to not meeting the SFSC in OP-902-002. The step that refers the operator to the appendix for realigning CS Pump A to replace LPSI pump A is only found in OP-902-008. Part 2 is plausible because ORPs do refer to appendices but not this particular appendix
D. Incorrect: Part 1 is correct. If the applicant concludes that LPSI Pump B will start due to a loss of control power, he could determine that the LPSI curves are still being met and should be verified.

Technical Reference(s): OP-902-008 step 3 page 103 revision 22
(Attach if not previously provided) OP-500-012 Attachment 4.43 Revision 12
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE08 obj. 4 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

SAFETY FUNCTION:

3. RCS Inventory Control (cont)

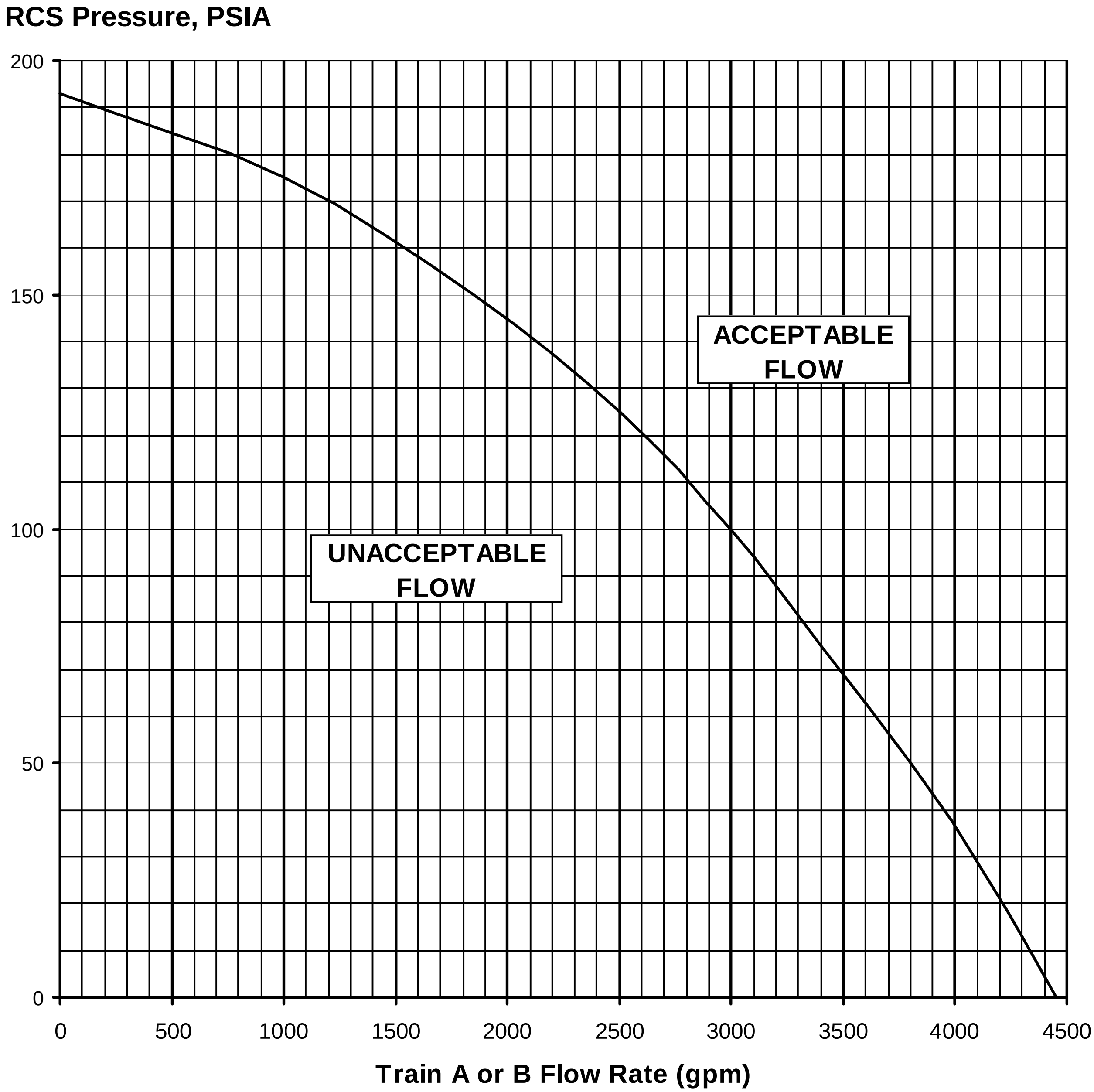
NOTE

Core uncovery and superheated conditions may be expected for up to 30 minutes for some LOCA events. If the SI flow is in accordance with the SI flow curves exiting this procedure to the OP-902-008, Functional Recovery Procedure, will not provide any additional guidance to restore inventory control.

PARAMETER	CRITERIA	CRITERIA SATISFIED
Condition 2 RAS NOT Actuated		
a. HPSI flow	Appendix 2-E, "HPSI Flow Curve"	____ ____ ____ ____
b. LPSI flow	Appendix 2-F, "LPSI Flow Curve"	____ ____ ____ ____
c. RVLMS LEVEL PLENUM	$\geq$ 20%	____ ____ ____ ____
Condition 3 RAS Actuated		
a. HPSI flow	Appendix 2-E, "HPSI Flow Curve"	____ ____ ____ ____
b. RVLMS LEVEL PLENUM	$\geq$ 20%	____ ____ ____ ____

Figures

Attachment 2-F: LPSI Flow Curve



End of Attachment 2-F

SAFETY FUNCTION: **RCS Inventory Control**
SUCCESS PATH: **IC-2: Safety Injection**
RESOURCE TREE: **Tree C**

INSTRUCTIONS	CONTINGENCY ACTIONS
HPSI Pump Restart Criteria	
* 7. **IF ANY** of the HPSI throttle criteria can **NOT** be maintained, **THEN**: a. Raise HPSI flow. b. Start HPSI pumps as necessary.	
LPSI Pump Stop Criteria	
* 8. **IF** pressurizer pressure is greater than 200 psia and controlled, **THEN**: a. Stop the LPSI pumps. b. Close the LPSI flow control valves.	
LPSI Pump Restart Criteria	
* 9. **IF** pressurizer pressure lowers to less than 200 psia and is **NOT** controlled **AND** RAS has **NOT** occured, **THEN** perform **BOTH** of the following as necessary: a. Start the LPSI pumps. b. Open the LPSI flow control valves.	

SAFETY FUNCTION: **RCS Inventory Control**
SUCCESS PATH: **IC-2: Safety Injection**
RESOURCE TREE: **Tree C**

INSTRUCTIONS	CONTINGENCY ACTIONS
Check IC-2 Acceptance Criteria	
* 15. Check IC-2, Safety Injection is satisfied by **ALL** of the following: a. **IF** RAS has **NOT** initiated, **THEN** at least one charging pump is operating. b. **IF** RAS has **NOT** initiated, **AND** LPSI pump stop criteria are **NOT** met, **THEN** LPSI flow within Appendix 2-F, "LPSI Flow Curve". c. **IF** HPSI throttle criteria are **NOT** met, **THEN** HPSI flow within Appendix 2-E, "HPSI Flow Curve". d. RVLMS LEVEL PLENUM indicates greater than or equal to 20%.	15.1 **IF** the RCS Inventory Control safety function is still **NOT** met, **THEN** GO TO RCS Inventory Control Continuing Actions.

SAFETY FUNCTION: RCS Inventory Control
SUCCESS PATH: IC Continuing Actions
RESOURCE TREE: Tree C

INSTRUCTIONS	CONTINGENCY ACTIONS

Support Restoration of Success Paths

* 3. Restore the RCS Inventory safety function to a success path by performing **ANY** of the following:

a. Restore the vital auxiliaries necessary to operate components or systems in the success path.

b. Manually operate alternate components to implement a success path.

c. **IF** the **BOTH** LPSI pumps are not available, **AND** the TSC concurs, **THEN** REFER TO Appendix 27, "Aligning CS to Replace LPSI", and align a CS Pump.

Evaluate Further Actions

* 4. Evaluate further actions based on the urgency of other safety functions that are **NOT** met.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	062 A2.12	
	Importance Rating		3.6

K/A Statement

A2.12 - Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Restoration of power to a system with a fault on it

Proposed Question: SRO 15 Rev: 0

Given:

- Plant is at 100% power
- A loss of the 3B Safety bus has occurred

All indications for this condition are normal with the following exceptions:

- Auxiliary Cooling Water Pump (ACCW) B tripped at the 17 second load block
- All Emergency Diesel Generator (EDG) B Sequencer load block lights have extinguished
- EDG B Sequencer LOCKOUT light is LIT

The ____(1)____ de-energized. The CRS will direct the crew to ____(2)____.

	(1)	(2)
A.	Train B loads only after block 17 are	secure EDG B IAW OP-009-002, Emergency Diesel Generator
B.	Train B loads only after block 17 are	rack down ACCW Pump B breaker IAW OP-901-311, Loss of Train B Safety Bus
C.	Entire 3B safety bus is	rack down ACCW Pump B breaker IAW OP-901-311, Loss of Train B Safety Bus
D.	Entire 3B safety bus is	secure EDG B IAW OP-009-002, Emergency Diesel Generator

Proposed Answer: C

Explanation: (Optional)

A. Incorrect: All loads on the 3B Bus are de-energized on a Sequencer Lockout condition, not just the loads after the lockout. OP-009-002 is the normal operating procedure for the EDG and sequencer. Plausible because there are other instances when portions of normal operating procedures are used when in off normal and emergency procedures. OP-009-002 has steps for securing an EDG, OP-901-311 does not
B. Incorrect: All loads on the 3B Bus are de-energized on a Sequencer Lockout condition, not just the loads after the lockout. OP-901-311 contains the actions required to reset a sequencer lockout even though a SLO is not expected on a loss of the 3 bus. Racking down the ACCW pump is the first step in restoring from a SLO.
C. **CORRECT**: All loads on the 3B Bus are de-energized on a Sequencer Lockout condition, not just the loads after the lockout. OP-901-311 contains the actions required to reset a sequencer lockout even though a SLO is not expected on a loss of the 3 bus. Racking down the ACCW pump is the first step in restoring from a SLO, this step is in OP-901-311.
D. Incorrect: All loads on the 3B Bus are de-energized on a Sequencer Lockout condition, not just the loads after the lockout. OP-009-002 is the normal operating procedure for the EDG and sequencer. Plausible because there are other instances when portions of normal operating procedures are used when in off normal and emergency procedures.

