Draft Submittal

BRUNSWICK OCT/NOV 2004

EXAM 50-325, 324/2004-301
OCTOBER 29, 2004 &
NOVEMBER 2 - 10, 2004

1. Senior Reactor Operator Written Exam

Brunswick Nuclear Plant

Initial Examination **DRAFT** SRO Written

Examination Report Nos.

05000325/2004301 - 05000324/204301

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Facility: Brunswick Date of Exam: NOV04 Exam Level: RO SRO

	Item Description				Initial a	Initial b*	Initial c#
1.	Questions and answers technically accurate and applicable to facility				CR	MAP	MB
2.	a. NRC K/As referenced for all questions b. Facility learning objectives referenced as available				CR	MAP	MB
3.	SRO questions are appropriate per Section D.2.d of ES-401				CR	MAP	MB
4.	Question selection and duplication from the last two NRC licensing exams appears consistent with a systematic sampling process						MB
5.	Question duplication from the license screening/audit exam was controlled as indicated below (check the item that applies) and appears appropriate: ✓ the audit exam was systematically and randomly developed; or ✓ the audit exam was completed before the license exam was started; or ✓ the examinations were developed independently; or ✓ the licensee certifies that there is no duplication; or __ other (explain)				CR	MAP	MB
6.	Bank use meets limits (no more than 75 percent from the bank at least 10 percent new, and the rest modified); enter the actual RO / SRO-only question distribution(s) at right	Bank 15 / 6	Modified 12 / 1	New 48 / 18	CR	MAP	MB
7.	Between 50 and 60 percent of the questions on the RO exam are written at the comprehension/analysis level; the SRO exam may exceed 60 percent if the randomly selected K/As support the higher cognitive levels; enter the actual RO / SRO question distribution(s) at right	Memory 32 / 8	C/A 43 / 17		CR	MAP	MB
8.	References/handouts provided do not give away answers				CR	MAP	MB
9.	Question content conforms with specific K/A statements in the previously approved examination outline and is appropriate for the Tier to which they are assigned; deviations are justified				CR	MAP	MB
10.	Question psychometric quality and format meet ES, Appendix B, guidelines				CR	MAP	MB
11.	The exam contains the required number of one-point, multiple choice items; the total is correct and agrees with value on cover sheet				CR	MAP	MB

	Printed Name / Signature	Date
a. Author	Curt Robert / Curt Robert	7/29/04
b. Facility Reviewer (*)	Marcus A. Pearson Jr. / [signature]	07/29/04
c. NRC Chief Examiner (#)	Richard S. Baldwin / [signature]	9/28/04
d. NRC Regional Supervisor	Michael E. Ernstes / [signature]	9/28/04

Note: * The facility reviewer's initials/signature are not applicable for NRC-developed examinations.
Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.

NRC 2004 SRO exam references to be provided:

1. 0OI-01.07, Notifications – Question 8 and 16, needed to allow operator to determine proper notification requirements.
2. Technical Specifications 3.3.2.1 and Unit 2 COLR – Question 10, needed to answer question since memorization of Tech Spec. operability determination is not required.
3. Technical Specifications 3.8.1 and 3.6.2.3 – Question 11, needed to answer question since memorization of Tech Spec. actions is not required.
4. Steam Tables – Question 17, not needed to answer question but applicant may want Steam Tables if an INCORRECT alternative is chosen.
5. Attachment 5 EOP User's Guide – Question 21, DWSIL and PSP limit curves needed to assess containment limits and then determine proper course of action based upon the limit exceeded.
6. Attachment 6 EOP User's Guide – Question 22, Needed to assess RPV level instrument operability.

NOTE: No references will aid applicant in answering other questions and none of the above questions result in being a direct lookup when reference is provided.

Reportability Evaluation Checklist

NOTE:	NUREG-1022, Rev. 2 should be referenced to provide additional guidance on reportability.
NOTE:	If the answer to any of the following questions is YES, then the event is reportable within one hour.
NOTE:	If all answers to the following questions are NO, the event is not reportable within one hour. Section 2.0 of this attachment contains the guidance for making four-hour reportability determinations.

1.0 ONE-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
1.1			Is the event a deviation from technical specifications as per 10CFR50.54(X)? [50.72(b)(1)]
1.2			Has any licensed material been lost, stolen, or missing in an aggregate quantity equal to or greater than 1,000 times the quantity specified in 10CFR20 Appendix C under such circumstances that it appears that an exposure could result to persons in unrestricted areas? [10CFR20.2201(a)(i)]
			NOTE: For further information related to this item, refer to SEC-NGGC-2147, Reporting of Safeguards and Fitness for Duty Events.
1.3			Does the event involve by-product, source or special nuclear material possessed by the licensee that might have or threatens to cause:
1.3.1			Any individual's exposure to reach or exceed 25 Rems total effective dose equivalent (TEDE); 75 Rems eye dose equivalent; or 250 Rads shallow-dose equivalent to the skin or extremities? [10CFR20.2202(a)(1)]
1.3.2			The release of radioactive material inside or outside of a restricted area, such that, had an individual been present for 24 hours, the individual could have received an intake 5 times the occupational annual limit on intake? [10CFR20.2202(a)(2)]
1.4			Has any safety/relief valve failed to close? (NUREG 0626 and NUREG 0660)

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, the event is reportable within four hours.
NOTE:	If all answers to the following questions are NO, then the event is not reportable within four hours. Section 3.0 of this attachment contains the guidance for making Eight-hour reportability determinations.

2.0 FOUR-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
2.1			Is plant shutdown required by technical specifications being initiated? [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? [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? [50.72(b)(2)(iv)(B)]
2.4			Is the event a situation, as related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made? [50.72(b)(2)(xi)]
			NOTE: Such an event may include an on-site fatality or an inadvertent release of radioactively contaminated materials.

Reportability Evaluation Checklist

NOTE: If the answer to any of the following questions is YES, the event is reportable within eight hours.

NOTE: If all the answers to the following questions are NO, the event is not reportable within eight hours. Section 4.0 of this attachment contains the guidance for making 24-hour reportability determinations.

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? [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? [50.72(b)(3)(ii)(B)]
3.3 3.3.1			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. [50.72(b)(3)(iv)(A)] These systems are: Reactor protection system (RPS) including: reactor scram and reactor trip. [50.72(b)(3)(iv)(B)(1)]
3.3.2			General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). • Main Steam Isolation. • Main Steam Line Drain Isolation. • HPCI Steam Line Isolation. • RCIC Steam Line Isolation. • RWCU Suction Isolation. • Primary Containment Isolation. • Secondary Containment Isolation. • SGTS Actuation. • Combustible Gas Control (CAD). [50.72(b)(3)(iv)(B)(2)]

Reportability Evaluation Checklist

ITEM #	YES	NO	DESCRIPTIVE QUESTION
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 [50.72(b)(3)(iv)(B)(4)]
3.3.4			Reactor Core Isolation Cooling (RCIC) [50.72(b)(3)(iv)(B)(5)]
3.3.5			Containment heat removal and depressurization systems including containment spray and fan cooler systems. • RHR Suppression Pool Cooling. • Drywell Spray System Actuation. [50.72(b)(3)(iv)(B)(7)]
3.3.6			Emergency Diesel Generators (DGs) [50.72(b)(3)(iv)(B)(8)]

Reportability Evaluation Checklist

ITEM #	YES	NO	DESCRIPTIVE QUESTION
3.4			Could the event or condition at the time of discovery have prevented the fulfillment of the safety function of structures or systems that are needed to: [50.72(b)(3)(v)] Events covered in this section may include: One or more procedural errors, Equipment failures, Discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported if redundant equipment in the same system was operable and available to perform the required safety function. [50.72(b)(3)(vi)]
3.4.1			Shut down the reactor and maintain it in a safe shutdown condition? [50.72(b)(3)(v)(A)]
3.4.2			Remove residual heat? [50.72(b)(3)(v)(B)]
3.4.3			Control the release of radioactive material? [50.72(b)(3)(v)(C)]
3.4.4			Mitigate the consequences of an accident? [50.72(b)(3)(v)(D)]
3.5			Does the event require the transport of a radioactively contaminated person to an off-site medical facility for treatment? [50.72(b)(3)(xii)]

Reportability Evaluation Checklist

NOTE:	Additional reportability guidance concerning loss of emergency preparedness capabilities is contained in NUREG-1022, Rev 2, Pages 75 through 79. It is also advisable to consult with an Emergency Preparedness representative when assessing the significance of the loss of capability.

ITEM #	YES	NO	DESCRIPTIVE QUESTION
3.6			Has the event resulted in a major loss of emergency assessment capability, off-site response capability, or communications capability (i.e., significant portion of the Main Control Room indication, emergency notification system, or off-site notification system)? [50.72(b)(3)(xiii)] Major loss of emergency or off-site notification system is considered to be/but not limited to: a. Loss of: 1) Selective signaling; OR NRC Emergency Notification System (ENS); AND 2) Commercial telephone network. b. Inoperability for ≥ one hour of: 1) Seven or more off-site sirens; OR 2) All off-site sirens in one county.

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, then the event is reportable within 24 hours.

