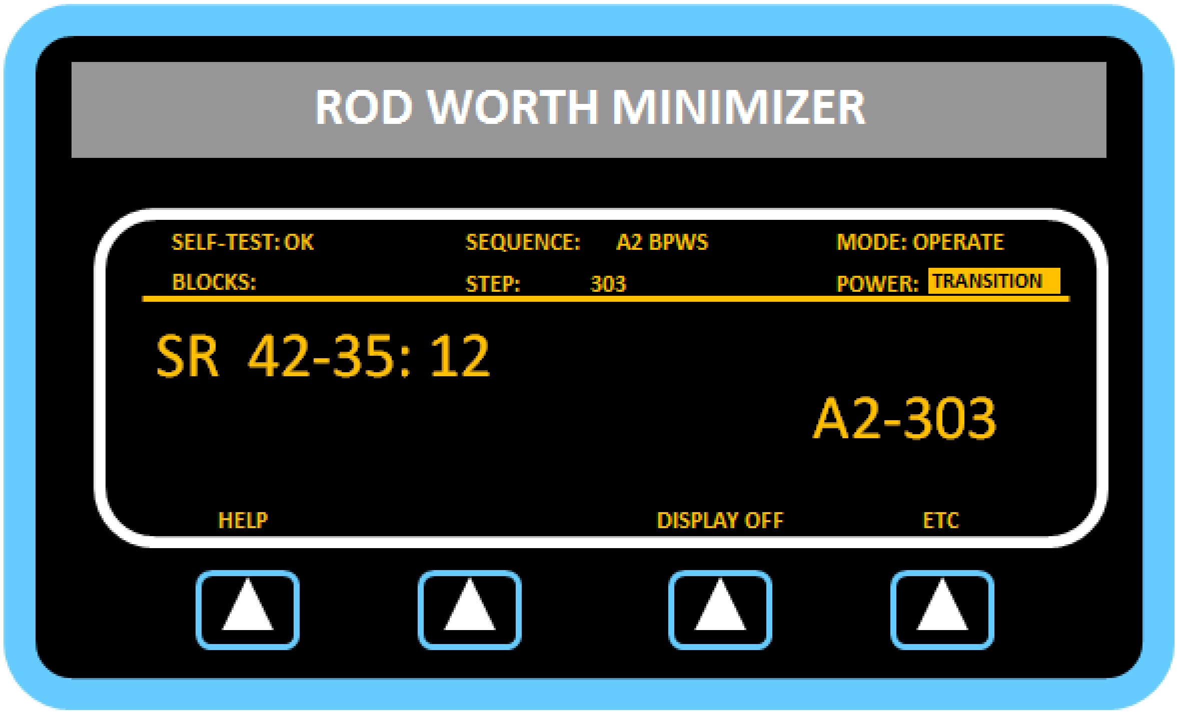
76. S201006 1



Which one of the following completes the statements below concerning the Rod Worth Minimizer (RWM)?

The RWM channel functional test was NOT required to be performed ___(1)___ any control rod was withdrawn at <8.75% RTP in MODE 2 IAW Surveillance Requirement 3.3.2.1.2.

Currently total steam flow is ___(2)___%.

A. (1) until
(2) 18

B. (1) until
(2) 20

C. (1) until 1 hour after
(2) 18

D. (1) until 1 hour after
(2) 20

Answer: D
K/A:
201006 ROD WORTH MINIMIZER SYSTEM (RWM)
G2.04.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.2

Pedigree: New

Objective:
LOI-CLS-LP-07.1, Obj. 3 - Describe the operation of the RWM above and below the Low Power Setpoint (LPSP) and the Low Power Alarm Point (LPAP), including the setpoints and where the input signal originates.
LOI-CLS-LP-07.1, Ob. 10- Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine the required action(s) to be taken in accordance with Technical Specifications, the TRM, or ODCM associated with the RWM System. (SRO/STA only) (LOCT)

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
The RWM uses total steam flow to determine reactor power. 19.1% is the LPSP and 27.8% is the LPAP. The area between these two setpoint is the transistion zone where the RWM will alarm but not enforce the blocks. A note in the surveillance requirements allows 1 hour after any control rod is withdrawn in MODE 2 to perform the SR.

Distractor Analysis:

Choice A: Plausible because when a control is withdrawn would be correct if it was not for the note in the SR. Since the Transition zone is indicated by the picture steam flow is greater than 19.1%.

Choice B: Plausible because when a control is withdrawn would be correct if it was not for the note in the SR. Part two is correct.

Choice C: Plausible because part one is correct and since the Transition zone is indicated by the picture steam flow is greater than 19.1%

Choice D: Correct Answer, see explanation

SRO Basis:
Facility operating limitations in the technical specifications and their bases. [10 CFR 55.43(b)(2)]

Control Rod Block Instrumentation
3.3.2.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	-------------------NOTE------------------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 8.75% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days

3.4.3 Reactor Power Level Inputs (LPSP, LPAP)

The RWM enforces control rod movement from all rods full-in to the Low Power Setpoint (LPSP). The RWM will alarm but not enforce control rod movement between the LPSP and the Low Power Alarm Point (LPAP). The region between the LPSP and LPAP is sometimes referred to as the transition region.

During a power descent through the transition region, the RWM identifies to the operator that they must establish an RWM compatible sequence soon. There are no control rod movement enforcements within the transition region.

The LPSP and LPAP signals are input to RWM from the feedwater control circuitry. Total steam flow is used to monitor reactor power level as seen by the RWM. The total steam flow is compared by the feedwater control circuitry to the LPSP and LPAP setpoints. The LPSP contact will open at approximately 19.1% steam flow (decreasing). The LPAP contact will open at approximately 27.8% steam flow (decreasing). Since RWM Reactor Power is derived from steam flow, an abrupt decrease in steam flow could cause the LPSP to be achieved, and thus enforcement of any RWM block. Refer to SD-32.2 for additional information on the Feedwater level Control System.

77. S203000 1

Following a large line break DBA LOCA, plant conditions are:

Reactor water level	-50 inches (N036/N037)
Reactor pressure	5 psig
Core Spray	One loop available, injecting at 4800 gpm
RHR	One loop available, injecting at 17,000 gpm
Suppression pool temp.	148°F

The CRS has reached step RC/L-44 in RVCP:

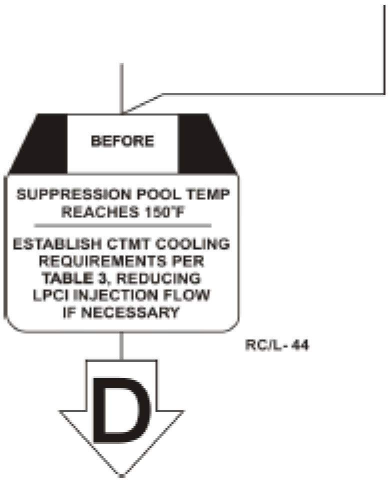


TABLE 3

MINIMUM CONTAINMENT COOLING REQUIREMENTS

MINIMUM PUMP REQUIREMENTS	FLOW (GPM)
1 - RHR PUMP PER LOOP WITH HX BYPASS VALVE CLOSED	7700/LOOP
1 - RHR SW PUMP PER LOOP	4000/LOOP
2 - RHR PUMPS WITH HX BYPASS VALVE CLOSED	11,500
2 - RHR SW PUMPS	8000

Which one of the following identifies the basis for reduction in RPV injection when reactor water level is below Minimum Steam Cooling Reactor Water Level?

A. Prevent exceeding the Heat Capacity Temperature Limit.

B. Maintain long term operation of the Core Spray and RHR Pumps.

C. Minimize off-site releases per Alternative Source Term calculations.

D. Prevent exceeding design temperature limits for Primary Containment.

Answer: B

K/A:

203000 RHR/LPCI: INJECTION MODE

A2 Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

16 Loss of coolant accident

RO/SRO Rating: 4.4/4.5

Pedigree: 2004 Audit Exam

Objective:

LOI-CLS-LP-017, Obj. 28 - Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event: d. Loss of coolant accident (LOCT)

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
The calculation for the NPSH to the Core Spray and RHR pumps specify establishing cooling at ten minutes into the design basis LOCA. The calculation also assumes that the temperature of the suppression pool will be at approximately 169°F at ten minutes. If containment cooling is not established, then it is possible that the Core Spray or RHR pumps will be lost due to inadequate NPSH.

Distractor Analysis:

Choice A: Plausible because the heat capacity temperature limit would be of concern if the reactor was pressurized to make sure the suppression pool could handle the heat loading.

Choice B: Correct Answer, see explanation

Choice C: Plausible because RVCP has actions within it for alternate source term analysis dealing with the suppression pool.

Choice D: Plausible because while these values are high but do not exceed the design limit of 220°F in the suppression pool.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]
This requires assessing plant conditions and based on a decision point the usage of a table and the knowledge of the EOP basis for establishing long term cooling for containment.

This step provides guidance on establishing cooling to the suppression pool during a design basis LOCA. The calculation for the NPSH to the Core Spray and RHR pumps specify establishing cooling at ten minutes into the design basis LOCA. The calculation also assumes that the temperature of the suppression pool will be at approximately 169°F at ten minutes. If containment cooling is not established at a suppression pool temperature of 169°F, then it is possible that the Core Spray or RHR pumps will be lost due to inadequate NPSH. The EPGs specify injecting irrespective of NPSH and vortex limits if reactor vessel water level is below Minimum Steam Cooling Reactor Water Level. This step provides guidance to reduce injection into the reactor vessel to establish the desired cooling for the containment. A value of 150°F has been selected for use in the step. This provides a margin of at least 19°F, to the limit used in the calculation for NPSH. These actions are incorporated to provide assurance that the unit can remain in the EOPs and not be required to go to primary containment flooding prematurely. The selection of a suppression pool temperature limit precludes establishing a specific time limit in the procedure.

78. S215003 1

Unit One is performing a reactor startup, prior to the point of adding heat.

IRM C is bypassed due to erratic operation.
IRM A fails downscale.

Which one of the following completes the statements below?

Addressing ONLY Technical Specification 3.3.1.1, Reactor Protection System (RPS) Instrumentation, requires placing the channel in trip in ___(1)___ hours.

IAW AD-OP-ALL-0101, Event Response and Notifications, the plant manager will be directly notified of this event by the ___(2)___.

(Reference provided)

A. (1) 6
   (2) Shift Manager

B. (1) 6
   (2) Site Duty Manager

C. (1) 12
   (2) Shift Manager

D. (1) 12
   (2) Site Duty Manager

Answer: D

K/A:

215003 INTERMEDIATE RANGE MONITOR (IRM) SYSTEM

G2.04.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

RO/SRO Rating: 2.7/4.1

Pedigree: new

Objective:

LOI-CLS-LP-009-A, Obj. 13 - Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine the required action(s) to be taken in accordance with Technical Specifications, TRM and COLR associated with the Source Range and Intermediate Range Monitoring Systems. (SRO/STA only)

Reference: TS 3.3.1.1

Cog Level: High

Explanation:
TS table 3.3.1.1-1requires 3 operable IRM channels per trip system. With IRM A & C inop, 3 channels are not operable for one trip system. IRM A and C are in the same trip system. Condition A must be entered. The question does NOT address TRM.

IAW with the procedure the SM notifies the Site Duty Manager who in turn makes all other notifications.

Distractor Analysis:

Choice A: Plausible because TS Condition B would be applicable if one or more required trip systems were inoperable and would require placing in trip condition within 6 hours and the SM does notify the Site Duty Manager but not the PM.

Choice B: Plausible because TS Condition B would be applicable if one or more required trip systems were inoperable and would require placing in trip condition within 6 hours and the Site Duty Manager does notify the PM.

Choice C: Plausible because TS condition A is required to be entered, placing the channel in trip in 12 hours and the SM does notify the Site Duty Manager but not the PM.

Choice D: Correct Answer, see explanation

SRO Basis:
Facility operating limitations in the technical specifications and their bases. [10 CFR 55.43(b)(2)]

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	OR	
	A.2 -----NOTE----- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------- Place associated trip system in trip.	12 hours

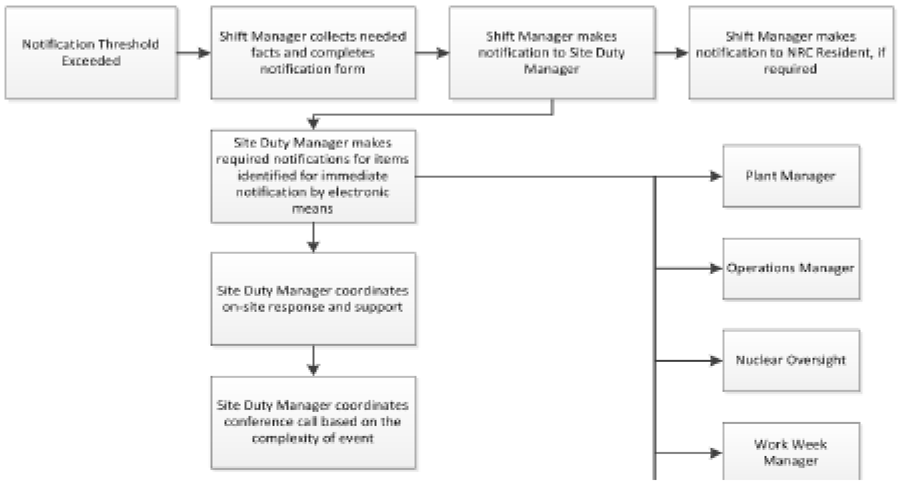
CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ---------NOTE------------ Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ------------------------------ One or more Functions with one or more required channels inoperable in both trip systems.	B.1 OR B.2	Place channel in one trip system in trip. Place one trip system in trip.	6 hours 6 hours

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA

ATTACHMENT 1
Page 1 of 1

SITE EVENT NOTIFICATION FLOWCHART



79. S215005 1

The following Unit One information is obtained in preparation for filling out the Emergency Notification Form for a low power ATWS:

1204 ONLY one rod out
1205 All rods in

SLC was NOT injected

<u>ERFIS Met Data:</u>

Upper Wind Direction	15.00	Deg
Lower Wind Direction	13.00	Deg

The Control Room Site Emergency Coordinator is evaluating the following lines:

9. METEOROLOGICAL DATA: Wind Direction* from ______ degrees Wind Speed* ______mph

12. UNIT STATUS: (Unaffected Unit(s) Status Not Required for Initial Notifications)
U1 __0__% Power Shutdown at Time __________ Date ___/___/___
U2 _____% Power Shutdown at Time __________ Date ___/___/___

Which one of the following completes the statements below IAW 0PEP-02.6.21, Emergency Communicator?