Technical Reference(s): OP-901-311 revision 309
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-SEQ00 obj. 2 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Undervoltage Override

An undervoltage override (UVO) feature prevents cycling of the sequencer due to momentary drops in the safety bus voltage when various safety transformers are being connected to the bus. It is active only if the 3-to-2 bus tie breaker, 3A-11, is open (EDG is only power supply to the bus), and even then, it is only active during the time interval between the 0.5-second and 17-second load blocks. When the sequencer initiates, the UVO relay is energized at step 0 (S0X, 0.5 seconds). It opens contacts to de-energize the 4160V and 480V bus auxiliary relays (27-1X, 27-2X, 27-3X, 27-11X, 27-21X, and 27-31X in the figures). This blocks any subsequent undervoltage/degraded voltage actuation signals, in the event bus voltage should again drop below the undervoltage/degraded voltage actuation setpoints during bus loading. It remains energized until step 4 (S4X), 17 seconds after the EDG re-energizes the bus. At this time, the UVO relay de-energizes and no longer prevents subsequent undervoltage/degraded voltage actuation signals.

Sequencer Lockout

A sequencer lockout (SLO) feature will terminate automatic loading of the EDG in the event EDG voltage cannot be maintained. This will most likely be due to an equipment malfunction. Like the undervoltage override feature, the sequencer lockout feature is only utilized when the EDG is supplying the bus alone (3-to-2 tie breaker 3A-11 open). Since the UVO feature blocks undervoltage signals during the time interval between 0.5 to 17 seconds, the SLO circuit is actually only enabled during the interval between 17 and 200 seconds. After step 4 (17 seconds), if 3/3 phases on either a 4.16 KV or 480 VAC safety bus have undervoltage/degraded voltage conditions, then a UV contact will close (27-1 for 4160V; 27-3 for 480V) to energize the SLO relay. When the SLO relay energizes, one SLO contact opens to de-energize (reset) all TDPU timer relays, one closes to turn on the LOCKOUT indicating light on CP-1, and one closes to seal-in the SLO relay through a RESET-TEST switch reset contact. Once the SLO circuit is actuated, it can only be reset manually from CP-1 by momentarily taking the sequencer RESET-TEST switch to the RESET position after the UV condition has cleared. This will drop out the SLO seal-in contact and de-energize the SLO reset to allow sequencer actuator to begin again. Thus, if an undervoltage/degraded voltage condition occurs after the undervoltage override times out, the loads previously loaded onto the bus will be respired (load shed) due to the undervoltage/degraded voltage relays, and the sequencer lockout feature will reset all timer relays to time 0 and prevent automatic sequencing from continuing until manually reset.

The 2A-to-3A tie breakers operate as follows: (CWDs 2231, 2277)

- OPEN on any of the following conditions:
  - UV (3/3 coincidence to actuate 86A2) on the 2A bus
  - Control switch to OPEN
  - Local pushbutton and control switch operation
  - Overcurrent (1/3 coincidence) on the 2A-to-3A tie bus (also actuates 74/HR)
- CLOSE when all of the following conditions are satisfied:

No UV (1/3 coincidence) on 2A bus

AND
- 86A2 reset

  AND
- 74/HR reset

  AND
- Control switch placed to CLOSE.

Automatic Tripping and Loading of Buses (CWDs 2337, 2338)

4.16 KV buses 3A and 3B have been provided with undervoltage relays to monitor the voltage condition on these buses. If a sustained degraded voltage condition or loss of voltage on either of these buses is sensed by its time-delayed undervoltage relays, all loads except the tie to swing bus 3AB and the 31A(B) are shed, the tie breaker opens automatically disconnecting the offsite source, and the EDG is started automatically. The 4.16 KV undervoltage relays are shown in Figures 9 and 10.

NOTE
To eliminate confusion, this discussion will cover the A Train breaker interlocks only. The B Train breakers operate similarly.

CWDs 2337 and 2338 show the 4.16 KV bus 3A UV relay arrangement. The various 4.16 KV bus 3A UV relays and their functions are as follows:

- <u>27-1/A3, 27-2/A3, 27-3/A3</u> - These relays are “loss of voltage” (LOV) relays. Designed to de-energize on a loss of bus voltage, these relays may take as long as 9 seconds (at 79.5% of rated voltage) or as little as 2 seconds (at 0% of rated voltage) to operate. Figure 11 displays the sliding scale at which these relays provide this protection.

  When the loss of voltage relays de-energize, they close contacts which energize relays 27-1X, 27-2X, 27-3X, respectively. Since these relays require power to energize when bus voltage is lost, they are powered from a 125 VDC supply.

- 27-1E/A3, 27-2E/A3, 27-3E/A3 - These relays are the "degraded voltage" (DGV) relays. They also sense bus voltage (one relay per phase) directly and are normally energized. If bus voltage drops below 93.1% of rated voltage for at least 12.5 seconds, they de-energize to energize relays 27-1X, 27-2X, 27-3X, respectively.
- 27-1X, 27-2X, 27-3X - These UV relays energize on bus LOV or DGV via the LOV/DGV relays. These relays basically provide for "contact multiplication" and activate most of the protection, control, and indication functions as follows:
  - Close contacts in series with corresponding 480 VAC UV contacts (27-11X, 27-21X, 27-31X). If both 4.16KV and 480 VAC UV conditions exist, these series contacts will energize three white indicating lights (phases A, B, C) on train A 4KV/480V PRIMARY UV RELAY TEST switch on CP-1 to indicate phase UV conditions.
  - Close contacts to energize three white indicating lights (phases A, B, C) on train A 4KV BACKUP UV RELAY TEST switch on CP-1 to indicate phase UV conditions.
  - Open parallel contacts (3/3 coincidence) to de-energize 1-second TDDO relay 27-1 [causes 4KV BUS SA VOLTAGE LOST alarm (P1, Cabinet D)].
  - Open series contacts (1/3 coincidence) to de-energize 2-second TDDO relay 27-2 (causes 4KV BUS SA FUSE BLOWN alarm (Q1, Cabinet D) if relay 27-1 is energized; that is, UV on only one or two phases but not all three).
  - Operates contacts in various circuits to perform the following functions (3/3 phase UV) (CWD 2338):
    - •Trip SST 32A feeder breaker
    - •Trip SST 315A feeder breaker
    - •Trip 3A-to-2A bus tie breaker
    - •Set up close permissive for 3A-to-2A bus tie breaker
    - •Reset load sequencer timer relays to "time 0"
    - •Emergency start EDG A (two separate start signals)
    - •Trip EDG A breaker if bus tie is energized (62BT)
    - •Set up close permissive to EDG A breaker circuit
    - •Trip HPSI Pump A (UV load shed)
    - •Trip LPSI Pump A (UV load shed)
    - •Trip CCW Pump A (UV load shed)
    - •Trip Containment Spray Pump A (UV load shed)
    - •Trip Water Chiller Compressor WC-1 (UV load shed)
    - •Trip EFW Pump A (UV load shed)
    - •Trip ACCW Pump A (UV load shed)

- •Set up start permissive for ACCW Pump A
- •Trip RAB Normal Exhaust Fan E-22

- 27-1/AA, 27-2/AA - These relays sense 2A-to-3A bus tie voltage (C-A and B-A phase-to-phase voltages, respectively) and are normally energized. They de-energize on tie bus UV and operate contacts to perform the following functions (coincidence):
  - Annunciate a BUS TIE A TO SA PT FUSE BLOWN alarm (M1, Cabinet D) if bus tie breaker 3A-11 is closed and either of these relays is de-energized. (1/2)
  - Remove a close permissive for tie breaker 3A-11 (1/2).
  - Trip tie breaker 3A-11 on bus tie UV (2/2).
  - Disallow manual closure of SST 32A feeder breaker (1/2).
  - Provide tie bus UV signal to EDG A Control Panel (1/2)

The design basis for operation of these relays is shown in Table 2. A complete loss of offsite power will result in approximately a two second delay in a loss of voltage signal (LOVS) actuation. The diesel starts and is available to accept loads in ten seconds. For a complete discussion on diesel load sequencing see the Emergency Diesel Generator System Description (SD-EDG).

SAFETY RELATED BUS 3AB

The design criterion governing the assignment of redundant loads (the third high pressure safety injection pump, the third component cooling water pump and the third essential chiller, and associated valves) is to ensure the availability of one component in each train during extensive maintenance. The third of a kind equipment consisting of the installed spares may be utilized by connecting bus 3AB to bus 3A or 3B (i.e.,to the bus that has a component requiring extensive maintenance). The reassignment of loads on the 3AB bus requires a "dead bus" transfer. It is therefore not a normal practice to transfer the 3AB bus because the momentary deenergization of the bus results in a temporary loss of various auxiliary components.

3A(B) TO 3AB BUS TIES
(4KV-EBKR-3A-1 and 4KV-EBKR-3B-1)

Refer to Figure 8 for a diagram of the bus tie breaker arrangement. Four tie breakers are provided to align 4.16 KV buses 3A(B) to supply swing bus 3AB, two series breakers from 3A to 3AB and two series breakers from 3B to 3AB. In the discussion below, the A(B) side breakers will be called the "outside" breakers, and the AB side breakers will be called the "inside" breakers. The only electrical protection for these breakers is overcurrent. The ties from buses 3A(B) to the standby bus 3AB consist of 1200-amp nonsegregated phase bus ducts.