4.0 24-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
4.1			Does the incident involve the loss of control of licensed material possessed by BNP which might have caused or threatens to cause:
4.1.1			Any individual's exposure in a period of 24 hours to exceed: 5 Rems total effective dose equivalent (TEDE); or 15 Rems eye dose equivalent; or 50 Rems shallow-dose equivalent to the skin or extremities? [10CFR20.2202(b)(1)]
4.1.2			The release of radioactive material inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake? [10CFR20.2202(b)(2)]

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1 Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two RBM channels inoperable.	B.1 Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 Suspend control rod movement except by scram. <u>OR</u>	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1.1	Verify ≥ 12 rods withdrawn.	Immediately
		OR	
	C.2.1.2	Verify by administrative methods that startup with RWM inoperable, for reasons other than bypassed control rod(s), has not been performed in the last calendar year.	Immediately
	AND		
	C.2.2	Verify movement of bypassed control rod(s) is in compliance with banked position withdrawal sequence (BPWS) by a second licensed operator or other qualified member of the technical staff.	During control rod movement
D. RWM inoperable during reactor shutdown.	D.1	Verify movement of bypassed control rod(s) is in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more Reactor Mode Switch—Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal. AND E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.
2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

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
SR 3.3.2.1.3	------------------------------NOTE------------------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 8.75% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	Verify the RBM: a. Low Power Range—Upscale Function OR Intermediate Power Range—Upscale Function OR High Power Range—Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ 29%. b. Intermediate Power Range—Upscale Function OR High Power Range—Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ Intermediate Power Range Setpoint specified in the COLR. c. High Power Range—Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ High Power Range Setpoint specified in the COLR.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 8.75% RTP.	24 months
SR 3.3.2.1.6	-----NOTE----- Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. ----- Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.2.1.7	-----NOTE----- Neutron detectors are excluded. ----- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range—Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
b. Intermediate Power Range—Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
c. High Power Range—Upscale	(c),(d)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
d. Inop	(d),(e)	2	SR 3.3.2.1.1	NA
e. Downscale	(d),(e)	2	SR 3.3.2.1.1 SR 3.3.2.1.7	NA
2. Rod Worth Minimizer	1[f],2[f]	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch—Shutdown Position	(g)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER is ≥ 29% RTP and MCPR less than the limit specified in the COLR except not required to be OPERABLE if the Intermediate Power Range—Upscale Function or High Power Range—Upscale Function is OPERABLE.

(b) THERMAL POWER is ≥ Intermediate Power Range Setpoint specified in the COLR and MCPR less than the limit specified in the COLR except not required to be OPERABLE if the High Power Range—Upscale Function is OPERABLE.

(c) THERMAL POWER ≥ High Power Range Setpoint specified in the COLR and < 90% RTP and MCPR less than the limit specified in the COLR.

(d) THERMAL POWER ≥ 90% RTP and MCPR less than the limit specified in the COLR.

(e) THERMAL POWER ≥ 29% and < 90% RTP and MCPR less than the limit specified in the COLR.

(f) With THERMAL POWER ≤ 8.75% RTP.

(g) Reactor mode switch in the shutdown position.

(h) Allowable Value specified in the COLR.

Table 2

RBM System Setpoints

Setpoint[a]	Trip Setpoint	Allowable Value
Lower Power Setpoint (LPSP[b])	27.7	≤ 29.0
Intermediate Power Setpoint (IPSP[b])	62.7	≤ 64.0
High Power Setpoint (HPSP[b])	82.7	≤ 84.0
Low Trip Setpoint (LTSP[c])	≤ 114.1	≤ 114.6
Intermediate Trip Setpoint (ITSP[c])	≤ 108.3	≤ 108.8
High Trip Setpoint (HTSP[c])	≤ 104.5	≤ 105.0
RBM Time Delay (t_{d2})	≤ 2.0 seconds	≤ 2.0 seconds

a RBM Operability requirements are not applicable:
(1) if MCPR ≥ 1.70; or
(2) if MCPR ≥ 1.45 and thermal power ≥ 90% Rated Thermal Power.

b Setpoints in percent of Rated Thermal Power.

c Setpoints relative to a full scale reading of 125.
For example, ≤ 114.1 means ≤ 114.1/125.0 of full scale.

This Table is referred to by Technical Specification 3.3.2.1 (Table 3.3.2.1-1).

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Four diesel generators (DGs); and

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

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ------------NOTE------------ Only applicable when Unit 1 is in MODE 4 or 5. ------------------------------ One Unit 1 offsite circuit inoperable.	A.1	Restore Unit 1 offsite circuit to OPERABLE status.	45 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ------------NOTES------------ 1. Only applicable when Unit 1 is in MODE 4 or 5. 2. Condition B shall not be entered in conjunction with Condition A. -------------------------------- Two Unit 1 offsite circuits inoperable due to one Unit 1 balance of plant circuit path to the downstream 4.16 kV emergency bus inoperable for planned maintenance. AND DG associated with the affected downstream 4.16 kV emergency bus inoperable for planned maintenance.	B.1	Declare required feature(s) with no power available inoperable when the redundant required feature(s) are inoperable.	Immediately from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.2	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	B.3	Restore both Unit 1 offsite circuits and DG to OPERABLE status.	7 days AND 10 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One offsite circuit inoperable for reasons other than Condition A or B.	C.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	C.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one 4.16 kV emergency bus concurrent with inoperability of redundant required feature(s)
	AND		
	C.3	Restore offsite circuit to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One DG inoperable for reasons other than Condition B.	D.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	D.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition D concurrent with inoperability of redundant required feature(s)
	AND		
	D.3.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	D.3.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		
	D.4	Restore DG to OPERABLE status.	7 days AND 10 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more offsite circuits inoperable for reasons other than Condition B.	E.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable. AND E.2 Restore all but one offsite circuit to OPERABLE status.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s) 24 hours
F. One offsite circuit inoperable for reasons other than Condition B. AND One DG inoperable for reasons other than Condition B.	-------------------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition F is entered with no AC power source to any 4.16 kV emergency bus. --------------------------------------- F.1 Restore offsite circuit to OPERABLE status. OR F.2 Restore DG to OPERABLE status.	 12 hours 12 hours
G. Two or more DGs inoperable.	G.1 Restore all but one DG to OPERABLE status.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. Required Action and associated Completion Time of Condition A, B, C, D, E, F or G not met.	H.1	Be in MODE 3.	12 hours
	AND		
	H.2	Be in MODE 4.	36 hours
I. One or more offsite circuits and two or more DGs inoperable. OR Two or more offsite circuits and one DG inoperable for reasons other than Condition B.	I.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

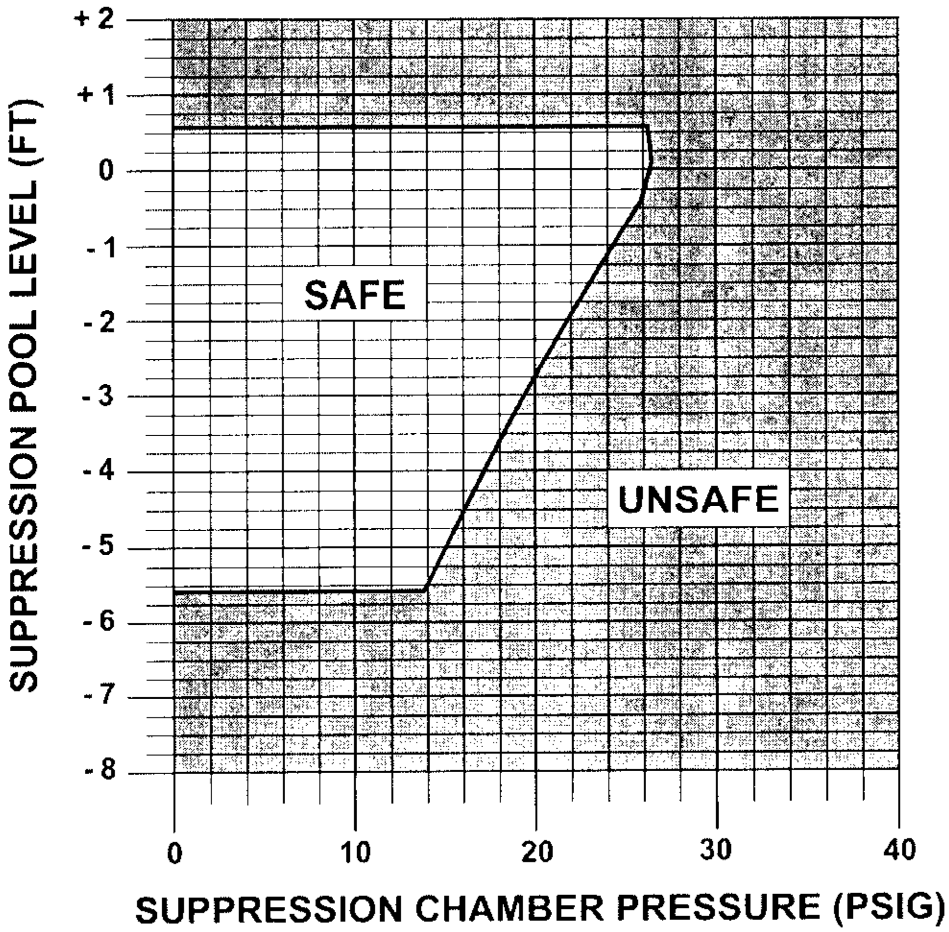
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ------------------------------ A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days
B. Two RHR suppression pool cooling subsystems inoperable.	B.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.3.2	Verify each RHR pump develops a flow rate $\geq$ 7700 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	92 days