On line 9, ___(1)___ degrees should be entered for "Wind Direction from".

On Line 12, the earliest time that can be entered for Unit One "Shutdown at Time" is ___(2)___ hours.

A. (1) 13
(2) 1204

B. (1) 13
(2) 1205

C. (1) 15
(2) 1204

D. (1) 15
(2) 1205

Answer: A

K/A:
215003 AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM
G2.04.30 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required. (CFR: 41.10 / 43.5 / 45.11)

RO/SRO Rating: 2.4/4.4

Pedigree: New

Objective:
Task - SRO Only - Complete And Approve Emergency Notification Forms For Initial And Follow-up Notifications To State/County Agencies In Accordance With 0PEP-02.6.21.

Reference: None

Cog Level: High

Explanation:
Recent OE has had operators not using the one rod out difinition that is allowed by the procedure for determining that the reactor is shutdown. One rod out meets the definition for Shutdown Margin and All Rods In also meets this definition. The ENF form uses the lower wind speed from the ERFIS Met Data screen in the control room.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the lower wind directions is used but all rods in is not the earliest time that the reactor can remain shutdown under all conditions without boron.

Choice C: Plausible because 15 degrees is the upper wind direction not the lower wind direction and 1204 hours is correct.

Choice D: Plausible because 15 degrees is the upper wind direction not the lower wind direction and all rods in is not the earliest time that the reactor can remain shutdown under all conditions without boron.

SRO Basis:
This task is an SRO Only task.
344020B502 Complete And Approve Emergency Notification Forms For Initial And Follow-up Notifications To State/County Agencies In Accordance With 0PEP-02.6.21.

NOTE:	Information for Line 9 may be obtained from the STA (if in the Control Room) or the Radiological Control Manager (if in the EOF).
NOTE:	Information may not be available for Initial Notifications.

_______9 **"METEOROLOGICAL DATA"**

If using WebEOC and importing Met Data you must first complete Line 6, 7, and 11, then select "Import Plant/MET Data." Imported Met Data is current data.

Enter lower wind direction and wind speed if completing hard copy ENF.

LINE NO.	INSTRUCTIONS
______11	**"AFFECTED UNIT(S)"** - mark appropriate block. If event affects both Units, indicate Unit 1 and 2. If using WebEOC and event affects Unit 1 and Unit 2 mark "Both"

> **NOTE:** For Line 12 once the EOF has responsibility, the TAM and EOF SRO are a source for the information
>
> **NOTE:** Unaffected Unit status is not required for Initial Notifications. Must be provided in a "Follow-Up" notification as soon as information is available.
>
> **NOTE:** For off-site notification purposes, the reactor is considered SHUTDOWN (under all conditions) by any ONE of the following:
> 1. All control rods fully inserted.
> 2. All control rods fully inserted with the exception of 1 (one) control rod at any withdrawn position.
> 3. All control rods fully inserted with the exception of up to 10 (ten) control rods withdrawn to position 02.
> 4. As determined by Reactor Engineering.

______12 **"UNIT STATUS"** - information for Unit(s) specified on Line 11. If both Units, then include status for both

- Complete blocks **"A"** and **"B"** for current reactor power level.

  If using WebEOC imported "Plant/Met Data" Unit Status will be entered for both units

  If using WebEOC and imported plant data, verify that the "% Power" does not exceed 100%. If "% Power" exceeds 100% change to indicate 100%.

- If the reactor is shutdown, place "0%" power and indicate the time/date of shutdown.

> **NOTE:** If using WebEOC, and the conditions of a reactor shutdown are not met under all conditions, even if the reactor power shows 0.0, then a 1 should be entered (a whole number is required) for % power, then the program will not require a shutdown time and date.

If using WebEOC the actual shutdown time/date must be determined and manually entered. If using "Get Date" for shutdown date, verify that the date is correct, correct if necessary. Shutdown time must be entered manually in 24 hour format (HHMM).

EOP User's Guide Definitions:

SHUTDOWN

As applied to the reactor, subcritical with reactor power below the heating range.

80. S261000 1

Unit One is operating at rated power.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days

On 10/13/14 at 1200 hours it is discovered that this SR was last performed on 09/01/14 at 0400 hours.

Which one of the following is correct IAW Technical Specifications?

A. The time between surveillances is acceptable IAW SR 3.0.2.

B. The time between surveillances is acceptable IAW SR 3.6.4.3.1.

C. The time between surveillances is unacceptable. Entry into the applicable Conditions and Required Actions for the missed surveillance is immediately required.

D. The time between surveillances is unacceptable. Entry into the applicable Conditions and Required Actions may be delayed for up to 31 days to permit performance of the surveillance.

Answer: D

K/A:

261000 STANDBY GAS TREATMENT SYSTEM

G2.02.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)

RO/SRO Rating: 3.6/4.5

Pedigree: New

Objective:

LOI-CLS-LP-200-B, Ob 9 - Given plant conditions, apply the rules of Section 3.0 to determine appropriate actions in accordance with Technical Specifications. (SRO/STA Only)(LOCT)

Reference: None

Cog Level: hi

Explanation:

SR has not been performed for 42 days.

38.75 max per 25% criteria of SR 3.0.2.

3.0.3 allows delay of up to 31 days from discovery of missed surv.

Distractor Analysis:

Choice A: Plausible because SR 3.0.2 allows an additional 25%.

Choice B: Plausible because if the student thinks that the surv. is monthly.

Choice C: Plausible because the surv. time is unacceptable and if SR 3.0.3 is not applied this is correct.

Choice D: Correct Answer, see explanation

SRO Basis:
Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]
Application of generic Limiting Condition for Operation (LCO) requirements (LCO 3.0.1 thru 3.0.7; SR 4.0.1 thru 4.0.4).

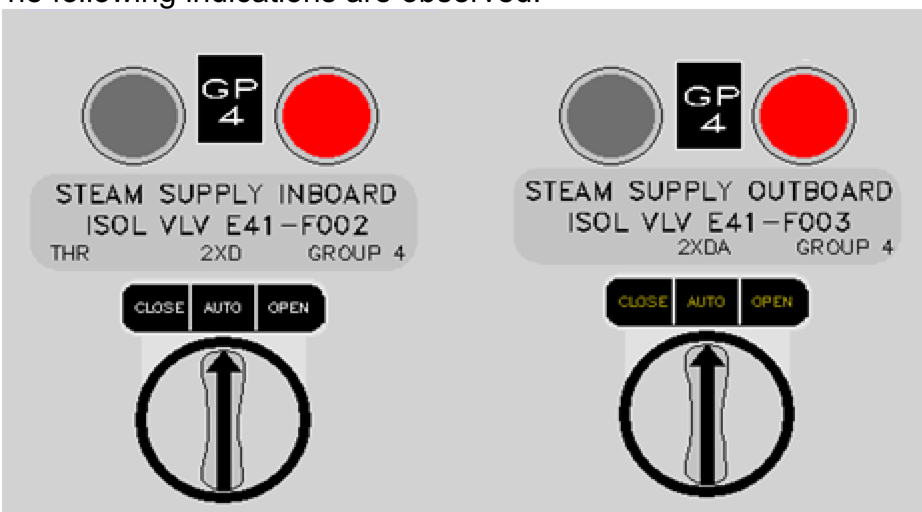
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

81. S290001 1

Unit Two was operating at rated power when the following alarms/conditions are reported:

UA-03 (2-7) *Area Rad Rx Bldg High*
UA-03 (4-5) *Process Rx Bldg Vent Rad Hi-Hi*
UA-05 (6-10) *Rx Bldg Isolated*
UA-05 (4-6) *SBGT Sys A Failure*
A-02 (5-7) *Stm Leak Det Ambient Temp High*
UA-05 (6-7) *Rx Bldg Static Press Dif-Low*
A-01 (3-5) *HPCI Isol Trip sig A Initiated*
A-01 (4-5) *HPCI Isol Trip sig B Initiated*
Steam is visible exiting the Reactor Building blowout panels

The following indications are observed:



Which one of the following completes the statements below?

The release through the Reactor Building blowout panels is considered ___(1)___ release.

The highest Emergency Action Level classification for the given conditions is ___(2)___.

(Reference provided)

A. (1) a ground
   (2) an Alert

B. (1) a ground
   (2) a Site Area Emergency

C. (1) an elevated
   (2) an Alert

D. (1) an elevated
   (2) a Site Area Emergency

Answer: B

K/A:
290001 SECONDARY CONTAINMENT
G2.04.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

RO/SRO Rating: 4.1/4.3

Pedigree: New

Objective:
LOI-CLS-LP-301-B, Obj. 9 - Given a hypothetical abnormal event and plant operating mode, use 0PEP-02.1 to properly classify or re-classify the event.

Reference: 0PEP-02.1

Cog Level: High

Explanation:
Even though the release is out the 117 foot elevation this is considered a ground release. The ambient Temp High alarm indicates that a Group 4 isolation should have occurred
With the inability to isolate the leak the EAL call is a loss of RCS and a loss of Containment which makes a SAE.

Distractor Analysis:

Choice A: Plausible because a ground release is correct and if only one barrier is assessed then this is correct.

Choice B: Correct Answer, see explanation

Choice C: Plausible because even though the release is out the 117 foot elevation this is considered a ground release and if only one barrier is assessed then this is correct.

Choice D: Plausible because even though the release is out the 117 foot elevation this is considered a ground release and a SAE is correct.

SRO Basis:
Fuel handling facilities and procedures. [10 CFR 55.43(b)(7)]

5. Select "Not an isolated stack" when the release is point is through the reactor building, turbine building, or radwaste building. Effective height is entered as 0 meters.

   - Enter **0 meters** for all "Not an isolated stack" (ground) releases

Reactor Coolant System Barrier	
Loss	**Potential Loss**
1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined	None
2. PC pressure > 1.7 psig due to RCS leakage	None
3. Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system): - Main steam line - HPCI steam line - RCIC steam line - RWCU - Feedwater 4. Emergency Depressurization is required	1. RCS leakage > 50 gpm inside the drywell 2. Unisolable primary system discharge outside primary containment as indicated by Secondary Containment area radiation or temperature above any Maximum Normal Operating Limit (0EOP-03-SCCP Tables 3, 1)

Containment Barrier	
Loss	**Potential Loss**
None	1. Primary Containment Flooding is required
1. PC pressure rise followed by a rapid unexplained drop in PC pressure 2. PC pressure response **not** consistent with LOCA conditions	2. PC pressure > 62 psig and rising 3. Deflagration concentrations exist inside PC (H2 ≥ 6% **AND** O2 ≥ 5%) 4. Suppression pool water temperature and RPV pressure cannot be maintained below the HCTL
3. Failure of **any** valve in **any** one line to close **AND** Direct release pathway to the environment outside PC exists after PC isolation signal (manual or automatic) 4. Intentional PC venting per EOPs 5. Unisolable primary system discharge outside primary containment as indicated by Secondary Containment area radiation or temperature above any Maximum Safe Operating Limit (0EOP-03-SCCP Tables 3,1)	None

82. S290003 1

Which one of the following identifies:

(1) the conditions which will cause the control room ventilation system to automatically align in the Fire Protection mode and

(2) the required Technical Specifications (TS) / Technical Requirements Manual (TRM) actions IAW 0OP-37, Control Building Ventilation System Operating Procedure, if the system was initiated for 20 minutes with minimum local smoke and with charcoal exposure doubtful?

A. (1) Smoke detected in the Unit One Electronic Equipment room **or** the manual pull station tripped in the Unit Two Electronic Equipment Room.
   (2) Initiate an LCO on the affected train IAW TS 3.7.3, CREV System, and TRM 3.18, CREV System-Smoke Protection Mode, ONLY.

B. (1) Smoke detected in the Unit One Electronic Equipment room **or** the manual pull station tripped in the Unit Two Electronic Equipment Room.
   (2) Initiate an LCO on the affected train IAW TS 3.7.3, CREV System, and TRM 3.18, CREV System-Smoke Protection Mode, and take the action specified in TS 5.5.7, Ventilation Filter Testing Program (VFTP).

C. (1) Smoke detected in the Unit One Electronic Equipment room **and** the manual pull station tripped in the Unit Two Electronic Equipment Room.
   (2) Initiate an LCO on the affected train IAW TS 3.7.3, CREV System, and TRM 3.18, CREV System-Smoke Protection Mode, ONLY.

D. (1) Smoke detected in the Unit One Electronic Equipment room **and** the manual pull station tripped in the Unit Two Electronic Equipment Room.
   (2) Initiate an LCO on the affected train IAW TS 3.7.3, CREV System, and TRM 3.18, CREV System-Smoke Protection Mode, and take the action specified in TS 5.5.7, Ventilation Filter Testing Program (VFTP).