PLACEKEEPER		
START	DONE	N/A

NOTE

(1) Loading additional components onto a degraded bus is prevented by a Sequencer Lockout circuit which performs the following:

- Stops Sequencer
- Illuminates LOCKOUT light
- Extinguishes all Load Block Lights

(2) Emergency Diesel Generator Ratings are 4.4 MW for continuous loading, not to exceed 4.84 MW for 2 hours out of any 24 hour period.

Step	START	DONE	N/A
5. If Emergency Diesel Generator B is running and a Sequencer Lock Out condition occurs, then locally check all breakers on BUS B3 and Bus B31 for any abnormal conditions and Rack Out all breakers which were observed to have protection flags actuated.	☐	☐	☐
5.1 When all breakers have been racked out as required, then place B Sequencer control switch to RESET.	☐	☐	

Sequencer Lockout

- After 200 seconds of sequencer operation the SLO feature is no longer active
- If SLO occurs the UV/DV relays will strip all required loads and the LOCKOUT light will illuminate on the control board. No loads will then sequence onto the bus from the sequencer. The operator would be required to RESET the sequencer after the fault condition is identified to restart the sequencer

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034 A2.02	
	Importance Rating		3.9

K/A Statement

A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped Cask.

Proposed Question: SRO 16 Rev: 0

The following plant conditions exist:

- The plant is MODE 6 and dry fuel operations are in progress
- Fuel Handling Building Normal Ventilation Train A is in service.
- Two minutes ago the crew received a report of a dropped dry fuel cask in the Fuel Handling Building
- A Fuel Handling Accident (FHA) Signal has not yet been generated
- Crew has implemented OP-901-405, Fuel Handling Incident.

Based on these conditions, the crew should expect the ____(1)____ to detect any initial airborne activity rise and should start the ____(2)____ following the FHA signal.

	(1)	(2)
A.	Fuel Handling Building PIG A Radiation Monitor	Fuel Handling Building HV Room Exhaust Fan A
B.	Fuel Handling Building PIG A Radiation Monitor	FHB WRGM sample pump
C.	Fuel Handling Building WRGM	Fuel Handling Building HV Room Exhaust Fan A
D.	Fuel Handling Building WRGM	FHB WRGM sample pump

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Part 1 is correct. After the FHA signal is generated, the FHB Normal Ventilation will trip and the Emergency Ventilation will auto start, which includes the Fuel Handling Building HV Room Exhaust Fan A or B.
B. **CORRECT:** Before the FHA signal is generated, the FHB Normal Ventilation will be running. The FHB PIG A will be monitoring the FHB Exhaust. After the FHA signal is generated, the FHB Normal Ventilation will trip and the Emergency Ventilation will auto start, which includes the Fuel Handling Building HV Room Exhaust Fan A or B. The FHB WRGM monitors this ventilation, but its sample pump does not auto start. OP-901-405 directs starting the FHB WRGM sample pump. The FHB WRGM will detect rising airborne activity, but not until the sample pump is started.
C. Incorrect: Before the FHA signal is generated, the FHB Normal Ventilation will be running. The FHB PIG A will be monitoring the FHB Exhaust. After the FHA signal is generated, the FHB Normal Ventilation will trip and the Emergency Ventilation will auto start, which includes the Fuel Handling Building HV Room Exhaust Fan A or B. The FHB WRGM monitors this ventilation, but its sample pump does not auto start. OP-901-405 directs starting the FHB WRGM sample pump. The FHB WRGM will detect rising airborne activity, but not until the sample pump is started.
D. Incorrect: **:** Before the FHA signal is generated, the FHB Normal Ventilation will be running. The FHB PIG A will be monitoring the FHB Exhaust. Part 2 is correct.

Technical Reference(s): OP-901-405 page 7, 8 and 12 revision 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO4 obj. 3 (As available)

Question Source: Bank # X Question #17
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2009 NRC SRO Exam

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

10 CFR Part 55 Content: 55.41
55.43 7

Comments:

C AUTOMATIC ACTIONS (cont'd)

2. IF incident occurred in Spent Fuel Pool area AND radiation setpoints are reached, THEN the following occur:

   a. Operating FHB Normal Exhaust Fan stops:
      - A FHB NORMAL VENT EXH FAN (HVF 0003A)
      - B FHB NORMAL VENT EXH FAN (HVF 0003B)

   b. FHB NORMAL VENT SUPPLY FAN (HVF 0002) stops

   c. FHB Sply Fan Inlet Damper (HVF 101) [PID D53641 OR Mimic HVF1] closes

   d. FHB Sply Fan Outlet Damper (HVF 102) [PID D53668 OR Mimic HVF1] closes

   e. For Train A Radiation Monitors (300.2 OR 300.4):

      - FUEL HANDLING VENT MODE SELECT in BYPASS
        - HVF 103 closed
        - HVF 110 open

      - FUEL HANDLING BUILDING ISOL DAMPERS close:
        - HVF 105
        - HVF 108

      - FUEL HANDLING BUILDING EMERGENCY FLTN (HVF 0005A) starts

      - FUEL HANDLING BUILDING HV ROOM EXH FAN (HVF 0006A) starts

      - FHB H&V Room Trn A Outside Air Intake Louver (HVF 301A) [PID D53664 OR Mimic HVF2] opens

C AUTOMATIC ACTIONS (cont'd)

2. (cont'd)

   f. For Train B Radiation Monitors (300.1 OR 300.3):

      - FUEL HANDLING BUILDING VENT MODE SELECT in BYPASS
         - HVF 104 closed
         - HVF 109 open
      - FUEL HANDLING BUILDING ISOL DAMPERS close
         - HVF 106
         - HVF 107
      - FUEL HANDLING BUILDING EMERGENCY FLTN (HVF 0005B) starts
      - FUEL HANDLING BUILDING HV ROOM EXH FAN (HVF 0006B) starts
      - FHB H&V Room Trn B Outside Air Intake Louver (HVF 301B) [PID D53666 OR Mimic HVF2] opens.

E0 GENERAL (CONT'D)

	PLACEKEEPER		
	START	DONE	N/A
8. IF FHB EFU is running, THEN Start FUEL HANDLING BUILDING WRGM EMERGENCY EXHAUST (PRM-IRI-3032) Sample Pump flow at CP-6 OR CP-52.	☐	☐	☐
9. Monitor radioactive release through FHB Emergency Ventilation Exhaust.	☐	☐	☐
10. IF ANY of the following Radiation Monitors reach Hi-Hi alarm setpoint, OR a rising trend indicates Hi-Hi alarm is imminent, THEN verify ALL of the following valves are closed:	☐	☐	☐

- Radiation monitors:
  - Containment Purge Isolation A (5025 OR 5026)
  - Plant Stack A (100.1)
  - Containment Purge Isolation B (5024 OR 5027)
  - Plant Stack B (100.2)
- Isolation valves:
  - CONTAINMENT ATMOS RELIEF CNTMT PRESS EXH INLET (CAR 200B)
  - CONTAINMENT ATMOS RELIEF EXH HDR B UPSTREAM ISOL (CAR 202B)
  - CNTMT ATMOS PURGE MAKEUP VALVES (CAP 103/CAP 104)
  - CNTMT ATMOS PURGE EXH ISOL VALVES (CAP 205)
  - CNTMT ATMOS PURGE MAKEUP VALVES (CAP 102)
  - CNTMT ATMOS PURGE EXH ISOL VALVES (CAP 203/CAP 204)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	001 2.2.12	
	Importance Rating		4.1

K/A Statement

2.2.12 - Knowledge of surveillance procedures.

Proposed Question: SRO 17 Rev: 0

Given:

- Plant is at 100% power
- Control Rod testing is in progress in accordance with OP-903-005, Control Element Assembly Operability Check
- CEA #28 in Shutdown Bank A is being tested.
- The initial position for CEA #28 is fully withdrawn at 149.5 inches.

During the surveillance testing for CEA #28, entry into TS 3.1.3.5, Shutdown CEA Insertion Limit ____(1)____ be required. The basis for the Shutdown CEA insertion limit is to _____(2)_____.

	(1)	(2)
A.	will	prevent axial Xenon redistribution
B.	will	maintain minimum shutdown margin
C.	will not	maintain minimum shutdown margin
D.	will not	prevent axial Xenon redistribution

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Part 1 is correct. The basis for TS 3.1.3.5 is to maintain minimum shutdown margin. The basis for TS 3.1.3.6 (regulating group insertion limits) is Xenon redistribution.
B. **CORRECT:** Per OP-903-005 and surveillance requirement 4.1.3.1.1, operability of the CEA is determined by movement of at least 5 inches in either direction. With an initial position of 149.5 inches withdrawn, an insertion of 5 inches will insert CEA #28 less than 145 inches withdrawn and require TS 3.1.3.5 entry. The basis for TS 3.1.3.5 is to maintain minimum shutdown margin.
C. Incorrect: Per OP-903-005 and surveillance requirement 4.1.3.1.1, operability of the CEA is determined by movement of at least 5 inches in either direction. With an initial position of 149.5 inches withdrawn, an insertion of 5 inches will insert CEA #28 less than 145 inches withdrawn and require TS 3.1.3.5 entry. The basis for TS 3.1.3.5 is to maintain minimum shutdown margin.
D. Per OP-903-005 and surveillance requirement 4.1.3.1.1, operability of the CEA is determined by movement of at least 5 inches in either direction. With an initial position of 149.5 inches withdrawn, an insertion of 5 inches will insert CEA #28 less than 145 inches withdrawn and require TS 3.1.3.5 entry. The basis for TS 3.1.3.5 is to maintain minimum shutdown margin. The basis for TS 3.1.3.6 (regulating group insertion limits) is Xenon redistribution

Technical Reference(s): OP-903-005 revision 13
(Attach if not previously provided) TS 3.1.3.5
(including version/revision number) TS 4.1.3.1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CED00 obj. 8 (As available)

Question Source:
Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ____
55.43 2

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1 Entry into Technical Specification 3.1.3.5 is required in Mode 1 for any shutdown CEA inserted below 145 inches withdrawn.