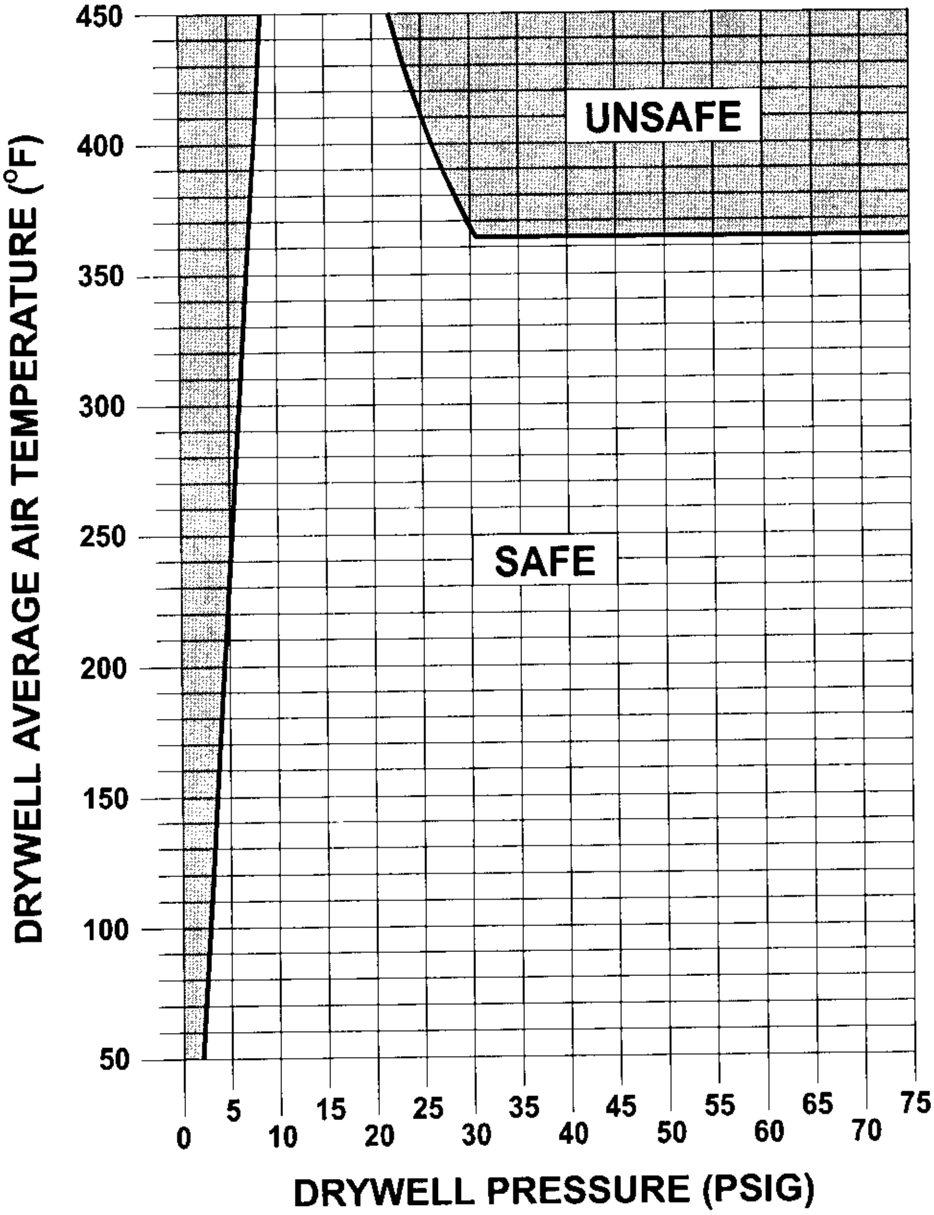
ATTACHMENT 5 (Cont'd)

FIGURE 7
PRESSURE SUPPRESSION PRESSURE



ATTACHMENT 5 (Cont'd)

FIGURE 1
DRYWELL SPRAY INITIATION LIMIT



NOTE

DRYWELL AVERAGE AIR TEMPERATURE MAY BE DETERMINED
USING ATTACHMENT 4 OF THE "USER'S GUIDE"

ATTACHMENT 6
REACTOR WATER LEVEL CAUTION
(Caution 1)

A reactor water level instrument may be used to determine reactor water level only when the conditions for use as listed in Table 1 are satisfied for that instrument.

TABLE 1

CONDITIONS FOR USE OF REACTOR WATER LEVEL INSTRUMENTS

NOTE

Reference leg area drywell temperature is determined using Figure 13, ERFIS, or Instructional Aid based on Figure 13.

NOTE

If the temperature near any instrument run is in the UNSAFE region of the REACTOR SATURATION LIMIT (Figure 14), the instrument may be unreliable due to boiling in the run.

NOTE

Immediate reference leg boiling is not expected to occur for short duration excursions into the unsafe region due to heating of the drywell. The thermal time constant associated with the mass of metal and water in the reference leg will prohibit immediate boiling of the reference leg. Reference leg boiling is an obvious phenomenon. Large scale oscillations of all water level instruments associated with the reference leg that is boiling will occur. This occurrence will be obvious and readily observable by the operator. Additionally, if the operator is not certain whether boiling has occurred, he can refer to plant history as provided on water level recorders or ERFIS. Reference leg boiling is indicated by level oscillations without corresponding pressure oscillations.

Instrument	Conditions for Use
Narrow Range Level Instruments C32-LI-R606A, B, C (N004A, B, C) C32-LPR-R608 (N004A, B) Indicating Range 150-210 Inches Cold Reference Leg	Unit 1 Only: The indicated level is in the SAFE region of Figure 15. Unit 2 Only: The indicated level is in the SAFE region of Figure 15A.
Shutdown Range Level Instruments B21-LI-R605A, B (N027A, B) Indicating Range 150-550 Inches Cold Reference Leg	The indicated level is in the SAFE region of Figure 16. NOTE To determine reactor water level at the Main Steam Line Flood Level (MSL), see Figure 21. NOTE Figure 21 has two curves: The upper curve is for reference leg area drywell temperature equal to or greater than 200°F. The lower curve is for reference leg area drywell temperature less than 200°F.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	1. Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, OR B21TA103) AND 2. IF the reference leg area drywell temperature is in the UNSAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 20 inches OR IF the reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 10 inches.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Fuel Zone Level Instruments B21-LI-R610 (N036) B21-LR-R615 (N037) Indicating Range -150 - +150 Inches Cold Reference Leg	1. IF the reference leg area drywell temperature is less than 440°F, THEN the indicated level is greater than -150 inches OR IF the reference leg area drywell temperature is greater than or equal to 440°F, THEN the indicated level is greater than -130 inches. AND 2. Reactor Recirculation Pumps are shutdown. NOTE To determine reactor water level at TAF, see Unit 1 Only: Figure 17 and Unit 2 Only: Figure 17A To determine reactor water level at the minimum steam cooling level (LL-4), see Unit 1 Only: Figure 18 and Unit 2 Only: Figure 18A To determine reactor water level at the minimum zero injection level (LL-5), see Unit 1 Only: Figure 19 and Unit 2 Only: Figure 19A To determine reactor water level at 90 inches, see Figure 20. Continued on next page.

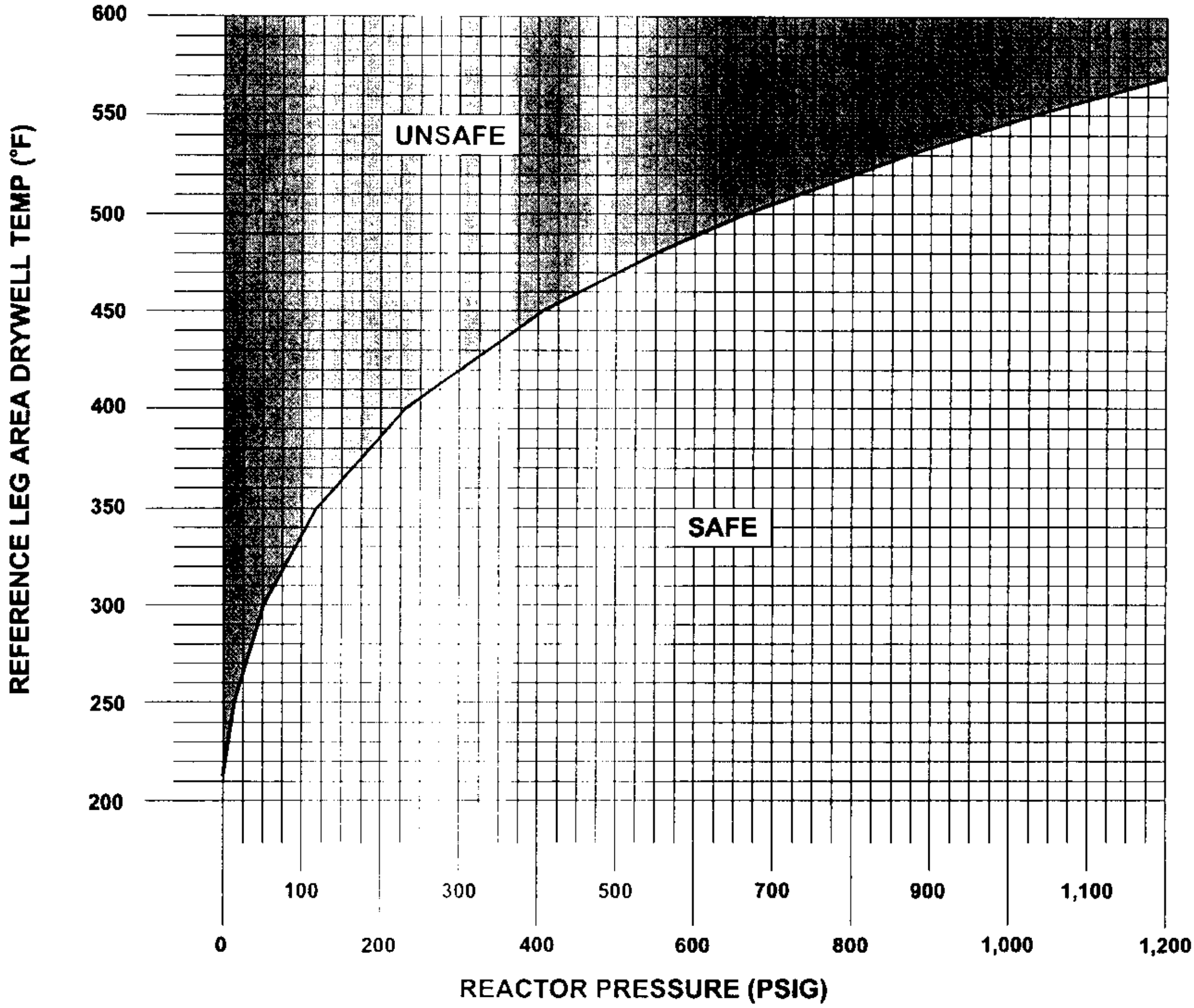
ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
	NOTE Each figure has two curves: The upper curve for reference leg area drywell temperature greater than 200°F. The lower curve for reference leg area drywell temperature less than or equal to 200°F. If containment conditions are such that reference leg area temperatures could not be controlled and maintained less than the 200°F requirement, then the upper lines on the graph should be utilized. NOTE These level instruments are valid for indication with RHR LPCI flow.