Answer: C

K/A:

290003 CONTROL ROOM HVAC

A2 Ability to (a) predict the impacts of the following on the CONTROL ROOM HVAC ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

04 Initiation/failure of fire protection system

RO/SRO Rating: 3.1/3.3

Pedigree: New

Objective:

LOI-CLS-LP-037, Obj. 10 - Given plant conditions associated with the CB HVAC and EAF System determine the required action(s) to be taken in accordance with Technical Specifications, TRM, and COLR (LOCT) (SRO/STA Only)

Reference: None

Cog Level: Hi

Explanation:
The initiation is from an initiation in the U1 and U2 rooms, it can be from the detector or the manual actuation lever. A table within the operating procedure gives guidance for the LCO that needs to be entered. since the fire was in the AEER and not the washroom then the actions of TS 5.5.7 do not need to be performed.

Distractor Analysis:

Choice A: Plausible because this is halve of the logic not the full initiation logic and the TS actions are correct.

Choice B: Plausible because this is halve of the logic not the full initiation logic and the TS actions would be correct for a fire in the washroom.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the initiation logic is correct and the TS actions would be correct for a fire in the washroom.

SRO Basis:
Facility operating limitations in the technical specifications and their bases. [10 CFR 55.43(b)(2)]

The Control Room Ventilation Subsystem will enter into a mode of FILTERED recirculation and outside makeup on the following signals:

1. High Radiation detected at the Control Building intake plenum

2. High Radiation detected in the Control Room

3. LOCA detected by one of the following:

    a. Reactor Water Low Level 2

    b. Drywell Pressure - High

4. A combination of:

    a. smoke detected in Zone C4 or manual pull station activated in Zone C4

    AND

    b. smoke detected in Zone C5 or manual pull station activated in Zone C5.

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From the operating procedure, OP-37.

TABLE 1

CAUSE	CHARCOAL EXPOSURE IMPACT	ACTION
Fire with minimum local smoke	Doubtful	1. If fire detector was reset within 15 minutes of ventilation system initiation, no action required. 2. If fire detector was not reset within 15 minutes, initiate LCO on affected train in accordance with Tech. Spec. 3.7.3, TRM 3.18 and contact Engineering for evaluation. 3. If fire exists in men's bathroom, initiate LCO on affected train in accordance with Tech. Spec. 3.7.3, TRM 3.18 and take action specified in Tech. Spec. 5.5.7.a, b, and c.2.
Fire with general dense smoke	Probable	1. Declare LCO on affected train in accordance with Tech. Spec. 3.7.3, TRM 3.18 and take action specified in Tech. Spec. 5.5.7a, b, and c.2.

83. S295001 1

Unit Two was operating at rated power when a trip of 2A RFP occurred followed immediately by a trip of the 2B Reactor Recirc pump.
The following plant conditions exist:

Reactor power	52%
Total core flow (WTCF)	36.96 Mlbm/hr
OPRM system	Inoperable

Which one of the following completes the statements below?

The current operating point on the appropriate Power to Flow Map is ___(1)___.

Verifying the current operating point on the Power to Flow Map is directed by ___(2)___ Supplementary actions.

(Reference provided)

A. (1) 5% Buffer Region
(2) 2AOP-04.0, Low Core Flow, ONLY

B. (1) 5% Buffer Region
(2) 2AOP-04.0, Low Core Flow AND 0AOP-23.0, Condensate/Feedwater System Failure,

C. (1) Region B - Immediate Exit
(2) 2AOP-04.0, Low Core Flow, ONLY

D. (1) Region B - Immediate Exit
(2) 2AOP-04.0, Low Core Flow AND 0AOP-23.0, Condensate/Feedwater System Failure,

Answer: B

K/A:
295001 PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION
G2.01.20 Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 4.6/4.6

Pedigree: Modified from NRC Exam 10-2 (Instead of asking which power to flow map this question asks the region on the power to flow map.)

Objective:
CLS-LP-302-C, Obj. - 4. Given plant conditions and AOP-04.0, determine the required supplementary actions.

Reference: B2C21 Core Operating Limits Report Figures 2 and 4.

Cog Level: High

Explanation:
AOP-04 & AOP-23 both provide guidance determine the current operating point on the Power to Flow Map. Without I&C doing adjustments to APRMs the current power to flow map is still the two loop map.

Distractor Analysis:

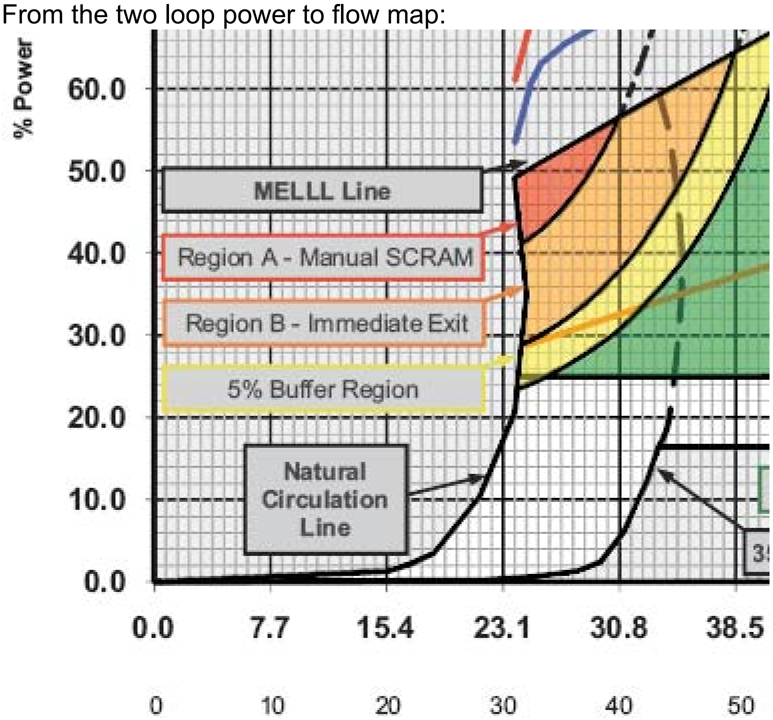
Choice A: Plausible because this is the correct region and AOP-4.0 does have an action but AOP-23 also has an action to determine the current operating point on the power to flow map

Choice B: Correct Answer, see explanation

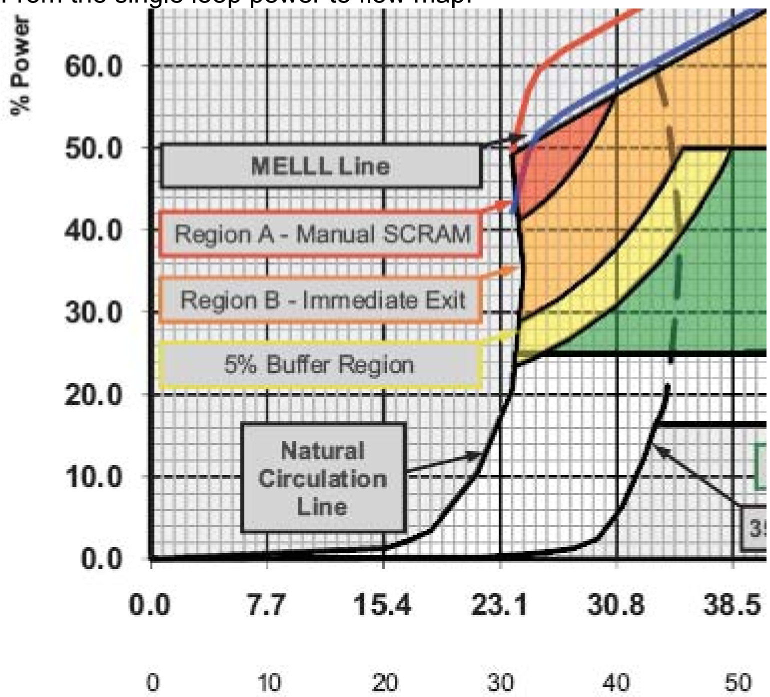
Choice C: Plausible because this is the region on the the single loop power to flow map, not the two loop and AOP-4.0 does have an action but AOP-23 also has an action to determine the current operating point on the power to flow map.

Choice D: Plausible because this is the region on the the single loop power to flow map, not the two loop.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]
From the two loop power to flow map:



From the single loop power to flow map:



84. S295005 1

Which one of the following completes the statements below IAW Technical Specification 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation?

Three channels of feedwater and main turbine high water level trip instrumentation are required to be operable ___(1)___.

The bases for the high water level trip instrumentation indirectly initiating a reactor scram from the main turbine trip on high reactor level is to ___(2)___.

A. (1) in MODE 1
   (2) mitigate the reduction in MCPR

B. (1) in MODE 1
   (2) protect the main turbine from damage

C. (1) when thermal power is >23% RTP
   (2) mitigate the reduction in MCPR

D. (1) when thermal power is >23% RTP
   (2) protect the main turbine from damage

Answer: C

K/A:
295005 MAIN TURBINE GENERATOR TRIP
G2.02.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

RO/SRO Rating: 4.0/4.7

Pedigree: New

Objective:
LOI-CLS-LP-026, Obj. 29 - Given plant conditions, determine whether given plant conditions meet minimum Technical Specifications requirements associated with the Main Turbine, Gland Seal, and Moisture Separator Reheater system.

Reference: None

Cog Level: Memory

Explanation:
This TS is applicable when thermal power is greater than 23% RTP. IAW the bases the reason for the scram is to mitigate MCPR.

Distractor Analysis:

Choice A: Plausible because MODE 1 is power operation.

Choice B: Plausible because MODE 1 is power operation and this is the reason for the turbine trip not the scram from the turbine trip.

Choice C: Correct Answer, see explanation

Choice D: Plausible because this is the reason for the turbine trip not the scram from the turbine trip.

SRO Basis:

Facility operating limitations in the technical specifications and their bases. [10 CFR 55.43(b)(2)]

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1).

APPLICABLE SAFETY ANALYSES (continued)

The high water level trip indirectly initiates a reactor scram from the main turbine trip (above 26% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR and increase in LHGR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

85. S295015 1

The following plant conditions exist on Unit Two:

An ATWS with a spurious Group I Isolation has occurred
HPCI is injecting to the RPV to maintain RPV level
A-01 (1-5) *Suppression Chamber Lvl Hi-Hi* is in alarm

Which one of the following identifies:

(1) the reason that HPCI is re-aligned from its current suction source and
(2) the procedure that contains the steps to perform the actions to transfer the HPCI suction valves?

A. (1) To prevent pump bearing damage
(2) 2OP-19, High Pressure Coolant Injection System Operating Procedure

B. (1) To prevent pump bearing damage
(2) SEP-10, Circuit Alteration Procedure

C. (1) To prevent HPCI exhaust check valve damage
(2) 2OP-19, High Pressure Coolant Injection System Operating Procedure

D. (1) To prevent HPCI exhaust check valve damage
(2) SEP-10, Circuit Alteration Procedure

Answer: B

K/A:
295015 INCOMPLETE SCRAM
G2.04.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.3

Pedigree: Last used on the 2010-1 NRC Exam

Objective:
CLS-LP-019-A, Obj. 26g - Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event: High Suppression Pool water level

Reference: None

Cog Level: High

Explanation:
Very high lube oil temperatures can result in loss of lubricating qualities in the oil and thus cause damage to the bearings. Suction for HPCI is normally aligned to the Condensate Storage Tank (CST) if it is available. In accordance with the caution on LPC, the HPCI automatic suction transfer logic can be defeated to allow this lineup if suppression pool temperature is approaching 140°F. Step RC/L-23.
Defeat HPCI Hi Suppression Pool Level Suction Transfer is performed per SEP-10.

Distractor Analysis:

Choice A: Plausible because 2OP-19 contains direction for transferring the HPCI suction from the torus to the CST. (Section 8.9)

Choice B: Correct Answer, see explanation

Choice C: Plausible because HPCI operation <2100 rpms can cause exhaust check valve damage. 2OP-19 contains direction for transferring the HPCI suction from the torus to the CST. (Section 8.9)

Choice D: Plausible because HPCI operation <2100 rpms can cause exhaust check valve damage.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]

CAUTION

OPERATION OF HPCI OR RCIC WITH SUCTION TEMPERATURES ABOVE 140°F MAY RESULT IN EQUIPMENT DAMAGE

RC/L-23

The lube oil and control oil for both HPCI and RCIC are cooled by the water being pumped. Very high lube oil temperatures can result in loss of lubricating qualities in the oil and thus cause damage to the bearings.

Three different ranges for maintaining reactor water level are now specified. The appropriate action to take depends upon whether reactor water level was deliberately lowered. As constructed, these steps direct the operator to maintain reactor water level between the point to which it was deliberately lowered and LL-4 (thereby continuing to suppress reactor power while maintaining adequate core cooling), between 90 inches and LL-4 (maintaining adequate core cooling), or between the high reactor water level trip and LL-4 (likewise maintaining adequate core cooling).

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86. S295022 1

Unit One was at full power when all offsite power was lost.
The following is the Emergency Diesel Generator status:

DG1	Locked out on fault
DG2	Running and loaded
DG3	Running and loaded
DG4	Locked out on fault

Which one of the following completes the statements below?

The ___(1)___ CRD pump must be started to re-establish the CRD system.

___(2)___ contains the step for placing the CRD Flow Control, C11-FC-R600, in manual with manual potentiometer at minimum setting?

A. (1) 1A
(2) 1OP-08, Control Rod Hydraulic System Operating Procedure.

B. (1) 1A
(2) 0AOP-02, Control Rod Malfunction/Misposition.

C. (1) 1B
(2) 1OP-08, Control Rod Hydraulic System Operating Procedure.

D. (1) 1B
(2) 0AOP-02, Control Rod Malfunction/Misposition.

Answer: C

K/A:
295022 LOSS OF CRD PUMPS
AA2 Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS:
(CFR: 41.10 / 43.5 / 45.13)
02 CRD system status

RO/SRO Rating: 3.3/3.4

Pedigree: 08 NRC Exam

Objective:
CLS-LP-302G, Obj. 4c. Given plant conditions and any of the following AOP's, determine the required supplementary actions: AOP-36.1.