3.2 Limitations

3.2.1 Regulating Group 6 CEAs should not be inserted below 120 inches.

3.2.2 The time that any CEA is misaligned > 7 inches from any other CEA in its group shall be minimized. If a CEA cannot be restored to within 7 inches of its group, refer to Tech Spec 3.1.3.1.

6.0 ACCEPTANCE CRITERIA

6.1 Each CEA not fully inserted shall be determined to be Operable by movement of at least 5 inches in any direction.

7.0 PROCEDURE

7.1 CEA OPERABILITY CHECK

CAUTION

RX THE FOLLOWING SECTION HAS BEEN DETERMINED TO HAVE THE POTENTIAL TO AFFECT CORE REACTIVITY.

NOTE

The following alarms will be initiated if a CEA deviates more than 5.5 inches inward or outward from any other CEA in its subgroup:

- CEA Channel B Deviation (H-12, Cabinet K)
- CEA Channel C Deviation (H-13, Cabinet K)
- Prepower Dependent Insertion Limit (H-9, Cabinet H)

CAUTION

(1) IF ANY CEA IS NOTED TO BE IMMOVABLE, MISALIGNED BY >19 INCHES, OR DROPS AT ANY TIME DURING PERFORMANCE OF THIS TEST, THEN GO TO OP-901-102, CEA OR CEDMCS MALFUNCTION.

(2) ENTRY INTO TECHNICAL SPECIFICATION 3.1.3.5 IS REQUIRED IN MODE 1 FOR ANY SHUTDOWN CEA INSERTED BELOW 145 INCHES WITHDRAWN.

(3) IF A REGULATING CEA OR A GROUP P CEA IS INSERTED OUT OF SEQUENCE <140 INCHES, THEN A GROUP OUT OF SEQUENCE ANNUNCIATOR (A-7, CABINET L) AND A TARGETED CPC CHANNEL TRIP WILL OCCUR.

(4) THE TIME THAT ANY CEA IS MISALIGNED >7 INCHES FROM ANY OTHER CEA IN ITS GROUP SHALL BE MINIMIZED. IF A CEA CANNOT BE RESTORED TO WITHIN 7 INCHES OF ITS GROUP, THEN REFER TO TECH SPEC 3.1.3.1.

(5) WHILE CEAS MUST BE INSERTED TO <145 INCHES TO SATISFY THE REQUIRED 5 INCHES MOVEMENT, THE PERIOD OF TIME THIS DEVIATION EXISTS SHALL BE MINIMIZED. IF A CEA CANNOT BE RESTORED TO >145 INCHES, THEN REFER TO T.S. 3.1.3.1 AND T.S. 3.1.3.6.

7.1.1 Obtain Shift Manager/Control Room Supervisor permission to begin test and document on Attachment 10.1, CEA Exercise Data Sheet.

7.1.2 Verify Departure from Nucleate Boiling Ratio (DNBR) within limits of Technical Specification 3.2.4, and document on Attachment 10.1.

RX 7.1.3 For each CEA inserted ■5 inches for Axial Shape Index control, perform the following on Attachment 10.1:

7.1.3.1 Record CEA position prior to ASI control as Initial Position.

7.1.3.2 Record present CEA Position as Test Position.

7.1.3.3 Document CEA movement ■5 inches.

7.1.3.4 Record NA for Final Position.

7.1.4 On Attachment 10.1, check the CEAC that will be used for indication.

7.1.5 Perform Attachment 10.2 for each CEA inserted <5 inches for Axial Shape Index control:

7.1.6 If every CEA has been parked at their final position, then perform CEA Initialization per OP-004-012, Plant Computer System.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 145 inches.

APPLICABILITY: MODES 1 and 2*#**.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 145 inches withdrawn, within 1 hour either:

a. Withdraw the CEA to greater than or equal to 145 inches, or

b. Declare the CEA inoperable and determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to greater than or equal to 145 inches withdrawn:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups or group P during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

#With Keff greater than or equal to 1.0.

**Except for surveillance testing pursuant to Specification 4.1.3.1.2.

MOVABLE CONTROL ASSEMBLIES (Continued)

continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The arithmetic average CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. The maximum CEA drop time restriction limits the CEA drop time distribution about the average to that used to support the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the meausured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. The CEA drop time restriction is representative of the design and operating conditions for Cycle 3 and reverification may be required for (1) any fuel management change that significantly affects the core wide axial or radial power profiles, and (2) any mechanical, flow, control, or CEA location changes that would significantly affect the CEA drop time distribution.

The establishment of LSSS and LCOs requires that the expected long and short-term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base loaded, or load maneuvering) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors. Insertion of Reg. Groups 5 and 6 is permitted to be essentially tip-to-tip within the limits imposed by the

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	045 A2.15	
	Importance Rating		2.6

K/A Statement

A2.15 - Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Turbine overspeed

Proposed Question: SRO 18 Rev: 0

Given:

- Plant is at 100% power
- The AUTO ACTUATE OUT OF SERVICE pushbutton on the RXC module is **not** illuminated.

A Turbine Trip occurred due to an overspeed condition

- The following annunciator is illuminated associated with the Main Turbine:
  - o Turbine Trip Overspeed/DEH Power Lost

The CRS will direct the crew to ____(1)____ and perform ____(2)____.

	(1)	(2)
A.	verify a reactor power cutback	OP-901-101, Reactor Power Cutback only
B.	verify a reactor power cutback	OP-901-101, Reactor Power Cutback and OP-901-210, Turbine Trip concurrently
C.	trip the reactor	OP-902-000, Standard Post Trip Actions and OP-901-210, Turbine Trip concurrently
D.	trip the reactor	OP-902-000, Standard Post Trip Actions only

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: A RXC would occur given that the AUTO ACTUATE OUT OF SERVICE pushbutton on the RXC module is not illuminated. OP-901-210 directs the crew to perform OP-901-101, Reactor power cutback concurrently with OP-901-210, Turbine Trip.
B. **CORRECT**: A RXC would occur given that the AUTO ACTUATE OUT OF SERVICE pushbutton on the RXC module is not illuminated. OP-901-210 directs the crew to perform OP-901-101, Reactor power cutback concurrently with OP-901-210, Turbine Trip.
C. Incorrect: A RXC would occur. The CRS would order a Reactor Trip per OP-901-210 if the turbine tripped due to a loss of Main Lube Oil or high vibration (Reactor trip is not required for an overspeed). The annunciators given do not indicate a loss of Main Lube Oil or high vibration.
D. Incorrect: A RXC would occur. The CRS would order a Reactor Trip per OP-901-210 if the turbine tripped due to a loss of Main Lube Oil or high vibration (Reactor trip is not required for an overspeed). The annunciators given do not indicate a loss of Main Lube Oil or high vibration. This answer would be correct if the applicant decides that a reactor trip is required for an overspeed of the main turbine.

Technical Reference(s): OP-004-015 pg 9 revision 10
(Attach if not previously provided) OP-901-210 revision 5
(including version/revision number) OP-901-101, revision 8

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO20 obj. 3 (As available)

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

B SYMPTOMS

B_1 ALARMS

- GENERATOR 86 RELAY TRIP (Cabinet D, G-7).
- 4.16KV BUS 3A3-S VOLTAGE LOW (Cabinet D, S-7)
- 4.16KV BUS 3B3-S VOLTAGE LOW (Cabinet D, S-8)
- TURBINE TRIP VACUUM LOST (Cabinet E, A-1)
- TURBINE TRIP OVERSPEED/DEH PWR LOST (Cabinet E, A-2)
- TURBINE TRIP THRUST BRNG FAILURE (Cabinet E, A-3)
- TURBINE TRIP LOSS OF LOAD (Cabinet E, A-4)
- TURBINE TRIP STR COIL WATER LOST (Cabinet E, C-4)
- TURBINE TRIP OIL LVL LO-LO (Cabinet E, A-5)
- TURBINE TRIP BRNG OIL PRESS LO (Cabinet E, B-5)
- TURBINE TRIP EXHAUST TEMP HI (Cabinet E, A-6)
- TURBINE TRIP DIFF EXP/VIBR HI (Cabinet E, C-6)
- TURBINE TRIP SEAL OIL TO H2 DP LO (Cabinet E, A-7)
- TURBINE TRIP MSR A LEVEL HI (Cabinet E, B-8)
- TURBINE TRIP HTR 5A LVL HI-HI (Cabinet E, C-8)
- TURBINE TRIP HTR 6A LVL HI-HI (Cabinet E, D-8)
- TURBINE TRIP MSR B LEVEL HI (Cabinet E, B-9)
- TURBINE TRIP HTR 5B LVL HI-HI (Cabinet E, C-9)
- TURBINE TRIP HTR 6B LVL HI-HI (Cabinet E, D-9)
- TURBINE TRIP HTR 5C LVL HI-HI (Cabinet E, C-10)
- TURBINE TRIP HTR 6C LVL HI-HI (Cabinet E, D-10)

C AUTOMATIC ACTIONS

1. Steam Bypass valves, Atmospheric Dump valves, OR Main Steam Safety valves open.

2. The following breakers trip:
   - Generator Breaker A
   - Generator Breaker B
   - Exciter Field Breaker

3. All Turbine Throttle, Governor, Reheat and Intercept Valves close.

4. Electrical loads shift to Startup Transformers.

5. If Reactor Power Cutback System is in service and Load Rejection is selected for cutback, then a Reactor Power Cutback will occur.