ATTACHMENT 6 (Cont'd)

FIGURE 14
REACTOR SATURATION LIMIT



BANK INFORMATION REPORT
for SRO NRC2004

Category 5 (Ref Req'd Y or N)	#Items	Title
NO	18	
Y ATT 5 PSPL	1	S295030EA201 SRO Q#21
Y CAUTION 1	1	S295031EA201 SRO Q#22
Y OI-01.07 ATT 1	2	S203000G2.4.30 SRO Q#8/S295018G2.4.30 SRO Q#1
Y STEAM TABLES	1	S295021AA203 SRO Q#17
Y TS 3.8.1 & 3.6.2.3	1	S219000A205 SRO Q#11
Y U2TS3.3.2.1& COLR	1	S215002A205 SRO Q#10

Category 7 (? Cognitive Level)	#Items	Title
C/A	17	
M OR FK	8	

Category 8 (? Source)	#Items	Title
BANK LOI	6	
MOD NRC	1	
NEW	18	

1. GSRO2.1.11 001

Which ONE of the following Reactivity Control Systems Technical Specification LCOs contain~~s~~ a $\boxed{\text{required action}}$ ~~statement~~ to manually scram the reactor immediately?
all caps

A. LCO 3.1.4 Control Rod Scram Times

B.✓ LCO 3.1.5 Control Rod Scram Accumulators

C. LCO 3.1.7 Standby Liquid Control (SLC) System

D. LCO 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

Feedback

REFERENCE - UNIT TWO TECH SPECS. AMMENDMENT NO. 233

DISTRACTOR ANALYSIS - Distractors chosen so that three out of four contain the word scram so as to not lead the applicant. Kept distractors in Reactivity Control Systems so that all are related. Question worded to maintain it Generic in that specific details of LCO condition are not required to answer the question.

A. INCORRECT - MODE 3 IN 12 HOURS
B. CORRECT - Condition D low charging hdr press w/ inop accumulators
C. INCORRECT - MODE 3 IN 12 HOURS
D. INCORRECT - MODE 3 IN 12 HOURS

Notes

2.1.11 Knowledge of less than one hour technical specification action statements for systems.
(CFR: 43.2 / 45.13)
IMPORTANCE RO 3.0 SRO 3.8

55.43 (2) Facility operating limitations in the technical specifications and their bases.

This question matches the k/a since it measures the SRO's knowledge of a less than one hour tech. Spec. action statement for control rod scram accumulators.

Categories

Tier:	TIER 3	Group:	SRO
Importance Rating:	SRO 3.8	Facility Objective:	CLS-LP-008*018
Ref Req'd Y or N:	NO	Technical Ref.:	TS
? Cognitive Level:	M OR FK	? Source:	NEW

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.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days

(continued)

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 $\geq$ 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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 950 psig.	B.1	Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND		
	B.2.1	------------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."	1 hour
	OR		
	B.2.2	Declare the associated control rod inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 950 psig.	C.1	Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	AND		
	C.2	Declare the associated control rod inoperable.	1 hour
D. Required Action B.1 or C.1 and associated Completion Time not met.	D.1	------------NOTE------------ Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. -------------------------------- Manually scram the reactor.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is ≥ 940 psig.	7 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SLC subsystem inoperable.	A.1	Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1	Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SDV vent and drain line.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SDV vent or drain lines with one valve inoperable.	A.1	Restore valve to OPERABLE status.	7 days
B. One or more SDV vent or drain lines with both valves inoperable.	B.1	-------------NOTE------------- An isolated line may be unisolated under administrative control to allow draining and venting of the SDV. ----------------------------------	
		Isolate the associated line.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

2. GSRO2.1.7 001

Unit One (1) has been operating at rated power for several weeks. While reviewing the CO DSR the SCO notes that feedwater temperature has decreased slowly over the past several days. Further investigation reveals that feedwater temperature is now 12°F below design.

Which ONE of the following identifies the ~~correct operational judgement~~ determination ~~action~~ that the SCO should make in response to the feedwater temperature reduction?

The SCO should require:

A.✓ use of a Power-Flow map with a larger region of instability identified because the probability of ~~a stability~~ an instability event has been increased.

B. that a Manual Heat Balance calculation be performed because the Heat Balance calculation provided by the Process Computer is no longer valid.

C. that more restrictive MCPR limits be invoked because the probability of exceeding the MCPR Safety Limit has been increased.

D. that more restrictive MCPR limits be invoked because the Turbine control valve fast closure, and stop valve trips will be bypassed at a higher (non-conservative) power.

Feedback

REFERENCE - CO DSR 0OI-03.1 Rev. 78 and ENP-24 Rev. 25 Page 19

DISTRACTOR ANALYSIS

A. CORRECT - Recent enhancement made to plant procedures to provide guidance in cases that involve a loss of feedwater heating.
B. INCORRECT - Heat Balance uses feedwater temperature (FWT) as an input thus the calculation should compensate for the lower feedwater temperature. Plausible however as alternate power verification program asks whether FWT is normal or reduced.
C. INCORRECT - CMFLCPR will increase due to the shift in flux towards the bottom of the core. However, no requirement to invoke more restrictive MCPR limits exists unless Turb. Bypass valve system is inoperable with FWT reduction. (0OI-01.08 Selected Equipt. OOS)
D. INCORRECT - This is the bases for a maximum of 84°F reduced FWT. Not a problem as long as rx. power is above 40%. (GP-13 Rev. 19 page 7)

Notes

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.
(CFR: 43.5 / 45.12 / 45.13)
IMPORTANCE RO 3.7 SRO 4.4

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to make an operational judgement by evaluating feedwater temperature outside of design limits.

Categories

Tier:	TIER 3	Group:	SRO
Importance Rating:	SRO 4.4	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	ENP-24
? Cognitive Level:	C/A	? Source:	NEW

2. Determine how long the rod was mispositioned.

3. Determine magnitude of mispositioned control rod.

4. Determine needed power reduction for recovery. (This is dependent on core location of control rod, fraction inserted, time mispositioned, and core conditions.) This power reduction can be significant (up to 40%).

5. Determine rate of control rod movement (single notch, continuous, etc.).

6. Determine what other control rod movements may be needed to support repositioning.

7. Recommend power maneuvers to accomplish recovery.

8. Monitor thermal limits.

8.5.5 Loss of Feedwater Heating

CAUTION

Continued operation with less than the full compliment of feedwater heaters requires a current feedwater heater out of service analysis.

CAUTION

Loss of feedwater heating increases the probability of an instability event when operating near a stability region. There should be increased surveillance of the indicators of instability at this time. See Section 8.8.

CAUTION

Any required thermal limit changes resulting from feedwater temperature reduction must be completed within 4 hours per Plant Technical Specification. Use guidance provided in 0ENP-26, the Core Operating Limits Report (COLR), 1(2)OP-32, and related procedure for applicability.

CAUTION

Reduction of feedwater heating in excess of 10 degrees of design (unless otherwise specified by cycle specific analysis) requires use of the applicable Feedwater Temperature Reduction (FWTR) Power-Flow map. Design feedwater temperature for a given reactor power level can be found in 1(2)OP-32.

3. GSRO2.2.17 001

Which ONE of the following identifies who is responsible for reviewing IST pump and valve performance data within 96 hours of data collection per 0OI-01.02, Shift Routines and Operating Practices?

A. Shift Superintendent (SS)

B. Shift Technical Adviser (STA)

C.✓ Unit Senior Control Operator (SCO)

D. Work Control Center Senior Reactor Operator (WCC SRO)

Feedback

REFERENCE - 0OI-01.02 Rev. 34 page 31

DISTRACTOR ANALYSIS- All alternatives are homogenous and simply asking to recall a specific fact regarding time therefore all are plausible.

A INCORRECT - SS reviews PTs but 96 hour time requirement to review IST data is specifically the responsibility of the Unit SCO
B - INCORRECT - STAs may assist in reviewing PTs but have no signature authority.
C - CORRECT - PTs containing IST pump and valve performance data shall be reviewed by the Unit SCO within 96 hours of data collection (as required by ENP-16.1, IST Pump and Valve data).
D - INCORRECT - WCC SRO is responsible for controlling authorization of work activities and not specifcally responsible for reviewing completed work.

Notes

2.2.17 Knowledge of the process for managing maintenance activities during power operations.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.3 SRO 3.5

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's knowledge of administrative requirements of managing maintenance activities at power.

Categories

Tier:	TIER 3	Group:	SRO
Importance Rating:	SRO 3.5	Facility Objective:	CLS-LP-201-E*12E
Ref Req'd Y or N:	NO	Technical Ref.:	0OI-01.02
? Cognitive Level:	M OR FK	? Source:	NEW

5.15 Control Operator Administrative Duties

5.15.2 Administrative duties assigned to the Control Operators should **NOT** compromise the operators' shift responsibilities, and shall **NOT** interfere with the ability to monitor their panels.

5.15.3 When one Control Operator is involved in administrative tasks, another Control Operator should assume responsibility for monitoring the unit, as discussed in Section 5.14.

5.16 Operations Control of Maintenance

5.16.1 The guidelines and recommendations of ADM-NGGC-0101, Maintenance Rule Program, shall be considered when taking equipment out of service.