Reference: None

Cog Level: High

Explanation:
With a loss of all offsite power the E-Buses will strip the loads (CRD Pumps), there are no auto starts for these pumps, so both CRD pumps will be off. DG1 is lost which means E1 is lost and A CRD pump will not be able to be started. The steps for restart are located in the OP. The DG4 loss is a loss of the 2B CRD pump.

Distractor Analysis:

Choice A: Plausible because A CRD pump is tripped but the E-bus has no power for restarting the pump. Unit 2 has power for the 2A CRD pump.

Choice B: Plausible because A CRD pump is tripped but the E-bus has no power for restarting the pump and this condition is an entry condition for AOP-02 but it does not provide direction to perform this step. Unit 2 has power for the 2A CRD pump.

Choice C: Correct Answer, see explanation

Choice D: Plausible because this condition is an entry condition for AOP-02 but it does not provide direction to perform this step.

SRO Basis:

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]

The first part can be answered by knowing the power supply to the pump (systems knowledge.
The second part of the question (SRO Knowledge) is not systems knowledge, is not an immediate operator action, is not an entry condition for AOP/EOP, and is not purpose or mitigative strategy of the procedure. It assesses plant abnormal conditions and then selects a procedure to recover or with which to proceed.

LOSS OF ANY 4160V BUSES OR 480V E-BUSES	0AOP-36.1
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ATTACHMENT 4
Page 1 of 6

4160V and 480V Emergency Bus Loads

Bus E1	Load (kw)	Bus E2	Load (kw)
RHR Pump 1C	750	RHR Pump 1D	750
RHRSW Pump 1C	600	RHRSW Pump 1D	600
CS Pump 1A	940	CS Pump 1B	940
CRD Pump 1A	190	CRD Pump 1B	190
NSW Pump 1A	225	NSW Pump 1B	225
CSW Pump 1B	225	CSW Pump 1C	225
RHR Pump 2C	750	RHR Pump 2D	750
RHRSW Pump 2C	600	RHRSW Pump 2D	600
CSW Pump 2C	225	Fire Pump (normal)	190

6.3.20 Restarting CRD Hydraulic System Following Loss Of CRD Pump

1. **Ensure** the following Initial Conditions are met:
   - CRD System was in operation per Section 6.1.1. .... _______
   - The operating CRD pump has STOPPED. .... _______
2. **Close** B32-V22 (Seal Injection Vlv) for Recirc Pump A. .... _______
3. **Close** B32-V30 (Seal Injection Vlv) for Recirc Pump B. .... _______
4. **Place** C11-FC-R600 (CRD Flow Control) in MAN. .... _______
5. **Reduce** C11-FC-R600 (CRD Flow Control) potentiometer to minimum setting. .... _______
6. **Ensure** C11-PCV-F003 (Drive Press Vlv) is OPEN. .... _______
7. **Ensure** RBCCW is in operation to supply cooling water to CRD pumps. .... _______
8. **Start** non-operating (desired) CRD Pump A or B. .... _______

CONTROL ROD MALFUNCTION/MISPOSITION	0AOP-02.0
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2.0 SYMPTOMS

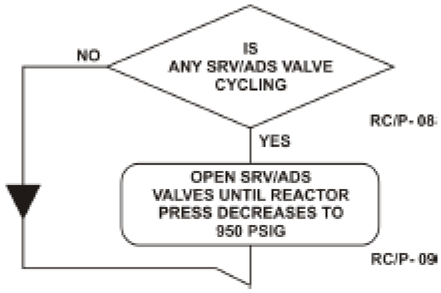
Any one of the following:

3. Inability to manually position control rods:
   a. Possible indications:
      - Loss of operating CRD pump

87. S295025 1

During an ATWS on Unit One the following annunciators/indications are observed:

A-05 (3-6) *Reactor Vess Hi Press Trip*
A-03 (1-10) *Safety / Relief Valve Open*
SRV A, C, F, and G are cycling open



Which one of the following completes the statements below?

The highest that reactor pressure reached was at least ___(1)___ psig.

The bases for Step RC/P-09 of LPC is to ___(2)___.

A. (1) 1060
(2) conserve SRV accumulator pressure

B. (1) 1060
(2) minimize heat discharged to the suppression pool

C. (1) 1130
(2) conserve SRV accumulator pressure

D. (1) 1130
(2) minimize heat discharged to the suppression pool

Answer: D

K/A:
295025 HIGH REACTOR PRESSURE
G2.04.45 Ability to prioritize and interpret the significance of each annunciator or alarm.
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

RO/SRO Rating: 4.1/4.3

Pedigree: Modified from 04 NRC Exam (added part one to the question to meet the k/a)

Objective:
LOI-CLS-LP-300-E, Obj. 11 - Given plant conditions and the Level/Power Control Procedure, determine the operator actions required to stabilize or reduce reactor pressure. (LOCT)

Reference: None

Cog Level: High

Explanation:
The SRVs listed all open at 1130 psig. The yellow lights are memory lights indicating that these had been open or are currently open. 1060 is the reactor trip signal. The bases for Step RC/P-30 is to conserve accumulator pressure (sustained opening of SRVs with no continuous pnuematic supply).

Distractor Analysis:

Choice A: Plausible because the reactor trip signal is 1060 and this is the bases for maintaining the SRVs open with a loss of pnuematic supply.

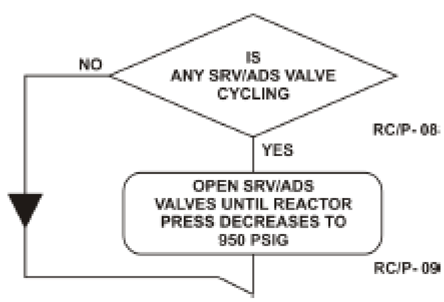
Choice B: Plausible because the reactor trip signal is 1060 and this is the bases for the step.

Choice C: Plausible because 1130 psig is correct and this is the bases for maintaining the SRVs open with a loss of pnuematic supply.

Choice D: Correct Answer, see explanation

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]
This question measures the SRO's assesment of high RPV pressure conditions and the knowledge of EOP symptom based steps used to prevent SRV cycling under high RPV pressure conditions.

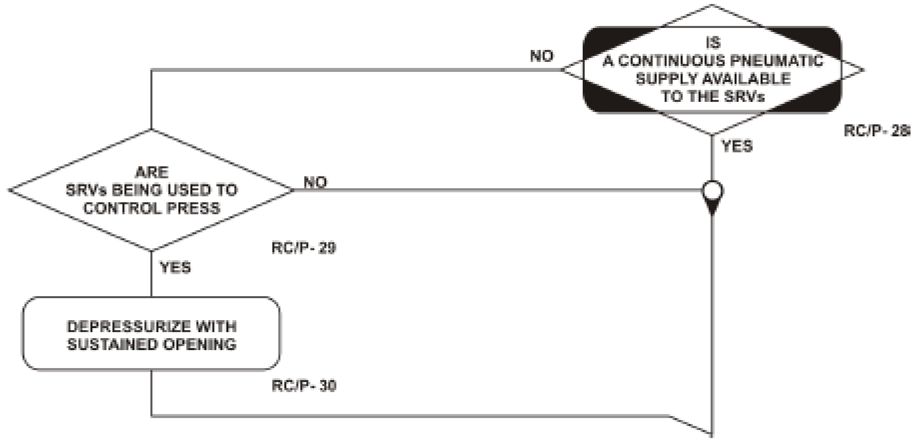
STEPS RC/P-08 and RC/P-09



STEP BASES:

If SRVs are used, they should be opened one at a time until a sufficient number have been opened to reduce reactor pressure at least to the value at which steam flow through the main turbine bypass valves is at 100% of bypass capacity (950 psig). If the MSIVs are open and the main turbine control system pressure regulator is in control, reducing pressure much below this value will cause the bypass valves to close. Heat which would otherwise be rejected to the main condenser would then be discharged to the suppression pool. If the MSIVs are not open or the bypass valve opening jack is in control, this value simply provides an adequate margin to the SRV lift setpoints.

STEPS RC/P-28 through RC/P-30



STEP BASES:

Sustained SRV opening, instead of permitting the valves to cycle, conserves accumulator pressure when the source of pressure to the SRV pneumatic supply is isolated or otherwise out of service. Such action to reduce the number of cycles on the SRVs prolongs SRV availability should more degraded conditions later require SRVs be opened for rapid depressurization of the RPV. However, the SRVs are operated so that the cooldown rate LCO of 100°F/hr is not exceeded.

88. S295026 1

Unit One is at rated power. 0AOP-30.0, Safety/Relief Valve Failures, has been entered for a stuck open SRV F and the supplementary actions are being performed.
The following torus temperatures are observed:

93° F	Suppression Pool Temp at location 45° on ERFIS
91° F	Suppression Pool Temp at location 90° on ERFIS
90° F	Suppression Pool Temp at location 135° on ERFIS
91° F	Suppression Pool Temp at location 180° on ERFIS
93° F	Suppression Pool Temp at location 225° on ERFIS
97° F	Suppression Pool Temp at location 270° on ERFIS
112° F	Suppression Pool Temp at location 315° on ERFIS
96.1° F	Blk Wtr Avg Supp Pool on CAC-TR-4426-1A

Which one of the following identifies:
(1) the required action IAW Technical Specification 3.6.2.1, Suppression Pool Average Temperature, and
(2) the consequences of pulling SRV F fuses in the incorrect order?

(Reference provided)

A. (1) Enter Condition A.
   (2) Loss of power to the SRV tailpipe temperature sensors.

B. (1) Enter Condition A.
   (2) The power sensing relay would be required to shift.

C. (1) Enter Condition D.
   (2) Loss of power to the SRV tailpipe temperature sensors.

D. (1) Enter Condition D.
   (2) The power sensing relay would be required to shift.

Answer: B

K/A:
295026 SUPPRESSION POOL HIGH WATER TEMPERATURE
EA2 Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13)
01 Suppression pool water temperature

RO/SRO Rating: 4.1/4.2

Pedigree: New

Objective:
LOI-CLS-LP-004-A, Obj 19 - Given plant conditions determine the required action(s) to be taken in accordance with Technical Specifications, associated with the Primary Containment System. (SRO/STA only) (LOCT)

Reference: TS 3.6.2.1 with LCO and applicability removed.

Cog Level: Hi

Explanation:
The operator will have to determine that the bulk water average temperatures are the tech spec required values not the individual azimuths readings. SRV F discharges at azimuth 310° in the torus.
Tech Spec 3.6.2.1 Condition A will be entered based on no testing and temp between 95 and 110°F. The note in the procedure states that removing the fuses out of order would cause the power sensing relay to shift. The fuses remove power to the SRV and acoustic monitor which provides green, red and amber lights for the SRV.

The torus temperatures bulk water average is calculated as follows:

Location	Weight	Temp	
4545A.	0.1734	93 93B.	16.1262
9090C.	0.1156	91 91D.	10.5196
135	0.1156	90	10.4040
180	0.1156	91	10.5196
225	0.1156	93	10.7508
270	0.1156	97	11.2132
315	0.1734	112	19.4208
AVG	0.0752	95.29	7.1655
Bulk Water Temp Average			96.1197

Distractor Analysis:

Choice A: Plausible because Condition A is correct and a loss of power would occur to the acoustic monitor but not the temperature sensors.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is a TS required action for 110°F in the suppression pool, but it is based on average water temperature not a local temperature and a loss of power would occur to the acoustic monitor but not the temperature sensors.

Choice D: Plausible because this is a TS required action for 110°F in the suppression pool, but it is based on average water temperature not a local temperature and the second part is correct.

SRO Basis:
Facility operating limitations in the technical specifications and their bases. [10 CFR 55.43(b)(2)]

Suppression Pool Average Temperature
3.6.2.1

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

a. ≤ 95°F with THERMAL POWER > 1% RTP and no testing that adds heat to the suppression pool is being performed;

b. ≤ 105°F with THERMAL POWER > 1% RTP and testing that adds heat to the suppression pool is being performed; and

c. ≤ 110°F with THERMAL POWER ≤ 1% RTP.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Suppression pool average temperature > 95°F but ≤ 110°F. AND THERMAL POWER > 1% RTP. AND Not performing testing that adds heat to the suppression pool.	A.1	Verify suppression pool average temperature ≤ 110°F.	Once per hour
	AND		
	A.2	Restore suppression pool average temperature to ≤ 95°F.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to ≤ 1% RTP.	12 hours
C. Suppression pool average temperature > 105°F. AND THERMAL POWER > 1% RTP. AND Performing testing that adds heat to the suppression pool.	C.1	Suspend all testing that adds heat to the suppression pool.	Immediately
D. Suppression pool average temperature > 110°F but ≤ 120°F.	D.1	Manually scram the reactor.	Immediately
	AND		
	D.2	Verify suppression pool average temperature ≤ 120°F.	Once per 30 minutes
	AND		
	D.3	Be in MODE 4.	36 hours

NOTE

Removing SRV fuses will de-energize the red and green indicating lights on Panel P601.. ☐

a. **Remove** fuses for the affected SRV in the order listed in Attachment 1, SRV Fuse Location/Number.. ☐

NOTE

SRV fuses should be removed in the listed order to ensure the power sensing relay is **NOT** required to shift... ☐

89. S295030 1

Unit Two is executing RVCP with the following conditions present:

Reactor water level	55 inches (N036/N037) and rising
Supp. chamber pressure	1.5 psig
Core Spray	One loop injecting at 3000 gpm
RHR	One loop injecting at 8800 gpm
Supp. pool level	-5 feet 7 inches
Supp. pool temp	178° F

Which one of the following actions are required IAW RVCP to ensure there is no Core Spray or RHR pump damage?