6. If Reactor Trip on Turbine Trip is enabled >65% power, then the Reactor will trip.

7. Heater Drain Pumps trip.

E SUBSEQUENT OPERATOR ACTIONS

E_0 GENERAL

	PLACEKEEPER	
START	DONE	N/A

NOTE

High vibration is defined for bearings #1 through #10 as a sustained vibration above 10 mils. Due to lower bearing loads, high vibration on Bearing #11 would not require breaking condenser vacuum unless it affected the adjacent bearing (#10). **[ER-W3-2002-0324-001]**

Step	START	DONE	N/A
1. If a Reactor trip occurs, then perform either of the following:	☐	☐	☐
1.1 If turbine trip was due to loss of Main Lube Oil or high vibration, then go to step E0.6.		☐	☐
1.2 If turbine trip was due to any reason other than those listed in step E0.1.1 above, then go to OP-902-000, Standard Post Trip Actions.		☐	☐
2. If the Turbine tripped due to loss of Main Lube Oil or high vibration, then go to step E0.6.		☐	☐
3. If a Reactor Power Cutback occurred, then perform OP-901-101, REACTOR POWER CUTBACK, concurrently with this procedure.	☐	Continuous ⬇	☐
4. If the turbine tripped due to loss of Seal Oil, then go to OP-901-211, Generator Malfunction.		☐	☐
5. If turbine trip was due to any reason other than those listed in step E0.2, then go to step E0.7.		☐	☐

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.5	
	Importance Rating		3.9

K/A Statement

Conduct of Operations: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question: SRO 19 Rev: 0

Given:

- Reactor is shutdown with all CEAs fully inserted
- RCS Boron concentration is 2100 ppm
- Keff is 0.945
- RCS temperature is 195°F

Which of the following correctly describes the minimum required shift staffing in accordance with Tech Spec 6.2.2?

A. 1 SM, 1 SRO, 2 ROs, 1 STA and 2 NAOs

B. 1 SM, 1 RO, and 1 NAO

C. 1 SM, 1 RO, 1 STA and 1 NAO

D. 1 SM, 1 SRO, 2 ROs and 2 NAOs

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: This crew composition exceeds the requirement of T.S. 6.2.2, Table 6.2-1 for MODE 5 by 1 SRO, 1 RO and 1 STA.

B. **CORRECT**: This crew composition meets the requirement of T.S. 6.2.2, Table 6.2-1 for MODE 5.

C. Incorrect: This crew composition exceeds the requirement of T.S. 6.2.2, Table 6.2-1 for MODE 5 by 1 STA.

D. Incorrect: This crew composition exceeds the requirement of T.S. 6.2.2, Table 6.2-1 for MODE 5 by 1 SRO and 1 NAO.

Technical Reference(s): Technical Specification 6.2.2
(Attach if not previously provided) Table 6.2-1 Minimum Shift Crew Composition
(including version/revision number) Technical Specification table 1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-TS03, Objective 6 (As available)

Question Source: Bank # X WF3-OPS-7256-A
Modified Bank # ______
New ______

Question History: Last NRC Exam 2012 SRO NRC Exam

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

6.1 RESPONSIBILITY

6.1.1 The General Manager Plant Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The General Manager Plant Operations or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that may affect nuclear safety.

6.1.2 The Shift Superintendent, or during his absence from the control room, a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President Operations, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The General Manager Plant Operations shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

UNIT STAFF (Continued)

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room.

c. A Health Physics Technician* shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. The Operations Manager or the Assistant Operations Manager (Shift) shall hold a senior reactor operator license.

*This requirement tolerates Health Physics Technician unexpected absences for not more than 2 hours, provided management takes immediate action to fill the required Health Physics Technician position.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SM	1*	1
SRO	1*	None
RO	2	1
AO	2	1
STA	1*	None

SM - Shift Manager with a Senior Operator License
SRO - Individual with a Senior Operator License
RO - Individual with an Operator License
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Manager, the shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Manager from the control room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Manager from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the control room command function.

*An individual with SRO/STA qualifications can satisfy the SM/STA or SRO/STA position requirements simultaneously.

OPERATIONAL MODES

OPERATIONAL MODE	REACTIVITY CONDITION, K_{eff}	% OF RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F> T_{avg}>200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.45	
	Importance Rating		4.3

K/A Statement

Conduct of Operations: Ability to identify and interpret diverse indications to validate the response of another indication.

Proposed Question: SRO 20 Rev: 1

Regarding EOP usage, to verify the RCS **is not** water solid with Pressurizer level indicating 100%, the crew will verify ____(1)____ . If the RCS **is** considered water solid, the crew will _____(2)_____.

	(1)	(2)
A.	RCS inventory or temperature changes do not produce severe pressure responses	remain in the selected Optimal Recovery Procedure and control RCS temperature and throttle HPSI flow
B.	RCS inventory or temperature changes do not produce severe pressure responses	transition to the Functional Recovery Procedure and establish a bubble in the Pressurizer
C.	single phase natural circulation criteria is met	remain in the selected Optimal Recovery Procedure and control RCS temperature and throttle HPSI flow
D.	single phase natural circulation criteria is met	transition to the Functional Recovery Procedure and establish a bubble in the Pressurizer

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT:** Optimal Recovery procedures (902-002, 004 and 007) and OI-038-000 (Operations Expectations procedure) includes guidance for verifying the RCS is not water solid. RCS inventory or temperature changes not producing severe pressure responses is listed as a diverse indication that the RCS is not water solid. Controlling RCS temperature and throttling HPSI flow is the contingency action for the RCS is water solid. Therefore the crew can remain in the selected ORP and transitioning to OP-902-008 is not required.

B. Incorrect: Part 1 is correct. Controlling RCS temperature and throttling HPSI flow is the contingency action for the RCS is water solid. Therefore, transitioning to OP-902-008 is not required.

C. Incorrect: Checking single natural circulation criteria met is a step in most ORPs but is not a listed diverse indication of RCS not water solid. RCS inventory or temperature changes not producing severe pressure responses is listed as a diverse indication that the RCS is not water solid. Part 2 is correct.

D. Incorrect: Checking single natural circulation criteria met is a step in most ORPs but is not a listed diverse indication of RCS not water solid. RCS inventory or temperature changes not producing severe pressure responses is listed as a diverse indication that the RCS is not water solid. Controlling RCS temperature and throttling HPSI flow is the contingency action for the RCS is water solid. Therefore, transitioning to OP-902-008 is not required.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)
- OP-902-004 step 29 revision 14
- OP-902-002 step 61 revision 18
- OP-902-007 step 36 revision 15
- OI-038-000 step 5.4.31 revision 7

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE04 obj. 7 (As available)

Question Source:
- Bank # ______
- Modified Bank # ______ (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 5

Comments:

INSTRUCTIONS | CONTINGENCY ACTIONS

Restore Letdown

* 28. **IF** letdown is isolated, **AND BOTH** of the following conditions exist:

- HPSI throttle criteria are met
- Letdown is needed or desired

THEN REFER TO Appendix 9, "Letdown Restoration" and restore letdown.

Verify RCS is NOT Water Solid

* 29. Check the RCS is **NOT** in a water solid condition as indicated by the following:

a. RCS inventory or temperature changes do **NOT** produce exaggerated or severe pressure responses.

b. At least one of the following conditions met:

- Pressurizer level less than 100%
- Reactor vessel head level less than 100%

29.1 Maintain the RCS within Appendix 2A-D, "RCS Pressure and Temperature Limits" by **ANY** of the following:

a. Control RCS temperature.

b. **IF** HPSI throttle criteria are met, **THEN** perform **ANY** of the following:

- Control charging and letdown
- Throttle HPSI flow

INSTRUCTIONS — CONTINGENCY ACTIONS

Restore Letdown

* 60. **IF** letdown is isolated, **AND BOTH** of the following conditions exist:

- HPSI throttle criteria are met
- Letdown is needed or desired

THEN REFER TO Appendix 9, "Letdown Restoration" and restore letdown.

Verify RCS is NOT Water Solid

* 61. Check the RCS is **NOT** in a water solid condition as indicated by the following:

a. RCS inventory or temperature changes do **NOT** produce exaggerated or severe pressure responses.

b. At least one of the following conditions met:

- Pressurizer level less than 100%
- Reactor vessel head level less than 100%

61.1 Maintain the RCS within Appendix 2A-D, "RCS Pressure and Temperature Limits" by **ANY** of the following:

a. Control RCS temperature.

b. **IF** HPSI throttle criteria are met, **THEN** perform **ANY** of the following:

- Control charging and letdown
- Throttle HPSI flow

INSTRUCTIONS | CONTINGENCY ACTIONS

Restore Letdown

* 35. **IF** letdown is isolated, **AND BOTH** of the following conditions exist:

- HPSI throttle criteria are met
- Letdown is needed or desired

THEN REFER TO Appendix 9, "Letdown Restoration" and restore letdown.

Verify RCS is NOT Water Solid

* 36. Check the RCS is **NOT** in a water solid condition, as indicated by the following:

a. RCS inventory or temperature changes do **NOT** produce exaggerated or severe pressure responses.

b. At least one of the following conditions met:

- Pressurizer level less than 100%
- Reactor Vessel Head level less than 100%

36.1 Maintain the RCS within Appendix 2A-D, "RCS Pressure and Temperature Limits" by **ANY** of the following:

a. Control RCS temperature.

b. **IF** HPSI throttle criteria are met, **THEN** perform **ANY** of the following:

- Control charging and letdown
- Throttle HPSI flow

5.4.29 HPSI Pump Minimum Flow Criteria

- None

5.4.30 HPSI Pump Restart Criteria

- If any throttle criteria is degrading rapidly, the HPSI pump may be started or valves may be opened in anticipation of exceeding the criteria.