1. Unavailability time of High Safety Significant systems shall be minimized.
2. Functional failures which would render any Maintenance Rule structure, system, train, or component (SSC) unavailable should be identified.

5.16.2 The Integrated Weekly Schedule and POD are developed to provide a thoroughly researched interface between plant maintenance and operational considerations and requirements.

5.16.3 The WCC SRO has responsibility for controlling authorization of work activities, and shall ensure that current/changing plant conditions can support work prior to authorization.

5.16.4 Reviews of Periodic Tests (PTs) containing IST pump and valve performance data shall be reviewed by the Unit SCO within 96 hours of data collection (as required by 0ENP-16.1, IST Pump and Valve Data).

1. Documentation of this review will be satisfied when the entry is made in the Unit SCO Log for satisfactory or unsatisfactory performance of the periodic test.

4. GSRO2.3.3 001

You are the Shift Superintendent. There has been an inadvertent activation of the off-site Alert and Notification (Siren) System. The 30 sirens in Brunswick County are not in alarm, the five sirens within New Hanover County are alarming.

Which ONE of the following describes the responsibilities of control room personnel in accordance with 0AI-113, Response to Inadvertent Siren Activation?

Control room personnel must ensure that all appropriate notifications to county, state, and federal agencies are made and that:

A.✓ Security personnel are dispatched to disable the continuously sounding sirens.

B. Operations personnel are dispatched to disable the continuously sounding sirens.

C. Wilmington Area Transmission personnel are dispatched to disable the continuously sounding sirens.

D. New Hanover County Emergency Operations Center personnel are dispatched to disable the continuously sounding sirens.

Feedback

Randomly Selected from BANK BSEP 2001 NRC Exam Q# GEN 2.3.3 BSEP
Original Question
You are the Shift Superintendent. There has been an inadvertent activation of the off-site Alert and Notification (Siren) System. The 30 sirens in Brunswick County are not in alarm, the five sirens within New Hanover County are alarming.

Which ONE of the following describes the responsibilities of control room personnel in accordance with 0AI-113, Response to Inadvertent Siren Activation?

A. Dispatch security personnel to the continuously sounding sirens and disable the sirens. Ensure all appropriate notifications to county, state, and federal agencies are made.
B. Dispatch operations personnel to the continuously sounding sirens and disable the sirens. Ensure all appropriate notifications to county, state, and federal agencies are made.
C. Notify New Hanover County Emergency Operations Center. Instruct them to dispatch personnel to the continuously sounding sirens and disable the sirens.
D. Notify New Hanover County Emergency Operations Center. Instruct them to dispatch personnel to the continuously sounding sirens and disable the sirens. Ensure all appropriate notifications to county, state, and federal agencies are made.

Original question modified slightly to correct psychometric flaw of distractor C being a subset of D. Additionally, new distractor C is more plausible in that Wilmington Area Transmission perform maintenance on sirens in New Hanover County but are not designated by procedure to disable continuously sounding sirens.

REFERENCE - 0AI-113 Rev. 3 Page 2 of 7

DISTRACTOR ANALYSIS
A. correct answer
B. Security Personnel disable the alarms.
C. Wilmington Area Transmission maintains the sirens but not designated by AI-113 to disable malfunctioning sirens.
D. CP&L maintains the sirens

Notes

2.3.3 Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g. / waste disposal and handling systems).
(CFR: 43.4 / 45.10)
IMPORTANCE RO 1.8 SRO 2.9

55.43 (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

This question matches the k/a since it measures the SRO's knowledge of administrative responsibilities regarding handling E-plan equipment outside the control room.

Categories

Tier:	TIER 3	Group:	SRO
Importance Rating:	SRO 2.9	Facility Objective:	CLS-LP-201-C021
Ref Req'd Y or N:	NO	Technical Ref.:	AI-113
? Cognitive Level:	M OR FK	? Source:	MOD NRC

1.0 PURPOSE

This procedure is intended to be used as a guideline for the Brunswick Nuclear Plant's response to an inadvertent activation of the off-site Alert and Notification (Siren) System. The system includes 31 sirens in Brunswick County and five sirens within New Hanover County. Although CP&L maintains the sirens, they are not normally activated by the plant. Activation is normally done from the county Emergency Operation Centers.

The population within the 10-mile EPZ has been instructed through a continuing community education program to tune their radios or televisions to the local Emergency Alert System (EAS) station following the sounding of the sirens. Information will be provided to the public concerning necessary actions.

2.0 REFERENCES

2.1 NRC Information Notice No. 90-34

2.2 0PEP-02.6.6, Environmental Monitoring Team Leader

2.3 EPL-001, Emergency Phone List

2.4 0OI-01.07, Notifications

3.0 RESPONSIBILITIES

3.1 Control Room

3.1.1 Coordinate response to inadvertent siren activation.

3.1.2 Ensure all appropriate notifications to county, state, and federal agencies are made.

3.2 Security

3.2.1 Dispatch personnel to continuously sounding siren(s) and disable the siren(s).

5. GSRO2.3.9 001 The entry is)

Unit ~~Two (2) is at 60% power performing~~ a scheduled shutdown to make a drywell entry. to investigate an increase in Drywell Floor Drain leakage. The Shift Superintendent requests that Primary Containment Purging (Deinerting) commence as soon as possible to minimize plant down time. E&RC reports that the drywell atmosphere is NOT suitable for unfiltered release.

Which ONE of the following identifies the earliest time that Primary Containment Purging (Deinerting) may commence under these conditions?

The Primary Containment Purging (Deinerting) flowpath can be aligned through the SBGT:

A. once the reactor is in MODE 3.

B.✓ once the reactor is in MODE 4.

C. once reactor power is less than 15%.

D. immediately as long as reactor power will be at less than 15% within 24 hours.

Feedback

REFERENCE - 2OP-24, Precaution 3.6 page 6, Section 5.3 page 31 and Section 8.13 page 124.

DISTRACTOR ANALYSIS
A, C, and D INCORRECT - 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.
B. CORRECT

Notes

2.3.9 Knowledge of the process for performing a containment purge.
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.5 SRO 3.4

55.43 (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

This question matches the k/a since it measures the SRO's knowledge of containment purge requirements.

Categories

Tier:	TIER 3	Group:	SRO
Importance Rating:	SRO 3.4	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	OP-24
? Cognitive Level:	C/A	? Source:	NEW

3.0 PRECAUTIONS AND LIMITATIONS

3.3 Purge or purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the containment.

3.4 Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is **NOT** provided or required. Vent used in system names does **NOT** imply a venting process.

3.5 *CAC V160, V161, V162*, and *V163*, will **NOT** reopen automatically after a Group 6 isolation **OR** loss of solenoid power. To open these valves, the solenoid power has to be on and the individual solenoid power *RESET* push buttons (XU-51) depressed.

3.6 Purging the drywell/torus through the SBGT System with the large purge exhaust valves *(CAC-V7, V8, V10)* shall **NOT** be performed with the plant above Mode 4. Otherwise, the SBGT System could be damaged if a LOCA occurred while purging.

3.7 On receipt of Group 6 isolation, E&RC should be contacted to secure the Remote Radiation Monitor Sample Station Remote Pump, if running. This will prevent blowing the power supply fuses to the remote sample pump.

3.8 Failure to notify E&RC Chemistry at least 24 hours in advance when primary containment purge samples are required may delay sample results.

R27 3.9 In Modes 1, 2, or 3, simultaneous inerting, de-inerting or ventilating the drywell and suppression chamber is prohibited.

3.10 Primary Containment should be considered inoperable while *SUPPRESSION POOL LEVEL SIGHTGLASS CAC-LG-4336* is valved in service.

4.0 PREREQUISITES

4.1 Instrument and Service Air System is in operation in accordance with 0OP-46.

4.2 Containment Atmosphere Control System Electrical Lineup complete in accordance with Attachment 1.

4.3 The Containment Atmosphere Control System Panel Lineup complete in accordance with Attachment 2.

5.3 Primary Containment Purging (Deinerting) Through the Purge Exhaust Fans

C
Continuous
Use

5.3.1 Initial Conditions

1. The reactor will be at less than 15% power within 24 hours. ☐

2. Personnel entry into primary containment is necessary. ☐

3. Reactor Building Ventilation System is in operation in accordance with 2OP-37.1. ☐

4. The Main Stack Radiation Monitor is in service in accordance with 2OP-11 **AND** *CAC PURGE VENT ISOL OVRD, CAC-CS-5519*, is in *OFF* on Panel XU-51. ☐

5. At least one of the primary containment H2/02 analyzers is in service in accordance with Section 5.5 or 5.6. ☐

5.3.2 Procedural Steps

1. **NOTIFY** E&RC to sample drywell prior to purging. ☐

NOTE: Purging the drywell/suppression chamber should be performed within 8 hours of the time E&RC reports noble gas sample results to the Control Room, provided atmospheric conditions within the drywell are **NOT** changing.

2. **IF** E&RC indicates the drywell atmosphere is **NOT** suitable for unfiltered release, **THEN GO TO** Section 8.13 to deinert containment. ☐

8.13 Primary Containment Purging (Deinerting) Through the SBGT System

C
Continuous
Use

8.13.1 Initial Conditions

1. All applicable prerequisites as listed in Section 4.0 are met. ☐
2. Personnel entry into primary containment is necessary. ☐
3. The Reactor Building Ventilation System is in operation in accordance with 2OP-37.1. ☐
4. Unit is in Mode 4. ☐
5. E&RC has determined by sampling that drywell atmosphere is **NOT** suitable for unfiltered release. ☐
6. At least one of the primary containment H2/O2 analyzers is in service in accordance with Section 5.5 or 5.6. ☐

8.13.2 Procedural Steps

1. **IF** desired, **THEN NOTIFY** E&RC to sample drywell prior to purging. ☐

NOTE: Purging the drywell should be performed within 8 hours of the time E&RC reports noble gas sample results to the Control Room, provided atmospheric conditions within the drywell are **NOT** changing.