(Reference provided)

A. Raise Core Spray flow to 5000 gpm and shutdown the RHR pump(s).

B. Raise Core Spray flow to 5000 gpm and lower RHR flow to 8,000 gpm.

C. Raise Core Spray flow to 3600 gpm and shutdown the RHR pump(s).

D. Raise Core Spray flow to 3600 gpm and lower RHR flow to 8,000 gpm.

Answer: C

K/A:

295030 LOW SUPPRESSION POOL WATER LEVEL

EA2 Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13)

02 Suppression pool temperature

RO/SRO Rating: 3.9/3.9

Pedigree: Bank

Objective:

LOI-CLS-LP-300-B, Obj. 17 - Given plant conditions, determine if the NPSH and/or Vortex Limit has been exceeded IAW the NPSH and Vortex Limit Graphs.

Pedigree: 07 NRC Exam

Reference: 0EOP-01-UG, Attachment 5, Figure 5, 6, 9, 10, 11, and 12.

Cog Level: Hi

Explanation:

The NPSH curve must be lowered to at or below to the 0 psig curve because suppression pool level is below -31 inches. For each foot below -31 inches the pressure must be lowered 0.5 psig. 5 feet 7 inches (-67 inches) is 3 feet below -31 inches therefore the NPSH curve is 0 psig. This results in 3600 gpm for core spray. However a suppression pool level of -67 inches is below the RHR vortex limit of 5.25 feet on Unit Two (not for Unit One). Since water level is above TAF and RPV level is rising the SCO should direct securing the RHR pump(s).

Distractor Analysis:

Choice A: Plausible because core spray injection flow should be raised to maintain or restore RPV level, this value is representative of using 1.5 psig in the supp. chamber without the adjustment for level and RHR should be secured based on the vortex figure.

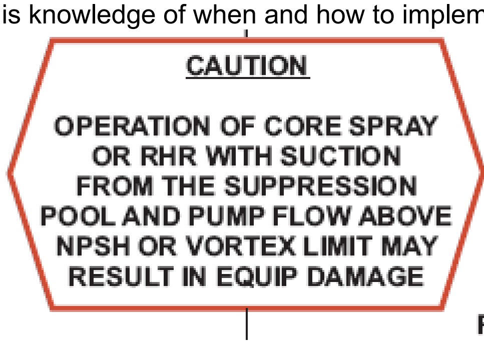
Choice B: Plausible because core spray flow should be raised to maintain or restore RPV level, this value is representative of using 1.5 psig in the supp. chamber without the adjustment for level.RHR flow is above the NPSH curve for the pumps, so this would be correct if the vortex curve was not exceeded.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the NPSH curve must be lowered to the 0 psig curve because suppression pool level is below -31 inches. The RHR flow value is representative of using 1.5 psig in the supp. chamber without the adjustment for level.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]

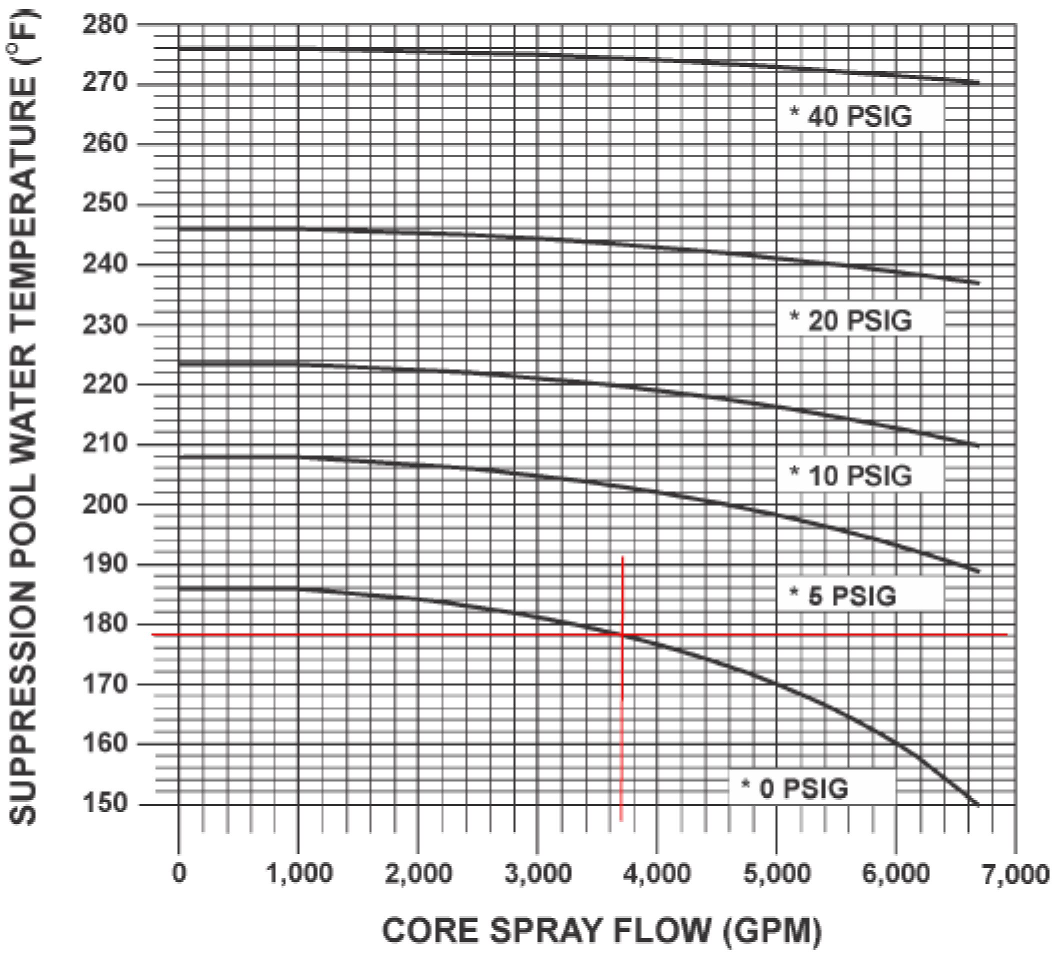
The question (SRO Knowledge) is not systems knowledge, is not an immediate operator action, is not an entry condition for AOP/EOP, and is not purpose or mitigative strategy of the procedure.
It is knowledge of when and how to implement attachments in the EOP network.



RC/L- 09

FIGURE 5

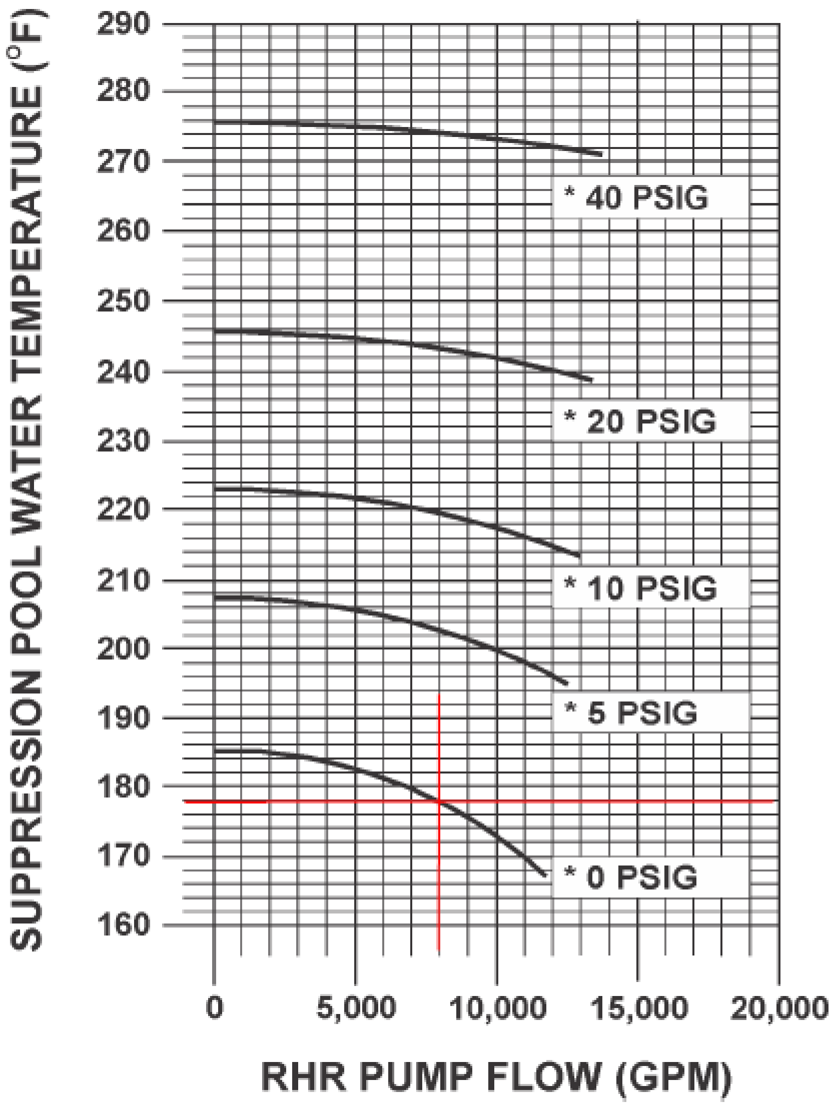
Core Spray NPSH Limit



SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

FIGURE 6

RHR NPSH Limit



SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

*SUPPRESSION CHAMBER PRESSURE (CAC-PI-1257-2A OR CAC-PI-1257-2B)

FIGURE 10
Unit 2 Core Spray Vortex Limit

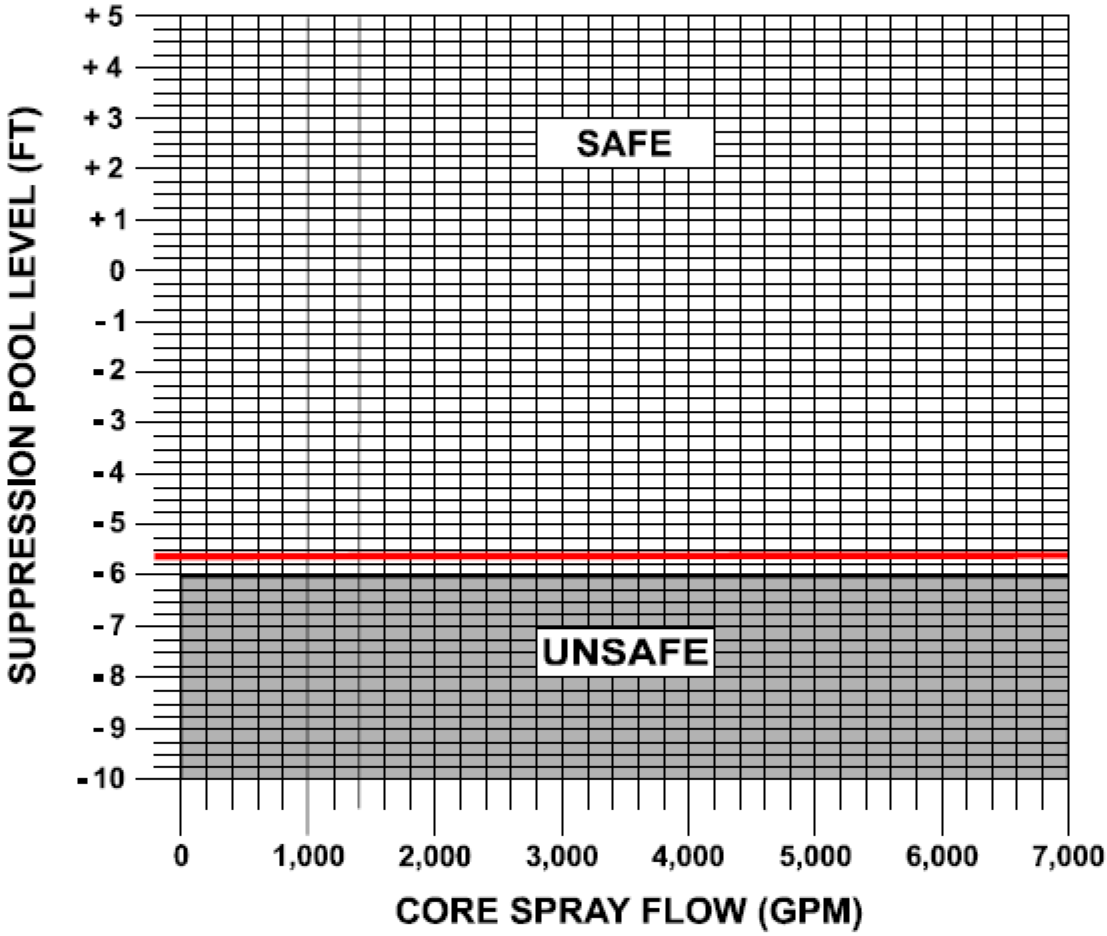
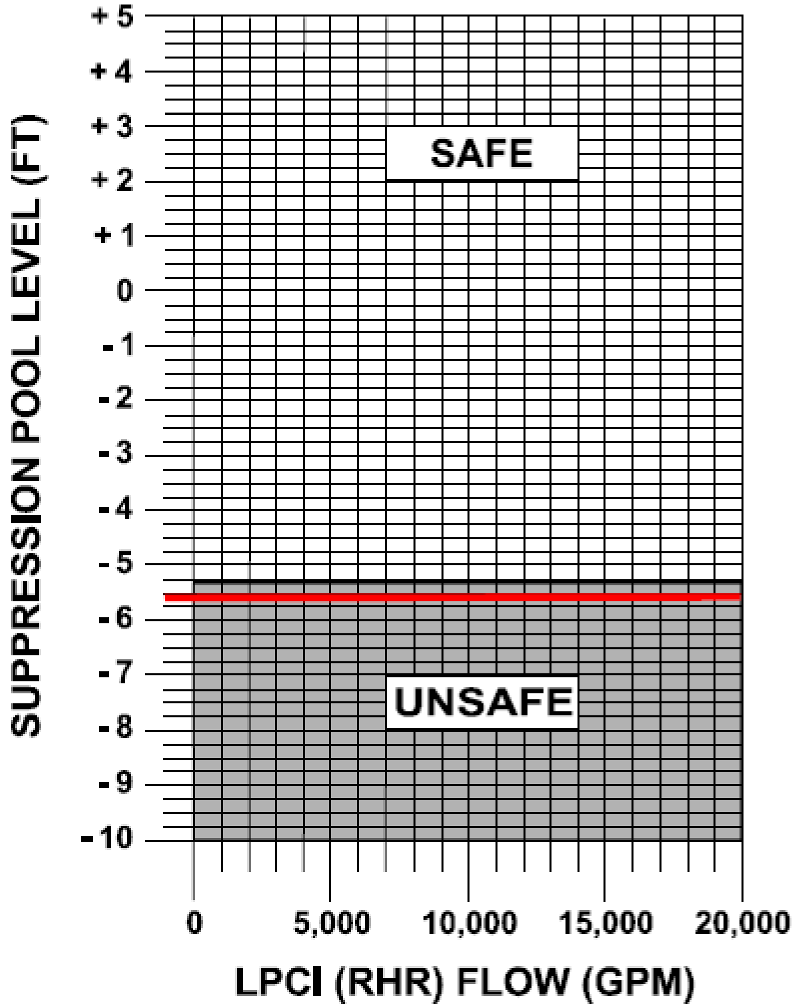


FIGURE 12
Unit 2 RHR Vortex Limit



90. S295031 1

Unit One is shutting down for a forced outage IAW GP-05, Unit Shutdown, due to a failing reactor recirculation pump seal #1.
RCIC is in day 4 of a 7 day LCO.
The reactor mode switch is placed in SHUTDOWN as directed by the procedure.
The startup level control valve fails and HPCI is manually placed in service to maintain RPV water level.
Reactor water level dropped to 150 inches before being restored and maintained in the normal band.