5.4.31 HPSI Throttle Criteria

- If RCS pressure is < 1000 psia, use other parameters in addition to the subcooled margin indicators to verify subcooling.
- Water solid operations of the pressurizer should be avoided unless minimum subcooling (28°F) cannot be maintained in the RCS. If the RCS is solid, closely monitor any makeup or draining and any system heatup or cooldown, to avoid any unfavorable rapid pressure excursions.
- If Emergency Boration is in progress to meet Reactivity Control Safety Function, at least one charging pump must remain in operation unless this creates a challenge to the pressurizer safety valves.
- When a void exists in the reactor vessel and RCPs are operating, it is not possible to obtain an accurate reactor vessel liquid level indication due to the effect of the RCP induced pressure head on the RVLMS. However, inventory trending may still be discerned.
- If the operator chooses to throttle SI flow control valves and the HPSI pumps remain running, consideration should be given to the amount of time the HPSI pumps are operated on recirc.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2.2.13	
	Importance Rating		4.3

K/A Statement

Equipment Control: Knowledge of tagging and clearance procedures.

Proposed Question: SRO 21 Rev: 0

If plant conditions require the removal of a danger tag, the ____(1)____ may perform Alternate Release Authorization. One procedural condition required for this authorization is that the Tagout/Work Order Holder____(2)____.

	(1)	(2)
A.	Duty Plant Manager	will be notified immediately upon return to the site and prior to starting work
B.	SM or designee	will be notified immediately upon return to the site and prior to starting work
C.	Duty Plant Manager	is contacted per-telecom if not on site
D.	SM or designee	is contacted per-telecom if not on site

Proposed Answer: B

Explanation: (Optional)

A. Incorrect: Per EN-OP-102, Protective and Caution Tagging (step 5.17[1]). The SM or his/her designee must authorize an alternate release of a tagout. The DPM is plausible because of the importance of an alternate release and various W-3 procedures require a DPM approval. Part 2 is correct.
B. **CORRECT**: Per EN-OP-102, Protective and Caution Tagging (step 5.17[1]). The SM or his/her designee must authorize an alternate release of a tagout. This same step states that the Tagout/Work Order Holder will be notified immediately upon return to the site and prior to starting work.
C. Incorrect: Per EN-OP-102, Protective and Caution Tagging (step 5.17[1]). The SM or his/her designee must authorize an alternate release of a tagout. The DPM is plausible because of the importance of an alternate release and various W-3 procedures require a DPM approval. If a tagout holder is not on site, he/she can be contacted per tele-com and removed as a holder, but it would no longer meet the definition of an alternate release because an alternate release is such that the holder cannot be contacted.
D. Incorrect: Part 1 is correct. Per EN-OP-102, Protective and Caution Tagging (step 5.17[1]). If a tagout holder is not on site, he/she can be contacted per tele-com and removed as a tagout holder, but it would no longer meet the definition of an alternate release because an alternate release is such that the holder cannot be contacted.

Technical Reference(s): EN-OP-102 revision 16 step 5.17 [1]
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: FLP-OPS-ESOMS obj. 4 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 ____
55.43 3

Comments:

5.16 Per Telecom

[1] Tagout Holder or Work Order Holder

NOTE

Per Telecom is not an Alternate release.

IF Holder is onsite then Per Telecom is not allowed.

(a) A Tagout Holder or Work Order Holder can be signed off a Tagout or Temporary Lift, after the holder and the Operations Supervisor concurs on the evolution.

(b) The Operations Supervisor will sign off the holder.

(c) If signing a Tagout Holder off of a Tagout then perform the following:

(1) Ensure a Tagout Holder Signoff Checklist is completed by a person knowledgeable on the work that the Original Tagout holder is signed onto. (CAPR ANO-1-2010-1915)

(2) If a knowledgeable person is not available then the Operations Supervisor will fill out the Checklist for the Tagout holder per Telecom.

(d) Write a Condition Report to document the **Per Telecom release.**

(e) An exception to the note above, VY site will be allow Per Telecom signing on and off for TOH/WOH for CRD Mechanism change out and Drywell Sump Work. The Operations Manager may authorize signing on and off per telecom in advance to minimize dose in situations of high dose areas and wearing of Plastic PC's.

[2] Tagger

(a) 1st and/or 2nd Placement verifier or 1st and/or 2nd Restoration verifier can be signed for per telecom after the Tagger and the Operations Supervisor concur on the evolution. This provision is not to be used on a routine basis, but will allow work to continue when areas cannot be readily exited due to plant conditions.

(b) The Operations Supervisor will sign the designated Tagger verifications.

5.17 Alternate Release Authorization

[1] In the event that release is required and a Tagout/Work Order Holder cannot be contacted and is not on site, the release can be authorized by the Shift Manager or his/her designee under the following conditions:

(a) The Tagout/Work Order Holder is not on Site.

(b) All reasonable attempts to contact the Tagout/Work Order Holder have been made.

(c) A knowledgeable Tagout Holder has conducted a check of the job and determined the release will not be detrimental to the plant or personnel.

(d) Positive steps have been taken to ensure the Tagout/Work Order Holder will be notified immediately upon return to the site and prior to starting work.

[2] The Tagout Holder Supervisor will obtain a copy of the "Tagout Alternate Release Checklist" form from EN-OP-102-01 Attachment 9.8 and perform the following:

(a) Write a Condition Report to document the Alternate Release.

(b) If it necessary to sign a Tagout Holder off of a Tagout using the Alternate release process Then ensure the Tagout Holder Signoff Checklist EN-OP-102-01 Attachment 9.21 is completed by a person knowledgeable of the work that the Tagout holder was signed onto prior to the Tagout holder being signed off.(CAPR ANO-1-2010-1915)

(c) Enter all appropriate information related to the reasons for the Alternate Release.

(d) Ensure the Shift Manager or designee signs the Tagout Alternate Release Checklist to approve the Alternate Release.

(e) Send the Tagout Alternate Release Checklist to the Security Shift Supervisor for deactivate/capture security badge.

[3] The Operations Supervisor will sign off the Tagout/Work Order Holder.

[4] The Shift Manager will designate an individual to remove any locking devices on components affected by the Tagout release.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.2.14	
	Importance Rating		4.3

K/A Statement

Equipment Control: Knowledge of the process for controlling equipment configuration or status.

Proposed Question: SRO 22 Rev: 0

The plant is at 100%. The rubber seal has been reported torn on Door 153, a CVAS boundary door, and the door has been declared inoperable. Door 161, the other door in that airlock, is operable. Which of the following is true concerning requirements to establish CVAS Boundary operability?

A. Perform OP-903-124, CVAS Pressure Boundary Testing, with Door 153 held open and Door 161 closed.

B. Perform OP-903-124, CVAS Pressure Boundary Testing, both Door 153 and Door 161 closed.

C. Perform OP-903-124, CVAS Pressure Boundary Testing, with Door 153 closed and Door 161 held open.

D. No testing is required; only one door is required to be operable in each CVAS air lock.

Proposed Answer: A

Explanation: (Optional)

A. **CORRECT**. Per OP-100-014 this is required guidance for an inoperable air lock door.

B. Incorrect. This selection would interfere with ensuring the operable door provides sufficient isolation to ensure the CVAS fans can perform their function.

C. Incorrect. This configuration tests the inoperable door.

D. Incorrect. This plausible if the candidate assumes that no requirement exists to verify the acceptability of the remaining door.

Technical Reference(s): OP-100-014 Revision 323 page 52
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPA00 Obj. 3 (As available)

Question Source: Bank # X Question #20
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 NRC SRO Exam

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

CVAS

Component	**Applicable Mode(s)**	**Affected Systems and Tech Specs**	**Required Action(s)**
Controlled Ventilation Area System (CVAS)	1, 2, 3, 4	CVAS - 3.7.7	• If a CVAS door is declared Inoperable, then verify that the other door in that airlock is Operable by performing OP-903-124, with the Inoperable door held Open. • After the Inoperable door is repaired, then verify the Inoperable door will perform its intended safety function by performing OP-903-124 with the Inoperable door Shut and the Operable door held Open.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		3
K/A #	G2.3.4	
Importance Rating		3.7

K/A Statement

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: SRO 23 Rev: 0

The stated limit in EP-002-030, Emergency Radiation Exposure Guidelines and Controls, for Emergency Team Members chosen to perform Life Saving activities is ___(1)____ REM TEDE. These limits may be exceeded only _____(2)_____.

	(1)	(2)
A.	25	if the NRC is consulted prior to authorizing the exposure
B.	30	if the NRC is consulted prior to authorizing the exposure
C.	30	by volunteers fully aware of the risks involved
D.	25	by volunteers fully aware of the risks involved

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: 25 rem is the limit for performing life saving activities. The note prior to step 5.1 in EP-002-030 states that the NRC should be contacted before authorizing exposure in excess of 10CFR20 limits, but may not be possible in all cases.

B. Incorrect. 30 rem is the limit to the eyes for accident saving activities. This number was selected because it is close to 25 rem and is referenced in EP-002-030. The note prior to step 5.1 in EP-002-030 states that the NRC should be contacted before authorizing exposure in excess of 10CFR20 limits, but may not be possible in all cases.

C. Incorrect: 30 rem is the limit to the eyes for accident saving activities. This number was selected because it is close to 25 rem and is referenced in EP-002-030. Step 5.2.3 of EP-002-030 states that the life saving limits may only be exceeded by volunteers fully aware of the risks involved.

D. **CORRECT:** 25 rem is the limit for performing life saving activities. Step 5.2.3 of EP-002-030 states that the life saving limits may only be exceeded by volunteers fully aware of the risks involved.