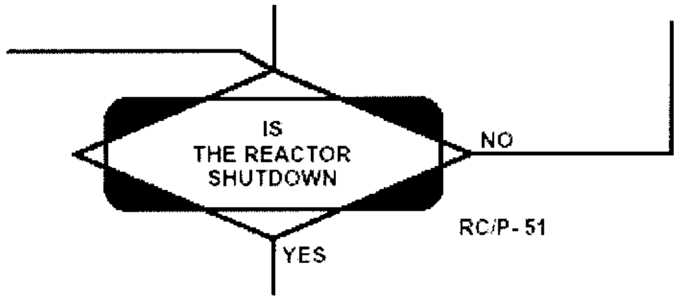
CAUTION

Purging the drywell/torus through the SBGT System, with the large purge exhaust valves (*CAC-V7, V8, V10*), shall **NOT** be performed with the plant above Mode 4.

2. **WHEN** authorization is received from E&RC that drywell atmosphere is suitable for filtered release, **THEN CLOSE** *SBGT 2A REACTOR BUILDING SUCTION VALVE, 2D-BFV-RB*. ☐

6. GSRO2.4.21 001

Following an incomplete reactor scram on Unit Two (2), the operating crew is executing 2EOP-01-LPC, Level/Power Control and the SCO has reached the following decision step:



Which ONE of the following conditions results in the SCO answering this question YES per 0OI-37.5, Level/Power Control Procedure Basis Document?

A.✓ The Reactor is subcritical on range 5 of IRMs.

B. All OPERABLE APRM Downscale lights are lit.

C. The entire SLC tank has been injected to the Reactor.

D. Hot Shutdown Boron Weight has been injected to the Reactor.

If "D" is correct then "C" is also correct.

Feedback

Randomly selected Bank Question CLS-LP-300B*008 001- Question similar to question in EOP bank however graphic has been used here instead of text description making question appear different. Also, added per OI-37.5 per operation's validation recommendation.

REFERENCE - 0OI-37.5 Rev. 6 pages 73 and 74

DISTRACTOR ANALYSIS
A. CORRECT - subcritical below heating range
B. INCORRECT - downscales are not an indication of being below heating range and no mention is made of criticality
C and D INCORRECT - SLC tank level or boron concentration is not considered in this assessment.

Suggest revising distractor "D" to reference a certain amount of rods being stuck out.

Notes

2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions including:

1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.

(CFR: 43.5 / 45.12)
IMPORTANCE RO 3.7 SRO 4.3

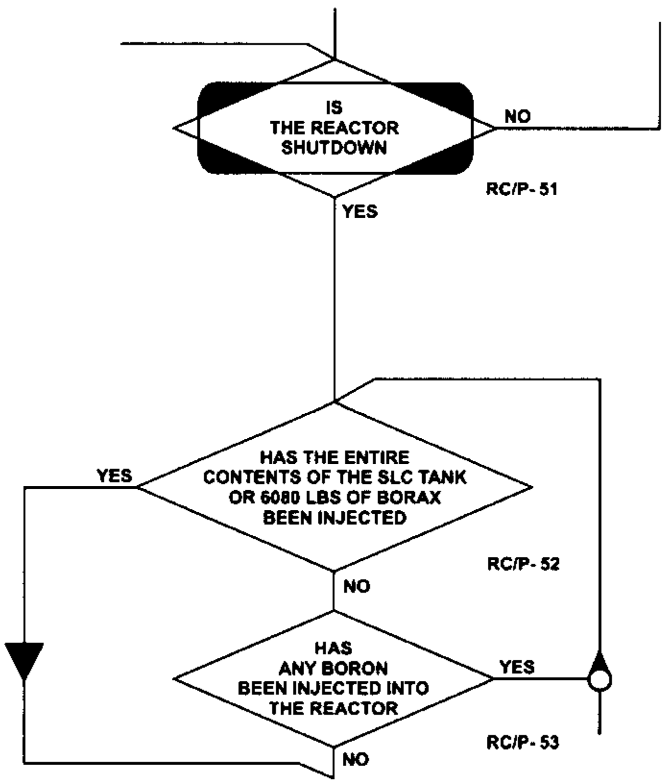
55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's knowledge of logic used to assess reactivity control during EOPs.

Categories

Tier:	TIER 3	Group:	SRO
Importance Rating:	SR0 4.3	Facility Objective:	CLS-LP-300-E*14E
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.5
? Cognitive Level:	C/A	? Source:	BANK LOI

STEPS RC/P-51 through RC/P-53



STEP BASES:

The Cold Shutdown Boron Weight (CSBW) is not a quantity which can be measured by the operator. An SLC tank level of 0% or 6080 pounds of borax injected are equivalent to the CSBW. The Cold Shutdown Boron Weight is defined to be the least weight of soluble boron which, if injected into the reactor vessel and uniformly mixed, will maintain the reactor shut down under all conditions. This weight is utilized to assure the reactor will remain shut down irrespective of control rod position or reactor temperature.

The Cold Shutdown Boron Weight is calculated as approximately 126 pounds of 47% enriched boron in calculation 0EOP-WS-01. The actual Cold Shutdown Boron Weight is not used in the procedure steps. These steps use a level in the SLC tank and a weight of borax as an equivalent for Cold Shutdown Boron Weight. These values are determined in calculation 0EOP-WS-15. It has been decided to use 0% to represent the Cold Shutdown Boron Weight in the procedure. This value can be read by the operator on the indication in the Control Room. The borax concentration for Cold Shutdown Boron Weight used in the procedure is 6080 pounds.

STEPS RC/P-51 through RC/P-53 (continued)

Continued reactor depressurization and cooldown to cold shutdown conditions may not proceed until the conditions listed in Steps RC/P-51 through RC/P-53 are satisfied.

Step RC/P-51 is used to direct the proper actions. If the reactor is not shutdown, then the pressure control actions remain in place. If the reactor is shutdown, then the subsequent steps can be used to determine if the reactor cooldown can proceed. Shutdown as applied to the reactor is defined as subcritical with reactor power below the heating range.

If no boron has been injected into the reactor vessel, depressurization and cooldown to cold shutdown may proceed as long as control rod insertion is sufficient to shut down the reactor. Such action is permitted even though the existing margin to criticality is small. A return to criticality under these conditions is acceptable because termination of the cooldown will stop the reactor power increase.

If any amount of boron less than the CSBW has been injected into the reactor vessel, cooldown to cold shutdown is not permitted unless it can be determined that control rod insertion alone ensures the reactor will remain shut down under all conditions. The core reactivity response from cooldown in a partially borated core is unpredictable and subsequent steps may not prescribe the correct actions for such conditions if criticality were to occur.

7. GSRO2.4.28 001

Unit Two (2) is operating at rated power when an AO reports that he has discovered apparent tampering of the Normal/Local keylock switch at the Remote Shutdown Panel (RSDP).

Which ONE of the following identifies an immediate action that must be performed by Operations in response to this event per 0SI-27, Security Event Response Guidelines, if a specific credible insider threat is determined to exist?

A. The Unit SCO shall direct insertion of a manual scram.

B. The Unit SCO shall ~~assess EALs and make the required notifications to off-site agencies.~~ require Engineering to perform an evaluation of the known plant condition upon the safe operation of the plant.

C. The Shift Superintendent shall implement the Two-Person Rule for personnel in vital areas.

D. The Shift Superintendent shall coordinate security inspections of the entire power block.

Feedback

REFERENCE - 0SI-27 Rev. 5 page 6

DISTRACTOR ANALYSIS

A. INCORRECT - AOP-40 requires that a manual scram be inserted if an attack on the plant or loss of plant control is imminent or an intruder appears to be armed and attempting destruction of vital areas. SI-27 provides no specific scram instructions
B. INCORRECT - This is an IA of SI-27 but notifications are the responsibilty of the Ops and Security Shift Superintendents not the SCO.
C. CORRECT
D. INCORRECT - This is a responsibilty of security and is also not an immediate action.

Notes

2.4.28 Knowledge of procedures relating to emergency response to sabotage.
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 2.3 SRO 3.3

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's knowledge of security instructions used during a sabotage event.

Categories

Tier:	TIER 3	Group:	SRO
Importance Rating:	SRO 3.3	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	SI-27
? Cognitive Level:	M OR FK	? Source:	NEW

3.0 DEFINITIONS

3.4 Tampering - With intent and potentially/does have minor effect on safe operations.

3.5 Sabotage - With intent and potentially/does have significant effect on safe operations.

3.6 Two-Person Rule - Line of sight contact by at least two persons when accessing vital areas upon existence of a specific credible insider threat.

4.0 RESPONSIBILITIES

4.1 The Operations Shift Superintendent or the Site Emergency Coordinator (SEC) or designee is responsible for notifying the NRC of reportable safeguard events. Notification will be accomplished in accordance with the established reporting procedure.

4.2 The Shift Security Supervisor is responsible for determining Reportability under 10CFR73.71 and 0SEC-NGGC-2147. This decision will be communicated directly with the on duty Superintendent - Operations or Supervisor - Operations.

4.3 The Operations Shift Superintendent, Operations Manager or Plant Manager are responsible for implementation of the Two Person Rule upon the existence of a credible insider threat specific to the Brunswick Nuclear Plant.

4.4 The senior member of the security force on duty is responsible for coordination of the following: (until relieved by the Superintendent - Security/designee).