Which one of the following completes the statement below?

IAW 0OI-01.07, Notifications, this event meets the conditions for reportability within:

(Reference provided)

A. 1 hour ONLY.

B. 8 hours ONLY.

C. 1 hour AND 4 hours.

D. 4 hours AND 8 hours.

Answer: B

K/A:
295031 REACTOR LOW WATER LEVEL
G2.04.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

RO/SRO Rating: 2.7/4.1

Pedigree: New

Objective:
LOI-CLS-LP-201-D, Obj. 12 - Given plant conditions and an event, determine any applicable reporting requirements per OI-01.07, Notifications. (LOCT)

Reference: 0OI-01.07, Notifications, Attachment 1, Reportability Evaluation Checklist

Cog Level: Hi

Explanation:
Reactor water level has dropped to less that the RPS actuation signal (166 inches) thereby making this a 4 hour report. The NOTE states that manual initiation is reportable for HPCI system thereby making the HPCI injection even though it was manually accomplished an 8 hour report.

Distractor Analysis:

Choice A: Plausible because even though the plant is in a tech spec action statement this is not the reason for the shutdown. If tech specs would be the reason for the shutdown then this would be correct.

Choice B: Correct Answer, see explanation

Choice C: Plausible because even though the plant is in a tech spec action statement this is not the reason for the shutdown. If tech specs would be the reason for the shutdown then this would be correct. An RPS actuation signal is recieved but not while the unit was critical which makes the four hour incorrect.

Choice D: Plausible because an RPS actuation signal is recieved but not while the unit was critical which makes the four hour incorrect and an 8 hour report for the HPCI injection is correct.

SRO Basis:
The question is linked to a task that is labeled as an SRO-only task, and the task is NOT listed in the RO task list. These are not RO tasks.
The STA task is Determine Non Emergency Reportability Requirements Independent Of The SRO Determination per OI-1.07 and NUREG-1022.
The SRO task is Determine Reportability Requirements per 0OI-1.07 Independent Of The STA Determination.

2.0 FOUR-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
2.1			**NOTE:** Includes any Safety Limit violation (Tech Spec 2.2). Is plant shutdown required by technical specifications being initiated? [10 CFR 50.72(b)(2)(i)]
2.2			Has the event resulted in or should have resulted in an Emergency Core Cooling System (ECCS) discharge into the Reactor Coolant System as a result of a valid signal, except when the actuation resulted from and was part of a pre-planned sequence during testing or reactor operation? [10 CFR 50.72(b)(2)(iv)(A)]
2.3			Did the event or condition result in actuation of the reactor protection system (RPS) when the reactor was critical, except when the actuation resulted from and was part of a pre-planned sequence during testing or reactor operation? [10 CFR 50.72(b)(2)(iv)(B)]

3.0 EIGHT-HOUR REPORTABILITY			
ITEM #	YES	NO	DESCRIPTIVE QUESTION
3.1			Has the event or condition resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded? [10 CFR 50.72(b)(3)(ii)(A)]
3.2			Has the event or condition resulted in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety? [10 CFR 50.72(b)(3)(ii)(B)]
3.3			Did the event or condition result in valid actuation of any of the systems listed below except when the actuation resulted from and is part of a pre-planned sequence during testing or reactor operation? [10 CFR 50.72(b)(3)(iv)(A)]
			NOTE: Automatic OR Manual initiation of the systems listed below is reportable. NUREG-1022, Section 3.2.6 discussion, should be referenced for additional information.
			These systems are:
3.3.1			Reactor protection system (RPS) including: reactor scram and reactor trip. [10 CFR 50.72(b)(3)(iv)(B)(1)]
3.3.3			Emergency core cooling systems (ECCS), including: • Core Spray (CS) • High Pressure Coolant Injection (HPCI) • Low Pressure Coolant Injection (LPCI) function of the • Residual Heat Removal (RHR) • Automatic Depressurization (ADS) System [10 CFR 50.72(b)(3)(iv)(B)(4)]

91. S295033 1

A fuel bundle has been dropped in the Unit Two Spent Fuel Pool with area radiation values as indicated on Attachment 1, Area Radiation Monitoring.

Which one of the following completes the statements below?

An area ___(1)___ exceeded the Max Norm Operating Radiation Limit.

The Alternate Source Term Implementation analysis dictates that the maximum allowable time to manually start the Control Room Emergency Ventilation system is ___(2)___ minutes.

(Reference provided)

A. (1) has
   (2) 15

B. (1) has
   (2) 20

C. (1) has NOT
   (2) 15

D. (1) has NOT
   (2) 20

Answer: B

K/A:
295033 HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS
G2.02.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
(CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Pedigree: New

Objective:
LOI-CLS-LP-300-M, Obj. 06e - Given plant conditions and the Secondary Containment Control Procedure, determine if any of the following have been exceeded: Maximum normal operating radiation levels (LOCT)

Reference: 0EOP-01-UG Att. 10 Fig. 24, & Attachment 1, Area Radiation Monitoring

Cog Level: High

Explanation:
AOP-05, Radioactive Spills, High Radiation, and Airborne Activity, would be entered for the high rad conditions and for the fuel handling accident.
For the fuel handling accident the CREV system must be manually started within 20 minutes to ensure habitability of the main control room (Alternative Source Term analysis). The 15 minute requirement is contained in AOP-05, for isolating fire protection on indications of a HELB.
IAW SCCP the table indicates that the values of the indications are above the conditions for Max Norm. and some of these values are greater than the values for an Alert classification

Distractor Analysis:

Choice A: Plausible because the Max Norm rad levels have been violated and 15 minutes is the bases for closing the Fire protection valve that must be isolated for a HELB (this step is in the same procedure as starting the CREV system).

Choice B: Correct Answer, see explanation

Choice C: Plausible because the student may misread the log scales for the meters and determine it has not been exceeded and 15 minutes is the bases for closing the Fire protection valve that must be isolated for a HELB (this step is in the same procedure as starting the CREV system).

Choice D: Plausible because the student may misread the log scales for the meters and determine it has not been exceeded and 20 minutes is correct.

SRO Basis:
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]
This question also requires the examinee to know the bases for performing actions stated in the emergency/abnormal procedures.

The Alternative Source Term analyses assume that the Control Room is manually isolated and that the Control Building Emergency Recirculation System is initiated within 20 minutes of a Control Rod Drop Accident (CRDA) or Fuel Handling Accident

TABLE 3

AREA RADIATION LIMITS

PLANT AREA	PLANT LOCATION DESCRIPTION	ARM CHANNEL	MAX NORM OPERATING VALUE (mR/HR)	MAX SAFE OPERATING VALUE (mR/HR)
N CORE SPRAY	N CORE SPRAY ROOM	15	200	* 7000
S CORE SPRAY	S CORE SPRAY ROOM	16	200	* 7000
N RHR	N RHR ROOM	17	200	* 7000
S RHR	S RHR ROOM	18	200	* 3000
HPCI	HPCI ROOM	N/A	N/A	* 3000
RX BLDG 20 FT ELEV	N ACROSS FROM TIP ROOM	19	80	* 2000
	DRYWELL ENTRANCE	20		
	DECON ROOM	22		
	RAILROAD DOORS	23		
RX BLDG 50 FT ELEV	SAMPLE STATION	24	80	* 2000
	RX BLDG AIR LOCK	25		
RX BLDG 117 FT ELEV	N OF FUEL STORAGE POOL	27	80	* 7000
	BETWEEN RX & FUEL POOL	28	1000	7000
	CASK WASH AREA	29	90	* 7000
RX BLDG 80 FT ELEV	SPENT FUEL COOLING SYSTEM	30	90	* 3000

* CONTACT E&RC TO DETERMINE IF MAX SAFE OPERATING VALUE IS EXCEEDED

92. S295038 1

During accident conditions, the source term from the Unit One Turbine Building Ventilation must be estimated IAW 0PEP-03.6.1, Release Estimates Based Upon Stack/Vent Readings. Available data:

1-D12-RR-4548-3-2-1	Reading 7.425 E-1 μCi/cc
1-VA-FT-3358	Failed low (Turbine Building Vent Flow)
Turb Bldg HVAC	2 exhaust fans running
Sample results	1.284 E+4 μCi/sec (taken 1 hour ago)

Which one of the following identifies the highest emergency action level classification that is required for these conditions?

(Reference provided)

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

K/A:
295038 HIGH OFF-SITE RELEASE RATE
EA2 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.10 / 43.5 / 45.13)
03 Radiation levels

RO/SRO Rating: 3.5/4.3

Pedigree: New

Objective:
LOI-CLS-LP-301-A, Obj. 6 - Determine data required for offsite dose projection in accordance with PEP-03.4.7, Automation of Offsite Dose Projection, and PEP-03.6.1, Release Estimates Based Upon Stack/Vent Readings. (LOCT)
Reference: 0PEP-03.6.1 Attachment 3 and 0PEP-02.1

Cog Level: High

Explanation:
Per Attachment 3 the estimated release is calculated as follows:
Monitor reading (μCi/cc) X Flow (15,500) X Conversion factor (472)
7.425 E-1 X 15,500 X 472 = 5.4321 E+6 μCi/sec
The threshold for an Alert is a value greater than 3.49 E+5 μCi/sec.

Distractor Analysis:

Choice A: Plausible because this is the EAL classification for the sample results.

Choice B: Correct Answer, see explanation

Choice C: Plausible because if 15,500 per exhaust fan is used the calculation would be 1.0864 E+7 which would make a site area emergency correct.

Choice D: Plausible because If 7.425 E+1 is used the calculation would be 5.4321 E+8 which would make a general emergency correct.

SRO Basis:
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

ATTACHMENT 3
Page 1 of 1
Source Term Calculation From #1 Turbine Vent

Release rate is read in μCi/sec directly from 1-D12-RR-4548-4 (effluent channel) when the 1-VA-FT-3358 flow instrument loop is operational. The following calculations are necessary when this loop is not operational.

TIME	MONITOR READING(1) (μCi/cc)	FLOW(2) (cfm)	CONVERSION FACTOR cc/sec / cfm	RELEASE RATE(3) (μCi/sec)
			472	

(1) The monitor automatically selects the most accurate operational channel, either low, mid, or high range. Read the μCi/cc from the appropriate channel (low, mid, or high) of 1-D12-RR-4548-3-2-1.

(2) If not available, use 15,500 cfm

(3) Release Rate (μCi/sec) = (μCi/cc) x (cfm) x (472)

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad Monitor	D12-RM-23S	2.13E+09 μCi/sec	2.13E+08 μCi/sec	1.96E+07 μCi/sec	1.80E+06 μCi/sec
Gaseous	Reactor Bldg Vent Noble Gas Monitor	CAC-AQH-1264-3	--------	--------	--------	6.14E+04 cpm
Gaseous	Turbine Building Vent Rad Monitor	D12-RM-23	1.07E+08 μCi/sec	1.07E+07 μCi/sec	3.49E+05 μCi/sec	1.13E+04 μCi/sec
Liquid	Service Water Effluent Radioactivity Monitor	D12-RM-K605	--------	--------	200 X high alarm	2 X high alarm
Liquid	Radwaste Effluent Rad Monitor	D12-RM-K604	--------	--------	200 X hi-hi alarm	2 X hi-hi alarm

93. S400000 1

3.7.2 Service Water (SW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 SW System and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

IAW Technical Specification 3.7.2, Service Water (SW) System and Ultimate Heat Sink (UHS), which one of the following inoperable pumps, by itself, would require entry into an action statement?

A. CSW Pump 1B

B. CSW Pump 2B

C. NSW Pump 1B

D. NSW Pump 2B

Answer: A

K/A:

400000 COMPONENT COOLING WATER SYSTEM (CCWS)

A2 Ability to (a) predict the impacts of the following on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

01 Loss of CCW pump

RO/SRO Rating: 3.3/3.4

Pedigree: New

Objective:

LOI-CLS-LP-043, Obj. 18 - Given plant conditions associated with the Service Water system determine the required action(s): b. to be taken IAW Technical Specifications. (LOCT) (SRO/STA Only)

Reference: none

Cog Level: Fundamental

Explanation:

Each unit has two NSW pumps and 3 are required IAW TS 3.7.2. So having one of these pumps inoperable does not require entry into the TS action statements. Each unit has 3 CSW pumps. IAW the bases for Unit 1 it requires specifically the 1A and 1B CSW pumps while Unit 2 bases specifically requires the 2A and 2C CSSW pumps.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this pump inoperable on the other unit would make this correct.