Technical Reference(s): EP-002-030 section 5.1 and 5.2 revision 10
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EP02 obj. 8 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ______
55.43 4

Comments:

5.0 PROCEDURE

5.1 RADIATION EXPOSURE IN EXCESS OF 10CFR20 LIMITS

> **NOTE**
>
> To the extent practicable, Nuclear Regulatory Commission (NRC) personnel should be consulted prior to authorizing exposures in excess of 10CFR20 limits. Either the NRC Headquarters Duty Officer or Senior NRC Region IV response personnel may be contacted. It is recognized that coordination with the NRC may not be possible in all cases due to the nature of the situation.

5.1.1 The Emergency Plant Manager or EOF Emergency Director determines the need to perform specific tasks which are anticipated to result in exposure in excess of 10CFR20 limits by evaluating the risk of not performing the tasks against the anticipated exposure.

5.2 GUIDELINES FOR EMERGENCY EXPOSURE

5.2.1 Emergency Team members chosen to perform Corrective Actions for accident-mitigating activities shall not exceed the following exposure guidelines:

A. TEDE: 10 rem

B. Extremities: 100 rem

C. Thyroid: 100 rem

D. Lens of Eye: 30 rem

5.2.2 Emergency Team members chosen to perform Life Saving activities shall not exceed the following exposure guidelines except as noted in 5.2.3 below:

A. TEDE: 25 rem

B. Extremities: 250 rem

C. Thyroid: 250 rem

D. Lens of Eye: 75 rem

5.2.3 The limits in section 5.2.2 may be exceeded for Life Saving activities only by volunteers fully aware of the risks involved. (Refer to Attachment 7.1 as necessary.)

5.2.4 Emergency Team Members shall not enter any area where dose rates are unknown or unmeasurable with dose rate instruments.

5.2.5 All reasonable precautions for minimizing the radiological consequences of the emergency action shall be taken (i.e., protective clothing, respiratory protection, thyroid prophylaxis, etc.).

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.3.6	
	Importance Rating		3.8

K/A Statement

Radiation Control: Ability to approve release permits.

Proposed Question: SRO 24 Rev: 0

Given:

- Waste Condensate Tank ‘B’ is being discharged to the Circ Water system
- EFFLUENT RAD MONT SYS ACT HI-HI annunciator is in alarm on CP-36
- WASTE LIQUID RAD MONITOR TROUBLE annunciator is in alarm on CP-4
- LWM-IFRR-0647, Liquid Waste Flow Recorder, indicates rising activity level
- The automatic isolations did not automatically close as designed
- The ATC operator has closed LWM-441 and LWM-442

Based on these conditions, for Waste Condensate Tank B to be approved for discharge before the Liquid Waste Radiation Monitor is repaired, which of the following is a required action per TRM 3.3.3.10, Radioactive Liquid Effluent?

A. Complete a valve lineup to ensure that Waste Condensate Tank A is isolated from the discharge.

B. Initiate a Department Action Statement Notice requiring samples every 4 hours during discharge.

C. Initiate a Technical Specification Addendum Log to calculate release flow rate every 4 hours during the discharge.

D. Ensure release rate calculations have been verified by two technically qualified personnel prior to the discharge.

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: Independent valve lineups of the discharge path of the tank to be released must be performed.

B. Incorrect: Independent analyzed samples are required prior to the release of the tank. (Action 1a of TRM 3.3.3.10)

C. Incorrect: This is a requirement for an inoperable flow instrument. (Action 2 of TRM 3.3.3.10)

D. **CORRECT**: This is a requirement of TRM 3.3.3.10 for an inoperable LWM Rad Monitor. (Action 1b of TRM 3.3.3.10) These TRM actions are the generic actions for an inoperable rad monitor for both LWM and BM.

Technical Reference(s): TRM 3.3.3.10
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-LWM obj. 7 (As available)

Question Source: Bank # X Question #22
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 NRC SRO Exam

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

➔ (DRN 02-216)
3/4.3 INSTRUMENTATION (See note below)
← (DRN 02-216)
3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Requirement 3.11.1.1 are not exceeded during releases to the environment. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Requirement, immediately suspend release to the environment of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within 30 days if release to the environment are in progress or, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report, pursuant to Technical Specification 6.9.1.8, why this inoperability was not corrected within the time specified. Releases need not be terminated after 30 days provided the specified ACTIONS are continued.

SURVEILLANCE REQUIREMENT

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-8.

➔ (DRN 02-216)

NOTE: TRM Specifications 3.3.3.10 and 4.3.3.10 are part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of these TRM Specifications requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.
← (DRN 02-216)

→ (DRN 02-216)

TABLE 3.3-12 (See note below)

← (DRN 02-216)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	RELEASE INFORMATION	ACTION
1. BORON WASTE MANAGEMENT SYSTEM (BWMS):			
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release(PRM-IRE-0627)	1	Batch Release from Boric Acid Condensate Tanks	1
b. Waste (Process) Flow Rate Measurement Device (BM-IFT-0627)	1		2
2. LIQUID WASTE MANAGEMENT SYSTEM DISCHARGE (LWMS):			
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release (PRM-IRE-0647)	1	Batch Release from Liquid Waste Management Tanks	1
b. Waste (Process) Flow Rate Measurement Device (LWM-IFT-0647)	1		2

→ (DRN 02-216)

NOTE: TRM Table 3.3-12 is part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of this TRM Table requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← (DRN 02-216)

→ (DRN 02-216)

TABLE 3.3-12 (Continued, See note below)

← (DRN 02-216)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	RELEASE INFORMATION	ACTION
3. DRY COOLING TOWER SUMPS (DCTS):			
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release [PRM-IRE-6775 (DCTS#1) and PRM-IRE-6776(DCTS#2)]	1/sump	Release Path is NOT Aligned to LWMS	3
b. Waste (Process) Flow Rate Measurement Device (See Table Note. #1)	N/A	(see Note #2)	N/A
[redacted]			
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release (PRM-IRE-6778)	1	Release Path is NOT Aligned to LWMS	3
b. Waste (Process) Flow Rate Measurement Device (See Table Note. #1)	N/A	(see Note #2)	N/A

→ (DRN 02-216)

NOTE: TRM Table 3.3-12 is part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of this TRM Table requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← (DRN 02-216)

→ (DRN 02-216)

TABLE 3.3-12 (Continued, See note below)

← (DRN 02-216)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	RELEASE INFORMATION	ACTION
5. CIRCULATING WATER DISCHARGE (CWD) - BLOWDOWN AND BLOWDOWN HEAT EXCHANGER DISCHARGES AND AUXILIARY COMPONENT COOLING WATER PUMPS:		1. Detectable Activity in Secondary Plant 2. During Blowdown of Steam Generators	
a. Radioactivity Monitor Providing Alarm and initiate Automatic Closure of Blowdown Valve BD-303 (PRM-IRE-1900)	1	to CW System. 3. During Discharge of ACCW Basins	4
b. Waste (Process) Flow Rate Measurement Device (See Table Note. #1)	N/A	to CW System	N/A
6. STEAM GENERATOR BLOWDOWN (SGB) EFFLUENT LINE		During Blowdown Of S / Gs	
a. Continuous Composite Sampler	1	to CW System or Metal Waste Ponds (see Note #3)	4

→ (DRN 02-216)

NOTE: TRM Table 3.3-12 is part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of this TRM Table requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.

← (DRN 02-216)

➔ (DRN 02-216)
TABLE 3.3-12 (Continued, See note below)
← (DRN 02-216)

TABLE NOTATIONS

NOTE #1 Waste (process) Flow Measurement Devices are not installed on the release paths for the DCTS, TBIWS or CWD monitors. For these release paths, pump performance curves generated in place or some form of volumetric estimate or measurement device may be used for effluent flow rate estimates.

NOTE #2 DCTS and TBIWS monitor operation should be maximized during releases to the environment, even when detectable activity is not present in the CCW/ACCW or secondary systems, to provide capability for release termination in the event that Primary to Secondary or Primary to CCW leakage occurs.

NOTE #3 The Steam Generator Blowdown Composite Sampler is capable of sampling blowdown discharge to either the CW System or Waste Ponds. Blowdown to the Waste Ponds is not allowed unless radiation monitoring capable of release termination is added to the release path.

ACTION STATEMENTS

ACTION 1 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement effluent releases via this pathway may continue provided best efforts are made to repair the instrument and that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Requirement 4.11.1.1.1 and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup.

ACTION 2 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided best efforts are made to repair the instrument and that the flow rate is estimated at least once per FOUR hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

➔ (DRN 02-216)
NOTE: TRM Table 3.3-12 is part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of this TRM Table requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.
← (DRN 02-216)

ACTION STATEMENTS

ACTION 3 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided best efforts are made to repair the instrument and that grab samples are collected and are analyzed within 24 hours of collection time for radioactivity at a lower limit of detection of at least 5 E-07 microcurie/ml. Sample collection and analysis is NOT required if the release path for the DCTS/TBIWS is aligned to the LWMS. The sample collection frequency is:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcurie/gram DOSE EQUIVALENT I-131.

ACTION 4 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided best efforts are made to repair the instrument and that grab samples are collected and are analyzed within 24 hours of collection time for radioactivity at a lower limit of detection of at least 5 E-07 microcurie/ml. Sample collection and analysis is NOT required if no detectable activity exists in either the secondary plant or CCW/ACCW systems. Sampling of Steam Generator Blowdown is required during blowdown to the CW System or Waste Ponds. The sample collection frequency is:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcurie/gram DOSE EQUIVALENT I-131.