4.4.1 Initial gathering of facts, data and evidence.

4.4.2 Notifying the on-call Security staff member and the Superintendent-Security.

4.4.3 Managing the investigation into the event.

4.4.4 Ensuring adequate Security response and compensatory measures including coordination of security inspections.

4.4.5 Notifying the Operations Shift Superintendent of actions taken.

4.4.6 Operational and functional assessment of security equipment.

4.4.7 Interface with Corporate Security.

4.4.8 Interface with Local Law Enforcement Agency (LLEA).

5.0 INSTRUCTIONS

Tampering or sabotage is considered an extremely serious event. It must be assumed that the individual who performed this act had sufficient plant access to impede the performance of additional plant equipment even though there is no evidence that this has taken place. The directions in this procedure are the basis for providing a high degree of assurance that the event has been limited to identified components and the plant equipment necessary for safe operation will perform as designed.

R2.9

Upon determining the existence of a credible insider threat implements the two-person rule. Implementation shall be as expeditious as resources permit recognizing that additional personnel may need to be called to the site.

All media contacts will be coordinated through site or corporate communications. Press releases or communication with outside agencies and organizations should be anticipated. Prompt notification of Site Communications personnel allows for early involvement. Caution should be exercised in the use of descriptions such as "sabotage", "bomb", "terrorism" or similar language unless this language precisely reflects the known facts.

5.1 Records and Notifications

Licensing/Regulatory Programs and/or Security shall be responsible for maintaining documentation, to include 30-day follow-up reports and the Safeguards Event Logs, for events under this procedure, as required by 10CFR73.71. Such documentation shall be maintained for three (3) years.

5.2 Immediate Actions

5.2.1 Immediate actions include the operational response appropriate to the event such as placing the equipment, component, system, or unit in a safe operating condition. **(Responsibility: Operations Shift Superintendent)**

5.2.2 Prompt notifications shall be made in accordance with site procedures and policies. **(Responsibility: Security Shift Supervisor/Operations Shift Superintendent)**

5.2.3 Implement the Two-Person Rule for personnel in vital areas upon existence of a specific credible insider threat. **(Responsibility: Operations Shift Superintendent)**

1. Security alarm station personnel or supervision will notify plant personnel within the protected area that the Two Person Rule has been implemented. This can be accomplished by use of the public address system, briefing personnel upon entry into the protected area or briefing personnel prior to entry into a vital area.

8. S203000G2.4.30 001

Unit Two (2) is operating at rated power with LPCI A INOPERABLE ~~and~~ when the following sequence of events occur~~s~~:

0000 7 day completion time for LCO 3.5.1, ECCS Operating, Condition A expires and Condition C is entered requiring that the Unit be placed in MODE 3 in 12 hours.
0010 Management decides to work other minor maintenance items during the shutdown.
0030 Plant shutdown is commenced per LCO 3.5.1, Condition C.
0050 LPCI A is repaired and declared OPERABLE; LCO 3.5.1 Conditions A and C are exited.
0100 Management decides to continue the plant shutdown as planned to complete the other maintenance items.
0230 Unit Two (2) in MODE 3

Which ONE of the following correctly identifies the Emergency Notification System (ENS) report requirements for ~~this event?~~ these plant conditions?

An ENS report:

A. is not required.

B.✓ should be submitted no later than 0430.

C. should be submitted no later than 0630.

D. should have been submitted no later than 0100.

Feedback

REFERENCE - 0OI-01.07 page 19 **PROVIDE ATT. 1 of OI-.01.07 as reference**

DISTRACTOR ANALYSIS
A. INCORRECT - Shutdown initiated as required by TS at 0030
B. CORRECT - per 10CFR50.72 (b)(2)(i). 4 hours from when Shutdown is initiated
C. INCORRECT- Corresponds to 4 hours after shutdown completed
D. INCORRECT- Corresponds to 1 hour after LCO 3.5.1 A expires. Candidate may interpret this as a TS deviation and select the 1 hour reportable per 10CFR50.72 (b)(1)

Notes

SYSTEM: 203000 RHR/LPCI: Injection Mode (Plant Specific)

2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies.

(CFR: 43.5 / 45.11)
IMPORTANCE RO 2.2 SRO 3.6

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's knowledge of reportability requirements.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	SRO 3.6	Facility Objective:	CLS-LP-200-D*019
Ref Req'd Y or N:	Y OI-01.07 ATT 1	Technical Ref.:	0OI-01.07
? Cognitive Level:	C/A	? Source:	NEW

Reportability Evaluation Checklist

NOTE:	NUREG-1022, Rev. 2 should be referenced to provide additional guidance on reportability.
NOTE:	If the answer to any of the following questions is YES, then the event is reportable within one hour.
NOTE:	If all answers to the following questions are NO, the event is not reportable within one hour. Section 2.0 of this attachment contains the guidance for making four-hour reportability determinations.

1.0 ONE-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
1.1			Is the event a deviation from technical specifications as per 10CFR50.54(X)? [50.72(b)(1)]
1.2			Has any licensed material been lost, stolen, or missing in an aggregate quantity equal to or greater than 1,000 times the quantity specified in 10CFR20 Appendix C under such circumstances that it appears that an exposure could result to persons in unrestricted areas? [10CFR20.2201(a)(i)]
			NOTE: For further information related to this item, refer to SEC-NGGC-2147, Reporting of Safeguards and Fitness for Duty Events.
1.3			Does the event involve by-product, source or special nuclear material possessed by the licensee that might have or threatens to cause:
1.3.1			Any individual's exposure to reach or exceed 25 Rems total effective dose equivalent (TEDE); 75 Rems eye dose equivalent; or 250 Rads shallow-dose equivalent to the skin or extremities? [10CFR20.2202(a)(1)]
1.3.2			The release of radioactive material inside or outside of a restricted area, such that, had an individual been present for 24 hours, the individual could have received an intake 5 times the occupational annual limit on intake? [10CFR20.2202(a)(2)]
1.4			Has any safety/relief valve failed to close? (NUREG 0626 and NUREG 0660)

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, the event is reportable within four hours.
NOTE:	If all answers to the following questions are NO, then the event is not reportable within four hours. Section 3.0 of this attachment contains the guidance for making Eight-hour reportability determinations.

2.0 FOUR-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
2.1			Is plant shutdown required by technical specifications being initiated? [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? [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? [50.72(b)(2)(iv)(B)]
2.4			Is the event a situation, as related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made? [50.72(b)(2)(xi)]
			NOTE: Such an event may include an on-site fatality or an inadvertent release of radioactively contaminated materials.

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, the event is reportable within eight hours.
NOTE:	If all the answers to the following questions are NO, the event is not reportable within eight hours. Section 4.0 of this attachment contains the guidance for making 24-hour reportability determinations.

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? [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? [50.72(b)(3)(ii)(B)]
3.3 3.3.1			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. [50.72(b)(3)(iv)(A)] These systems are: Reactor protection system (RPS) including: reactor scram and reactor trip. [50.72(b)(3)(iv)(B)(1)]
3.3.2			General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). • Main Steam Isolation. • Main Steam Line Drain Isolation. • HPCI Steam Line Isolation. • RCIC Steam Line Isolation. • RWCU Suction Isolation. • Primary Containment Isolation. • Secondary Containment Isolation. • SGTS Actuation. • Combustible Gas Control (CAD). [50.72(b)(3)(iv)(B)(2)]

Reportability Evaluation Checklist

ITEM #	YES	NO	DESCRIPTIVE QUESTION
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 [50.72(b)(3)(iv)(B)(4)]
3.3.4			Reactor Core Isolation Cooling (RCIC) [50.72(b)(3)(iv)(B)(5)]
3.3.5			Containment heat removal and depressurization systems including containment spray and fan cooler systems. • RHR Suppression Pool Cooling. • Drywell Spray System Actuation. [50.72(b)(3)(iv)(B)(7)]
3.3.6			Emergency Diesel Generators (DGs) [50.72(b)(3)(iv)(B)(8)]

Reportability Evaluation Checklist

ITEM #	YES	NO	DESCRIPTIVE QUESTION
3.4			Could the event or condition at the time of discovery have prevented the fulfillment of the safety function of structures or systems that are needed to: [50.72(b)(3)(v)] Events covered in this section may include: One or more procedural errors, Equipment failures, Discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported if redundant equipment in the same system was operable and available to perform the required safety function. [50.72(b)(3)(vi)]
3.4.1			Shut down the reactor and maintain it in a safe shutdown condition? [50.72(b)(3)(v)(A)]
3.4.2			Remove residual heat? [50.72(b)(3)(v)(B)]
3.4.3			Control the release of radioactive material? [50.72(b)(3)(v)(C)]
3.4.4			Mitigate the consequences of an accident? [50.72(b)(3)(v)(D)]
3.5			Does the event require the transport of a radioactively contaminated person to an off-site medical facility for treatment? [50.72(b)(3)(xii)]

Reportability Evaluation Checklist

NOTE:	Additional reportability guidance concerning loss of emergency preparedness capabilities is contained in NUREG-1022, Rev 2, Pages 75 through 79. It is also advisable to consult with an Emergency Preparedness representative when assessing the significance of the loss of capability.

ITEM #	YES	NO	DESCRIPTIVE QUESTION
3.6			Has the event resulted in a major loss of emergency assessment capability, off-site response capability, or communications capability (i.e., significant portion of the Main Control Room indication, emergency notification system, or off-site notification system)? [50.72(b)(3)(xiii)] Major loss of emergency or off-site notification system is considered to be/but not limited to: a. Loss of: 1) Selective signaling; OR NRC Emergency Notification System (ENS); AND 2) Commercial telephone network. b. Inoperability for ≥ one hour of: 1) Seven or more off-site sirens; OR 2) All off-site sirens in one county.

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, then the event is reportable within 24 hours.