Choice C: Plausible because one of the two pumps on the unit is inoperable, but theere are still 3 pumps between the units combined.

Choice D: Plausible because one of the two pumps on the unit is inoperable, but theere are still 3 pumps between the units combined.

SRO Basis:
Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]
This requires knowledge of the bases document in what pumps are required for the system to be considered operable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required NSW pump inoperable for reasons other than Condition A.	B.1	--------------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.8.1 for DGs made inoperable by NSW. ------------------------------------- Restore required NSW pump to OPERABLE status.	7 days <u>AND</u> 14 days from discovery of failure to meet LCO
C. One required conventional service water (CSW) pump inoperable.	C.1	Verify the one OPERABLE CSW pump and one OPERABLE Unit 1 NSW pump are powered from separate 4.16 kV emergency buses.	Immediately
	<u>AND</u>		
	C.2	Restore required CSW pump to OPERABLE status.	7 days <u>AND</u> 14 days from discovery of failure to meet LCO

LCO

In the event of a DBA, the NSW header and associated components are adequate to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. However, the CSW header and associated components are required to ensure maximum reliability in the event of a single failure. To ensure this requirement is met, the appropriate equipment to supply the unit NSW and CSW headers must be OPERABLE. In addition, at least three site NSW pumps are required to ensure adequate NSW pump redundancy is available to ensure cooling to the DGs in the event of an active single failure.

The SW System is considered OPERABLE when it has two OPERABLE CSW pumps (specifically the CSW 1A and CSW 1B pumps), three site NSW pumps (any combination of Unit 1 and Unit 2 NSW pumps), and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the ECCS equipment and the DGs. In addition, for a site NSW pump to be considered OPERABLE, it must be capable of supplying its associated unit NSW header. For a CSW pump to be considered OPERABLE, it must be capable of supplying the CSW header and the NSW header. The CSW 1A and CSW 1B pumps must be OPERABLE to ensure that two pumps can supply the CSW header with one specific postulated single failure, Loss of Division II power. This failure disables both the flow path from the NSW header to the RHRSW pumps and it disables the CSW 1C pump.

The OPERABILITY of the UHS is based on having a minimum water level in the pump well of the intake structure of -6 ft mean sea level and a maximum UHS 24-hour average water temperature of 90.5°F with the maximum UHS actual water temperature not to exceed 92°F.

(continued)

LCO

In the event of a DBA, the NSW header and associated components are adequate to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. However, the CSW header and associated components are required to ensure maximum reliability in the event of a single failure. To ensure this requirement is met, the appropriate equipment to supply the unit NSW and CSW headers must be OPERABLE. In addition, at least three site NSW pumps are required to ensure adequate NSW pump redundancy is available to ensure cooling to the DGs in the event of an active single failure.

The SW System is considered OPERABLE when it has two OPERABLE CSW pumps (specifically the CSW 2A and CSW 2C pumps), three site NSW pumps (any combination of Unit 1 and Unit 2 NSW pumps), and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the ECCS equipment and the DGs. In addition, for a site NSW pump to be considered OPERABLE, it must be capable of supplying its associated unit NSW header. For a CSW pump to be considered OPERABLE, it must be capable of supplying the CSW header and the NSW header. The CSW 2A and CSW 2C pumps must be OPERABLE to ensure that two pumps can supply the CSW header with one specific postulated single failure, Loss of Division II power. This failure disables both the flow path from the NSW header to the RHRSW pumps and it disables the CSW 2B pump.

The OPERABILITY of the UHS is based on having a minimum water level in the pump well of the intake structure of -6 ft mean sea level and a maximum UHS 24-hour average water temperature of 90.5°F with the maximum UHS actual water temperature not to exceed 92°F.

(continued)

94. SG2.01.39 1

Which one of the following events would require you to direct a Reactor Scram in order to maintain safe operation of the facility?

1. With Unit Two in MODE 1 an electrical fire has resulted in erratic or questionable indications on numerous main control room nuclear instruments

2. An earthquake has occurred in which the National Earthquake Center reports horizontal ground accelerations of 0.08 g were registered. Operating Basis Earthquake (OBE) exceedance light is energized. All other plant indications indicate the plant is currently stable.

3. Power is at 50% and power ascension is in progress after a refuel outage. An accident on the Refuel Floor involving spent fuel has caused the Shift Manager to declare a Site Area Emergency.

4. Department of Homeland Security has increased the National Threat Advisory System (NTAS) Level to elevated threat for Brunswick County.

A. Event 1

B. Event 2

C. Event 3

D. Event 4

Answer: A

K/A:
G2.01.39 Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 3.6/4.3

Pedigree: Bank, modified from Browns Ferry bank

Objective:
LOI-CLS-LP-201-C, Obj. 14 -

Reference: None

Cog Level: High

Explanation:
With an electrical fire and errratic operation of key equipment/indications then a manual scram would be required.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because a severe event has occurred.

Choice C: Plausible because a declaration of a SAE does not always necessitate a reactor scram.

Choice D: Plausible because on an immenent threat the reactor will be scrammed.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

From Pre Fire Plans:

3.2.5 **MONITOR** RTGB for affects from fire such as auto starts and actuations. ☐

3.2.6 **REFERENCE** the pre-fire plan. ☐

3.2.7 **WHEN** informed by the Shift Incident Commander, **THEN ANNOUNCE** the location of the Command Post. ☐

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From ASSD-01, Operator Actions:

3.4 **IF** the Unit CRS determines the ability to confirm reactor power less than 2% is in jeopardy, **THEN PERFORM** the following:

3.4.1 **MANUALLY SCRAM** the reactor. ☐

3.4.2 **CONFIRM** reactor power is less than 2% using one of the following:

– Neutron Monitoring System ☐

– Rod Position Indication System ☐

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Threat Conditions and EP Actions for Power Reactors

NORMAL – Low risk of terrorist activities

ELEVATED THREAT – General risk of attack

IMMINENT THREAT – High risk of terrorist attack (this differs from the notification of an aircraft threat (see EALs)

Elevated Threat

1. Upon notification from Security that the Department of Homeland Security (DHS) has increased the National Threat Advisory System (NTAS) Level to **Elevated Threat**, EP should perform the following actions:

   a. **Verify** the EOF, JIC, OSC, and TSC are available and ready for use.

   b. **Contact** the Control Room to **verify** they are aware of the increase in threat level, let them know that the following communications will be made, **and review** the methods available for ERO notifications

   c. **Notify** ERO personnel via the ERO Notification System of the change in threat level. The following may be used as the script. The National Threat Advisory System Level has been increased to ELEVATED THREAT. There is no known credible threat to the (*insert plant*) Plant at this time. ERO Members should be sensitive to the possibility of activation.

   d. **Distribute** a site-wide e-mail. The following text may be used:

      "Homeland Security National Threat Advisory System Level has been elevated to ELEVATED THREAT. Though there is **NO** known credible threat to (*insert plant)*, we must be prepared at all times to quickly and efficiently respond to ERO callouts. Security events could cause us to activate our facilities (primary or remote) with the same degree of urgency as the radiological scenarios that we commonly practice. All ERO members are reminded to have your company picture identification badges with you when not at the site. The badges will be needed for rapid access to the emergency response facilities, and possibly through law enforcement traffic control points. If there is a security threat on plant site or your normal facility is inaccessible, then you may be directed to report to the Remote Emergency Response Facility on (*insert location of remote facility)*. JIC Staff Members will report to *(insert location)* location under all conditions."

   e. **Test** the notification systems to ensure contact can be made with at-risk county and State EOCs and /or warning points.

   f. **Contact** the Main Control Room to let them know that the above items have been completed.

95. SG2.02.03 1

Which one of the following completes the statements below concerning the Unit Two Turbine Bypass System?

The **minimum** number of inoperable turbine bypass valves that would require entry into an action statement of LCO 3.7.6, Main Turbine Bypass System, is ___(1)___.

The capacity of the Unit Two Turbine Bypass System is ___(2)___%.

A. (1) two
   (2) 20.6

B. (1) two
   (2) 69.6

C. (1) three
   (2) 20.6

D. (1) three
   (2) 69.6

Answer: D

K/A:

G2.02.03 (multi-unit license) Knowledge of the design, procedural, and operational differences between units. (CFR: 41.5 / 41.6 / 41.7 / 41.10 / 45.12)

RO/SRO Rating: 3.8/3.9

Pedigree: New

Objective:

LOI-CLS-LP-25 Obj. 9, Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine whether given plant conditions meet minimum Technical Specifications, the TRM or ODCM requirements for the Main Steam system

Reference: None

Cog Level: memory

Explanation:

From the bases document Turbine Bypass System is inoperable if 2 Bypass valves are inoperable on U1 and 3 bypass valves are inoperable on U2. The U2 system contains 10 bypass valves while U1 only has 4 bypass valves.

Distractor Analysis:

Choice A: Plausible because U1 would be inoperable with this number of inoperable bypass valves and this is the capacity of Unit 1.

Choice B: Plausible because U1 would be inoperable with this number of inoperable bypass valves and this is the capacity of Unit 2.

Choice C: Plausible because U2 is inoperable and this is the capacity of Unit 1.

Choice D: Correct Answer, see explanation

SRO Basis:
Facility operating limitations in the TS and their bases [10 CFR 55.43(b)(2)]

From Unit One Bases:

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Turbine Bypass System

BASES

BACKGROUND	The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 20.6% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of four valves connected to the main steam lines between the main steam isolation valves and the turbine stop valves. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the UFSAR, Section 7.7.1.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the Speed Control System or load limit restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows through connecting piping and bypass valve pressure reducers to the condenser.
ACTIONS	A.1 If the Main Turbine Bypass System is inoperable (two or more bypass valves as specified in the COLR inoperable), and the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR, MCPR, and LHGR limits accordingly. The 4 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

From Unit Two Bases:

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 69.6% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of ten valves connected to the main steam lines between the main steam isolation valves and the turbine stop valves. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the UFSAR, Section 7.7.1.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the Speed Control System or load limit restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows through connecting piping and bypass valve pressure reducers to the condenser.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (three or more bypass valves as specified in the COLR inoperable), and the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR, MCPR, and LHGR limits accordingly. The 4 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

From SD-25:

2.10 Bypass Valve Steam Chests

The Bypass Valve (BPV) Steam Chests (Figure 25-1, 25-1A and 25-9) provide a cross-connect point between the four steam lines such that any combination of steam lines may be in service and still provide a path directly to the Main Condenser. The bypass valve positioning is controlled by the Electro-Hydraulic Control (EHC) System depending upon available reactor pressure and desired turbine loading. A significant difference exists between Unit 1 and Unit 2 bypass capacities.

Unit 1 has one bypass chest connecting all 4 steam lines and containing 4 bypass valves for a total steam bypass capacity of 20.6%. Unit 2 has two bypass chests, each containing 5 bypass valves (10 valves total) for a total capacity of 69.6%. Either chest in Unit 2 cross-connects all four Main Steam Lines. The EHC System and Bypass Valve positioning is discussed in more detail in the EHC (SD-26.2 and SD-26.3).

96. SG2.02.37 1
The following information was obtained during the last scram timing for control rod 18-19 IAW 0PT-14.2.1, Single Rod Scram Insertion Times Test.

Control Rod	Insertion	Position Notch	Time (Secs)
18-19	5%	46	0.438
	20%	36	1.188
	50%	26	2.026
	90%	6	3.349

Unit One is operating at rated power when control rod 18-19 scram accumulator has depressurized and cannot be repaired for two days.

All other control rods and control rod scram accumulators are operable.

Concerning control rod 18-19, which one of the following completes the statements below?

The scram times ___(1)___ within Technical Specification 3.1.4, Control Rod Scram Times.

IAW Technical Specification 3.1.5, Control Rod Scram Accumulators the control rod ___(2)___ be declared SLOW.

(Reference provided)

A. (1) are
   (2) can

B. (1) are
   (2) cannot

C. (1) are NOT
   (2) can

D. (1) are NOT
   (2) cannot

Answer: D
K/A:
G2.02.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 3.6/4.6

Pedigree: Modified from 08 NRC Exam, made the control rod slow so that it must be declared inoperable

Objective:
CLS-LP-08, Obj. 18. Given plant conditions and TS, including the bases, TRM, ODCM, and COLR, determine the required actions to be taken in accordance with TS associated with CRD system. (SRO/STA Only)

Reference: TS 3.1.5 and TS 3.1.4

Cog Level: high

Explanation:
IAW TS 3.1.4 Since the last scram timing was outside of the limits for notches 26 and 36 but within the total allowable time of 7 seconds this rod would be declared SLOW based on this TS. If the examinee adds the times together it will be outside of the 7 seconds and could think that the rod is inoperable.

Control rod scram accumulators shall be operable in Modes 1and 2.

One control rod scram accumulator inoperable with reactor steam dome pressure >950 psig the required action is to declare the associated control rod scram time slow (only applicable if it was within the limits of Table 3.1.4-1 during the last scram time surv.) or declare the associated control rod inoperable within 8 hours. Since it was slow this means it must be declared inop.

Distractor Analysis:

Choice A: Plausible because if the scram times were good then this answer could be correct.

Choice B: Plausible because if the scram times were good then this answer could be correct.

Choice C: Plausible because if the examinee looks at only the 3.1.4 TS then the rod is slow.

Choice D: Correct Answer, see explanation.