➔ (DRN 02-216)
NOTE: TRM Table 3.3-12 is part of the Offsite Dose Calculation Manual (ODCM), reference UNT-005-014. Revision of this TRM Table requires the approval of the General Manager Plant Operations (GMPO) in accordance with Technical Specification 6.14.
← (DRN 02-216)

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL SOURCE CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST	RELEASE INFORMATION
1. BORON WASTE MANAGEMENT SYSTEM DISCHARGE (BWMS):					
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release (PRM-IRE-0627)	Prior to Release (6)	Prior to Release (6)	18 Months (3)	Quarterly (1)	Batch Release from Boric Acid Condensate Tanks
b. Waste (Process) Flow Rate Measurement Device.(BM-IFT-0627)	Daily (4)	N/A	18 Months		
2. LIQUID WASTE MANAGEMENT SYSTEM DISCHARGE (LWMS):					
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release (PRM-IRE-0647)	Prior to Release (6)	Prior to Release (6)	18 Months (3)	Quarterly (1)	Batch Release from Liquid Waste Management Tanks
b. Waste (Process) Flow Rate Measurement Device (LWM-IFT-0647)	Daily (4)	N/A	18 Months		
3. DRY COOLING TOWER SUMPS (DCTS):					
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release (PRM-IRE-6775 and PRM-IRE-6776)	Daily	Monthly	18 Months (3)	Quarterly (2)	Release Path is NOT Aligned to LWMS
b. Waste (Process) Flow Rate Measurement	N/A	N/A	N/A	N/A	

TABLE 4.3-8 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL SOURCE CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST	RELEASE INFORMATION
4. INDUSTRIAL WASTE SUMPS TURBINE BUILDING (IWSTB):					
a. Radioactivity Monitor Providing Alarm and Automatic Termination of Release (PRM-IRE-6778)	Daily	Monthly	18 Months(3)	Quarterly (2)	Release Path is NOT Aligned to LWMS
b. Waste (Process) Flow Rate Measurement	N/A	N/A	N/A	N/A	
5. CIRCULATING WATER DISCHARGE (CWD) BLOWDOWN AND BLOWDOWN HEAT EXCHANGER DISCHARGE AND AUXILIARY COMPONENT COOLING WATER PUMPS: (TERMINATION OF BLOWDOWN DISCHARGE ONLY)					1. Steam Generators Blowdown to CW System
a. Radioactivity Monitor Providing Alarm and initiate Automatic Closure of Blowdown Discharge Valve BD-303	Daily	Monthly	18 Months (3)	Quarterly (2)	2. Discharge of ACCW Basins to CW System
b. Waste (Process) Flow Rate Measurement	N/A	N/A	N/A	N/A	
6. STEAM GENERATOR BLOWDOWN (SGB) EFFLUENT LINE:					Blowdown of S/Gs to CW System or Metal Waste Ponds
a. Continuous Composite Sampler	Daily (5)	N/A	18 Months	Quarterly	

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

1. The CHANNEL FUNCTIONAL TEST for BWM and LWM shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists.

   a. Instrument indicates measured levels above the alarm/trip setpoint.
   b. Circuit failure.
   c. Instrument indicates a downscale failure.

2. The CHANNEL FUNCTIONAL TEST for DCTS, TBIWS and CWD shall also demonstrate that automatic isolation of this pathway occurs if the instrument indicates measured levels above the alarm/trip setpoint and that control room alarm annunciation occurs if any of the following conditions exists:

   a. Instrument indicates measured levels above the alarm setpoint.
   b. Circuit failure.
   c. Instrument controls not set in operate mode.

3. The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology NIST or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system for over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration may be used in lieu of the reference standards associated with the initial calibration.

4. CHANNEL CHECK for BWM and LWM shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

5. CHANNEL CHECK for Steam Generator Blowdown Composite Sampler shall be made at least once per 24 hours on days on which releases are made to the Circulating Water System or Waterford 3 waste pond.

6. CHANNEL CHECK for BWM and LWM shall consist of observing a satisfactory channel source check which is also performed prior to the release.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.4.42	
	Importance Rating		3.8

K/A Statement

Emergency Procedures / Plan: Knowledge of emergency response facilities.

Proposed Question: SRO 25 Rev: 0

Per EP-002-100, Technical Support Center Activation, Operation and Deactivation, the lowest level of event classification that requires activation of the Technical Support Center (TSC) is an ____(1)____ . It is the responsibility of the ____(2)____ to direct the activities of the TSC including overall management of the onsite Emergency Response Organization.

	(1)	(2)
A.	Unusual Event	TSC Manager
B.	Unusual Event	Emergency Plant Manager
C.	Alert	TSC Manager
D.	Alert	Emergency Plant Manager

Proposed Answer: D

Explanation: (Optional)

A. Incorrect: The TSC must be activated at an Alert Condition (EP-002-100 step 4.2). The Emergency Plant Manager is in charge of the TSC and overall management of the onsite ERO. (EP-002-100 step 3.1.2). The TSC Manager is located in the TSC but reports to the EPM (EP-002-100 step 3.2).
B. Incorrect: The TSC must be activated at an Alert Condition (EP-002-100 step 4.2). The Emergency Plant Manager is in charge of the TSC and overall management of the onsite ERO. (EP-002-100 step 3.1.2). The TSC Manager is located in the TSC but reports to the EPM (EP-002-100 step 3.2).
C. Incorrect: The TSC must be activated at an Alert Condition (EP-002-100 step 4.2). The Emergency Plant Manager is in charge of the TSC and overall management of the onsite ERO. (EP-002-100 step 3.1.2). The TSC Manager is located in the TSC but reports to the EPM (EP-002-100 step 3.2).
D. **CORRECT:** The TSC must be activated at an Alert Condition (EP-002-100 step 4.2). The Emergency Plant Manager is in charge of the TSC and overall management of the onsite ERO. (EP-002-100 step 3.1.2). The TSC Manager is located in the TSC but reports to the EPM (EP-002-100 step 3.2).

Technical Reference(s): EP-002-100 revision 40 pages 3 and 4 and 6
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EP02 objective 5 (As available)

Question Source:
Bank #
Modified Bank #
New X

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41
55.43 1

Comments:

2.27 OP-004-016, Seismic Monitoring

2.28 Drawing 5817-12134

2.29 HP-CAL8C-2001-001, PASS System Elimination (Dose Rate Calculation)

2.30 Emergency Planning Desk Guides

2.31 OP-901-401, High Airborne Activity in Control Room

2.32 OP-902-002, Loss Of Coolant Accident Recovery

2.33 OP-902-004, Excess Steam Demand Recovery

2.34 OP-902-008, Safety Function Recovery

2.35 EC530 – Ultimate Heat Sink Water Replenishment for Tornado Event

3.0 RESPONSIBILITIES

3.1 Emergency Plant Manager

3.1.1 Overall management of the onsite Emergency Response Organization.

3.1.2 Directing the activities of the Technical Support Center, including activation and deactivation, until the accident is terminated

3.1.3 Coordinating with the Shift Manager (SM) and the Emergency Director (ED) those response actions necessary for control of the accident and protection of emergency personnel and the public.

3.1.4 Request firefighting and emergency medical assistance from offsite agencies.

3.1.5 Keeping the Emergency Director informed on the status of the plant and conditions within the Protected Area Fence.

3.1.6 Initiate protective actions at the site.

3.1.7 Authorize radiation exposure in excess of 10CFR20 limits and administration of KI for onsite personnel.

3.1.8 Invoke provisions of 10 CFR 50.54 (x) if needed.

3.1.9 Implement severe accident management procedures as necessary

3.2 TSC Manager

3.2.1 Reports to the Emergency Plant Manager.

3.2.2 Ensures the coordination of the TSC and Operational Support Center (OSC) activities.

3.2.3 Coordinate the efforts of the TSC Engineers.

3.2.4 Coordinate TSC staffing and activation.

3.2.5 Accountability of TSC personnel.

3.3 Radiological Coordinator (RC)

3.3.1 Reports to the TSC Manager.

3.3.2 Coordinate onsite radiological monitoring activities.

3.3.3 Maintain communications with the Radiological Assessment Coordinator in the EOF.

3.3.4 Coordinate in-plant radiation protection activites by directing the activities through the OSC Rad/Chem Coordinator (RCC).

3.3.5 Overall management of site Radiation Protection activities including Emergency Personnel Exposure Limits and the administration of Potassium Iodide (KI) for onsite personnel.

3.3.6 Contact for the NRC on the Health Physics Network (HPN) Line.

3.3.7 Coordinates with Security Coordinator to determine evacuation routes for site personnel and personnel located in BREs.

3.4 Operations Coordinator

3.4.1 Operations Coordinator reports to the TSC Manager.

3.4.2 Keep the Emergency Plant Manager and TSC Manager informed of plant status.

3.4.3 Operation of the SDS console to obtain necessary plant conditions and status.

3.4.4 Coordinate with the EPM, TSC Manager and OSC Manager to provide support to the Control Room to mitigate the effects of the event and return the plant to a safe condition.

3.4.5 Maintains communications with the Operations Communicator for updates of activities of the Control Room.

3.5 Chemistry Engineer

3.5.1 Chemistry Engineer reports to the Operations Coordinator.

3.5.2 Coordinates chemistry related emergency response activities.

3.6 Engineering Coordinator

3.6.1 Engineering Coordinator reports to the TSC Manager.

3.6.2 Provides the overall direction of the engineers conducting plant assessment and technical evaluation.

3.6.3 Assists with individual reviews and approval of 10CFR50.54(x) considerations

3.7 Maintenance Coordinator

3.7.1 Assists the CR, EPM, TSC Manager and OSC Manager in planning and preparing for any OSC maintenance activities needed to return the plant to a safe condition.

3.7.2 Monitors OSC manpower status.

3.7.3 Maintains status of OSC Work Assignments.

3.7.4 Tracks activities of Fire Brigade and First Aid teams.

4.0 INITIATING CONDITIONS

[redacted]

4.2 The TSC shall be activated at an Alert, Site Area Emergency or General Emergency declaration.

4.3 The TSC shall become operational as soon as possible after declaration of any of these emergency classifications.

4.3.1 When facility minimum staffing can be accomplished with onsite personnel, then the goal is to become operational within 45 minutes.

4.3.2 When facility minimum staffing must be accomplished using offsite personnel, then the TSC shall become operational within 90 minutes.