SRO Basis:
Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 950 psig.	A.1	---------------NOTE------------ Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ---------------------------------- Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>		
	A.2	Declare the associated control rod inoperable.	8 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4
a. No more than 10 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and
b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

------------------------------NOTES------------------------------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.3, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES WHEN REACTOR STEAM DOME PRESSURE $\geq$ 800 psig[a][b] (seconds)
46	0.44
36	1.08
26	1.83
06	3.35

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) When reactor steam dome pressure is < 800 psig, established scram time limits apply.

97. SG2.03.04 1

During accident conditions, an auxiliary operator is needed to enter the reactor building for local emergency actions to prevent fuel damage. Due to elevated reactor building radiation levels, it is estimated the operator will receive 7.5 rem.

Which one of the following completes the statements below?

The estimated dose of 7.5 rem ___(1)___ exceed EPA-400 limits.

The Site Emergency Coordinator ___(2)___ authorize exceeding 10CFR20 limits IAW 0PEP-3.7.6, Emergency Exposure Controls.

A. (1) will not
   (2) can

B. (1) will not
   (2) cannot

C. (1) will
   (2) can

D. (1) will
   (2) cannot

Answer: A

K/A:

G2.03.04 Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

RO/SRO Rating: 3.2/3.7

Pedigree: 07 NRC Exam

Objective:

CLS-LP-102-A, Obj. 11 - State the emergency worker exposure limits listed in EPA 400 for each of the following conditions:
b. Protection of valuable property

Reference: None

Cog Level: Memory

Explanation:

Per PEP-03.7.6, emergency limits follow EPA-400 guidelines of 10 rem for protection of valuable property and 25 rem for life saving action. Exceeding 10 CFR 20 limits (5 rem) requires authorization of the SEC for on site activities.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: incorrect because it may be authorized by the SEC.

Choice C: incorrect because the 7.5 rem does not exceed the EPA-400 limit.

Choice D: incorrect because the 7.5 rem does not exceed the EPA-400 limit and it may be authorized by the SEC.

SRO Basis:
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

4.0 RESPONSIBILITIES

4.2 The Site Emergency Coordinator is responsible for authorization of exposures in excess of 10CFR20 limits and approval of the administration of potassium iodide (KI) for station ERO personnel performing onsite functions.

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ATTACHMENT 2
Page 1 of 3
Emergency Exposure Guidelines

Exposure guidelines in this attachment are consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides described in EPA 400-R-92-001.

Every reasonable effort will be used to ensure that an emergency is handled in such a manner that no worker exceeds the normal exposure limits, including the administering of radioprotective drugs. In emergency situations, workers may receive exposure under a variety of circumstances in order to assure safety and protection of others and of valuable property. These exposures will be justified if the maximum risks or costs by the actions outweigh the risks to which the workers are subjected. The Emergency Worker Dose Limit Guidelines are as follows:

Dose Limit (Rem TEDE)	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Lifesaving or protection of large populations	Lower dose not practicable
> 25	Lifesaving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved.

98. SG2.03.14 1

Unit Two is shutdown to support drywell entry due to Recirculation Pump oil level concerns. Reactor coolant temperature is 200°F.

E&RC has determined that the drywell atmosphere is **not** suitable for unfiltered release.

Which one of the following completes the statements below IAW 2OP-24, Section 6.3.13, Primary Containment Purging (Deinerting) Through the SBGT System?

This section ___(1)___ be performed under the current plant conditions.

If drywell pressure was above 0.7 psig, deinerting could result in ___(2)___.

A. (1) can
   (2) contamination of the RB 50'

B. (1) cannot
   (2) contamination of the RB 50'

C. (1) can
   (2) exceeding ODCM Main Stack release rates

D. (1) cannot
   (2) exceeding ODCM Main Stack release rates

Answer: A

K/A:

G2.03.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)

RO/SRO Rating: 3.4/3.8

Pedigree: 10-2 NRC Exam

Objective:

CLS-LP-10 Obj. 9. Determine the effect that the following will have on SBGT:
j.High Drywell Pressure

Reference: None

Cog Level: High

Explanation:

ONLY allowed to purge through purge fans if drywell atmosphere is suitable for unfiltered release. Must purge through SBGT and procedure ONLY allows purge through SBGT in Mode 4 due to LOCA concerns. Drywell pressure above .7 psig can cause the loss of the SBGT System water seal which will cause the RB 50' to be contaminated with airborne activity.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because contaminating the RB 50' is correct and deinerting the drywell cannot be commenced greater than 24 hours before the unit being gets 15% power. Based on not suitable for unfiltered release, deinerting will be delayed until Mode 4 is reached.

Choice C: Plausible because higher drywell pressure would provide for higher Main Stack release rates but the Main Stack Rad Hi-Hi isolation is set to ensure ODCM release rates will not be exceeded and Mode 4 is correct.

Choice D: Plausible because higher drywell pressure would provide for higher Main Stack release rates but the Main Stack Rad Hi-Hi isolation is set to ensure ODCM release rates will not be exceeded and deinerting the drywell cannot be commenced greater than 24 hours before the unit gets below 15% power. Based on not suitable for unfiltered release, deinerting will be delayed until Mode 4 is reached.

SRO Basis:
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

99. SG2.04.27 1

A fire in the control building fire area requires entry into 0ASSD-01, Alternative Safe Shutdown Procedure Index. The CRS has determined that alternate safe shutdown actions are required. Both Unit One and Unit Two have been manually scrammed.

Which one of the following completes the statements below IAW 0ASSD-01?

The next action that is required is to ___(1)___.

Following this action both units will ___(2)___.

A. (1) place MSIV control switches in close
   (2) perform 0ASSD-01, Alternative Safe Shutdown Procedure Index concurrently with 0ASSD-02, Control Building.

B. (1) trip both Reactor Recirc pumps
   (2) perform 0ASSD-01, Alternative Safe Shutdown Procedure Index concurrently with 0ASSD-02, Control Building.

C. (1) place MSIV control switches in close
   (2) exit 0ASSD-01, Alternative Safe Shutdown Procedure Index and enter 0ASSD-02, Control Building

D. (1) trip both Reactor Recirc pumps
   (2) exit 0ASSD-01, Alternative Safe Shutdown Procedure Index and enter 0ASSD-02, Control Building

Answer: C

K/A:
G2.04.27 Knowledge of “fire in the plant” procedures. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.4/3.9

Pedigree: 08 NRC Exam

Objective:
CLS-LP-304, Obj. 12. Given plant conditions with an ASSD fire and the ASSD procedures, determine the appropriate operator actions to be performed for the fire.

Reference: None

Cog Level: High

Explanation:
The tripping of the Recirc pumps is required for AOP-32 Plant Shutdown from Outside the Control Room, therefore a plausible option.The only action directed from the applicable section of ASSD-01 is to place the MSIV control switches to close. For a fire in the control building ASSD-01 is exited, for the other areas the procedure is perform concurrently.

Distractor Analysis:

Choice A: Plausible because number one is correct and if the fire was in a different area then perfoming concurrently would be correct.

Choice B: Plausible because tripping of the recirc pumps is an action that is performed on a control room evacuation and if the fire was in a different area then perfoming concurrently would be correct.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because tripping of the recirc pumps is an action that is performed on a control room evacuation and exiting the procedure is correct.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]

3.5.2 **IF** the fire is in the Control Building fire area, **THEN PERFORM** the following:

a. **MANUALLY SCRAM** Unit 1 reactor. ☐

b. **PLACE** Unit 1 MSIV control switches in *CLOSE*. ☐

c. **MANUALLY SCRAM** Unit 2 reactor. ☐

d. **PLACE** Unit 2 MSIV control switches in *CLOSE*. ☐

e. **OBTAIN** Control Room controlled copy of the Plant Emergency Procedures (PEPs) manual. ☐

f. Both units **EXIT** this procedure **AND ENTER** 0ASSD-02, Control Building. ☐

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From AOP-32

4.1 Immediate Actions (continued)

2. **<u>WHEN</u>** Control Room evacuation is determined to be required, **<u>THEN</u> complete** as many of the following actions as possible in the sequence listed prior to the evacuation:

   a. **Insert** a manual scram. .... ☐

   b. **<u>Unit 1 Only</u>: Place** Reactor Mode Switch to SHUTDOWN. .... ☐

   c. **<u>Unit 2 Only</u>: <u>WHEN</u>** steam flow is less than 3×10^6 lb/hr, **<u>THEN</u> place** Reactor Mode Switch to SHUTDOWN. .... ☐

   d. **Trip** the main turbine. .... ☐

CAUTION

Auxiliary power should automatically transfer from the UAT to the SAT. If transfer does **<u>NOT</u>** occur, **<u>AND</u>** manual actions are taken to restore the buses, reenergizing the SAT may result in auto start of plant equipment such as CWIPs, condensate pumps, and condensate booster pumps. .... ☐

   e. **Observe** auxiliary power transferred to the SAT. .... ☐

   f. **Trip** Recirc VFD 1(2)A using the Emerg Stop pushbutton. .... ☐

   g. **Trip** Recirc VFD 1(2)B using the Emerg Stop pushbutton. .... ☐

100. SG2.04.45 1
The following alarms and indications exist on Unit One:

A-05 (5-6) *Pri Ctmt Press Hi Trip* is in alarm
A-03 (6-9) *Reactor Low Wtr Level Initiation* is in alarm
A-03 (5-1) *Auto Depress Timers Initiated* is in alarm
Reactor coolant sample yields a result of 310 µCi/gm Iodine-131 Inboard and Outboard MSIV logic lights are illuminated
No area radiation or temperatures are above Max Normal Operating Levels

Which one of the following completes the statement below?

These alarms and indications establish that a loss of the _____ exists.

(Reference provided)

A. Containment AND Fuel Clad Barriers ONLY

B. Reactor Coolant System AND Fuel Clad Barriers ONLY

C. Reactor Coolant System AND Containment Barriers ONLY

D. Containment, Reactor Coolant System AND Fuel Clad Barriers

Answer: B
K/A:
G2.04.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

RO/SRO Rating: 4.1/4.3

Pedigree: New

Objective:
LOI-CLS-LP-301-B, Obj. 7 - Identify how each of the following symptoms and indicators relate to one or more fission product barrier loss or potential loss: a. Coolant Activity d. Containment Pressure g. Containment Isoation Status

Reference: 0PEP-02.1, Brunswick Nuclear Plant Initial Emergency Actions

Cog Level: High

Explanation:
Pri Ctmt Press Hi Trip alarm indicates that Drywell pressure is $\geq$ 1.7 psig which is a loss of the RCS barrier. The Iodine levels of the sample indicate a loss of the Fuel Cladding Barrier. While there is a failure of the Group 1 to isolate (level less than LL3) there are no given indications of the system discharging outside of its normal pathway, so a loss of the containment barrier does not exist.

Distractor Analysis:

Choice A: Plausible because a Group 1 isolation has failed and if there were indications of this discharging outside of its normal pathway then the Containment barrier would be lost and the Fuel Clad barrier is lost based on iodine levels.

Choice B: Correct Answer, see explanation

Choice C: Plausible because a Group 1 isolation has failed and if there were indications of this discharging outside of its normal pathway then the Containment barrier would be lost and the RCS barrier is lost based on drywell pressure

Choice D: Plausible because a Group 1 isolation has failed and if there were indications of this discharging outside of its normal pathway then the Containment barrier would be lost and the RCS barrier is lost based on drywell pressure and the Fuel Clad barrier is lost based on iodine levels.

SRO Basis:
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

	Fuel Clad Barrier	
	Loss	**Potential Loss**
A. RPV Level	1. Primary Containment Flooding required due to any of the following: - RPV level cannot be restored and maintained above -57.5 inches (Jet Pump Suction) with at least one core spray pump injecting into the reactor vessel - RPV level cannot be restored and maintained above LL-4 (MSCRWL) - RPV water level cannot be determined and RPV flooding conditions cannot be maintained	1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined
B. PC Pressure / Temperature	None	None
C. Isolation	None	None
D. Rad	2. Drywell radiation > 2,000 R/hr 3. Primary coolant activity > 300 μCi/gm I-131dose equivalent	None

Reactor Coolant System Barrier	
Loss	**Potential Loss**
1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined	None
2. PC pressure > 1.7 psig due to RCS leakage	None
3. Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system): - Main steam line - HPCI steam line - RCIC steam line - RWCU - Feedwater 4. Emergency Depressurization is required	1. RCS leakage > 50 gpm inside the drywell 2. Unisolable primary system discharge outside primary containment as indicated by Secondary Containment area radiation or temperature above any Maximum Normal Operating Limit (0EOP-03-SCCP Tables 3, 1)
5. Drywell radiation > 27 R/hr with reactor shutdown	None

Containment Barrier	
Loss	**Potential Loss**
None	1. Primary Containment Flooding is required
1. PC pressure rise followed by a rapid unexplained drop in PC pressure 2. PC pressure response not consistent with LOCA conditions	2. PC pressure > 62 psig and rising 3. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ **AND** $O_2 \geq 5\%$) 4. Suppression pool water temperature and RPV pressure cannot be maintained below the HCTL
3. Failure of **any** valve in **any** one line to close **AND** Direct release pathway to the environment outside PC exists after PC isolation signal (manual or automatic) 4. Intentional PC venting per EOPs 5. Unisolable primary system discharge outside primary containment as indicated by Secondary Containment area radiation or temperature above any Maximum Safe Operating Limit (0EOP-03-SCCP Tables 3,1)	None
None	5. Drywell radiation > 20,000 R/hr

Barrier: Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

3. Failure of **any** valve in **any** one line to close

   AND

   Direct release pathway to the environment outside PC exists after PC isolation signal (manual or automatic)

BNP Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of primary containment integrity.

As stated above, the adjective "Direct" modifies "release pathway" to discriminate against release paths through interfacing liquid systems. Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable Main steam line, HPCI steam line or RCIC steam line breaks, unisolable RWCU system breaks, and unisloable containment atmosphere vent paths. If the main condenser is available with an unisolable main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using the Category R, Abnormal Rad Release / Rad Effluent, EALs.