4.0 24-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
4.1			Does the incident involve the loss of control of licensed material possessed by BNP which might have caused or threatens to cause:
4.1.1			Any individual's exposure in a period of 24 hours to exceed: 5 Rems total effective dose equivalent (TEDE); or 15 Rems eye dose equivalent; or 50 Rems shallow-dose equivalent to the skin or extremities? [10CFR20.2202(b)(1)]
4.1.2			The release of radioactive material inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake? [10CFR20.2202(b)(2)]

9. S209001G2.1.33 001

Unit Two (2) is operating at rated power. While performing 0PT-07.2.4A, Core Spray Loop A Operability, Core Spray Room Cooler A fails to start when Core Spray Pump A is started. The Reactor Building AO reports that the Room Cooler breaker has tripped on thermal overload.

Which ONE of the following identifies the action that is required by the SCO in response to the tripped Core Spray Room Cooler A breaker?

The SCO should:

A.✓ immediately declare Core Spray Subsystem A Inoperable.

B. contact Engineering to perform an operability determination.

C. ensure that Core Spray Room Cooler B is Operable and continue the test.

D. direct the AO to attempt one reset of the tripped breaker and continue the test.

Feedback

REFERENCE 0OI-01.08 Rev. 56 page 16

DISTRACTOR ANALYSIS
A. CORRECT - Per OI-01.08 an inop ECCS room cooler requires the associated ECCS be declared inoperable.
B. INCORRECT - 0OI-01.08 already clarifies OPERABILITY determination.
C. INCORRECT - Core Spray room coolers are not considered redundant like RHR room coolers as Core Spray room cooler can only cool its respective room.
D. INCORRECT - Per AP-13 a tripped breaker cannot be reset until an investigation has been performed, except in case of an emergency.

Notes

SYSTEM: 209001 Low Pressure Core Spray System
2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.
(CFR: 43.2 / 43.3 / 45.3)
IMPORTANCE RO 3.4 SRO 4.0

55.43 (2) Facility operating limitations in the technical specifications and their bases.
55.43 (3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

This question matches the k/a since it measures the SRO's knowledge of Core Spray LCO requirements.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	SRO 4.0	Facility Objective:	CLS-LP-018*017
Ref Req'd Y or N:	NO	Technical Ref.:	0OI-01.08
? Cognitive Level:	C/A	? Source:	NEW

5.1.2 System/Component Related Guidance

3. Emergency Bus Power Supplies

If a normal or emergency power supply to E1, E2, E3, or E4 is being removed from service, then perform the following:

a. Reference TRM Appendix F, Safety Function Determination Program (SFDP), Attachments 1 and 2 to assist with determination of Technical Specification 3.8.1 and 3.8.7 requirements and to assess the possible impact on supported systems.

b. If an evaluation of the SFDP is performed, then document the evaluation in the LCOS computer program or on Attachment 15, Safety Function Determination Program Evaluation (SFDP), if LCOS is not available.

R2

4. ECCS Room Coolers

NOTE:	The following step is not required to be performed if the ECCS Room Cooler is inoperable due to the loss of a 4160V or 480V E-Bus. E-Bus inoperability impacts the operability of ECCS subsystems. Technical Specifications and the SFDP will provide Required Actions to be taken for the loss of the E-Bus.

When any ECCS Room Cooler is determined to be inoperable, then the ECCS equipment associated with that room cooler is to be declared inoperable per the applicable Technical Specifications.

EXAMPLE:	The RHR Room Coolers are to be considered redundant components required to support the operation of RHR. Therefore, should a room cooler be found or made inoperable, a seven (7) day Active LCO is required to be established on the RHR system. Likewise, should both room coolers be found inoperable, the action required is the same as if both RHR loops and HPCI were inoperable. Should it be identified that one RHR Room Cooler is inoperable and one RHR Loop is also inoperable (specific combinations do not matter), the action is as if only one RHR Loop is inoperable (7 days).

10. S215002A205 001

Unit Two (2) is operating at 38% power performing a control rod pattern adjustment. While preparing to move a control rod the CO notices that the bar graph for Rod Block Monitor (RBM) A indicates three upward pointing triangles at 104.5% while the bar graph for RBM B has one upward pointing triangle at ~~114.1%~~. 114.5%

Which ONE of the following predicts the impact that these indications have on the Rod Block Monitor System and identifies the correct Technical Specification operability determination corresponding to these indications?

RBM A is enforcing the:

A. Low Trip Setpoint (LTSP) and RBM B is enforcing the High Trip Setpoint (HTSP). Per Technical Specifications both RBMs may be considered OPERABLE.

B.✓ High Trip Setpoint (HTSP) and RBM B is enforcing the Low Trip Setpoint (LTSP). Per Technical Specifications both RBMs may be considered OPERABLE.

C. Low Trip Setpoint (LTSP) and RBM B is enforcing the High Trip Setpoint (HTSP). Per Technical Specifications RBM A must be declared INOPERABLE while RBM B may be considered OPERABLE.

D. High Trip Setpoint (HTSP) and RBM B is enforcing the Low Trip Setpoint (LTSP). Per Technical Specifications RBM B must be declared INOPERABLE while RBM A may be considered OPERABLE.

Feedback

REFERENCE - Unit 2 Tech. Spec. 3.3.2.1 and COLR should be provided as a reference.

DISTRACTOR ANALYSIS

A. INCORRECT - RBM A is enforcing HTSP (3 up triangles) and RBM B is enforcing LTSP (one up triangle)

B. CORRECT - RBM A is not enforcing the correct setpoint but may be considered OPERABLE as Table 3.3.2.1-1 of TS footnotes (a) and (b) state that LTSP and ITSP are not required to be OPERABLE if the Upscale Function is OPERABLE. RBM B is enforcing the correct LTSP for 38%.

C. INCORRECT - RBM A is enforcing HTSP (3 up triangles) and RBM B is enforcing LTSP (one up triangle). RBM B can be considered OPERABLE since it is enforcing the correct setpoint LTSP for 38% power.

D. INCORRECT - RBM B can be considered OPERABLE since it is enforcing the ~~HTSP~~ LTSP.

Notes

SYSTEM: 215002 Rod Block Monitor System

A2. Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM ; 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) No 43 reference

A2.05 RBM high ~~or~~ inoperable: BWR-3,4,5 . . . . 3.2 3.3

This question matches the k/a since it measures the SRO's ability to diagnose the impact of abnormal RBM upscale indications and the ability to use Tech Specs. to determine operability impact.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	SRO 3.2	Facility Objective:	CLS-LP-09.6*026
Ref Req'd Y or N:	Y U2TS3.3.2.1& COLR	Technical Ref.:	TS 3.3.2.1 AND COLR
? Cognitive Level:	C/A	? Source:	NEW

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1 Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR Two RBM channels inoperable.	B.1 Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 Suspend control rod movement except by scram. OR	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1.1	Verify ≥ 12 rods withdrawn.	Immediately
		OR	
	C.2.1.2	Verify by administrative methods that startup with RWM inoperable, for reasons other than bypassed control rod(s), has not been performed in the last calendar year.	Immediately
	AND		
	C.2.2	Verify movement of bypassed control rod(s) is in compliance with banked position withdrawal sequence (BPWS) by a second licensed operator or other qualified member of the technical staff.	During control rod movement
D. RWM inoperable during reactor shutdown.	D.1	Verify movement of bypassed control rod(s) is in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more Reactor Mode Switch—Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal.	Immediately
	AND	
	E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.
2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

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
SR 3.3.2.1.3	-----NOTE----- Not required to be performed until 1 hour after THERMAL POWER is ≤ 8.75% RTP in MODE 1. ----- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	Verify the RBM: a. Low Power Range—Upscale Function OR Intermediate Power Range—Upscale Function OR High Power Range—Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ 29%. b. Intermediate Power Range—Upscale Function OR High Power Range—Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ Intermediate Power Range Setpoint specified in the COLR. c. High Power Range—Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ High Power Range Setpoint specified in the COLR.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 8.75% RTP.	24 months
SR 3.3.2.1.6	------------------------------NOTE------------------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. --- Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.2.1.7	------------------------------NOTE------------------------------ Neutron detectors are excluded. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range—Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
b. Intermediate Power Range—Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
c. High Power Range—Upscale	(c),(d)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
d. Inop	(d),(e)	2	SR 3.3.2.1.1	NA
e. Downscale	(d),(e)	2	SR 3.3.2.1.1 SR 3.3.2.1.7	NA
2. Rod Worth Minimizer	1[(f)],2[(f)]	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch—Shutdown Position	(g)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER is ≥ 29% RTP and MCPR less than the limit specified in the COLR except not required to be OPERABLE if the Intermediate Power Range—Upscale Function or High Power Range—Upscale Function is OPERABLE.

(b) THERMAL POWER is ≥ Intermediate Power Range Setpoint specified in the COLR and MCPR less than the limit specified in the COLR except not required to be OPERABLE if the High Power Range—Upscale Function is OPERABLE.

(c) THERMAL POWER ≥ High Power Range Setpoint specified in the COLR and < 90% RTP and MCPR less than the limit specified in the COLR.

(d) THERMAL POWER ≥ 90% RTP and MCPR less than the limit specified in the COLR.

(e) THERMAL POWER ≥ 29% and < 90% RTP and MCPR less than the limit specified in the COLR.

(f) With THERMAL POWER ≤ 8.75% RTP.

(g) Reactor mode switch in the shutdown position.

(h) Allowable Value specified in the COLR.

Table 2

RBM System Setpoints

Setpoint[a]	Trip Setpoint	Allowable Value
Lower Power Setpoint (LPSP[b])	27.7	≤ 29.0
Intermediate Power Setpoint (IPSP[b])	62.7	≤ 64.0
High Power Setpoint (HPSP[b])	82.7	≤ 84.0
Low Trip Setpoint (LTSP[c])	≤ 114.1	≤ 114.6
Intermediate Trip Setpoint (ITSP[c])	≤ 108.3	≤ 108.8
High Trip Setpoint (HTSP[c])	≤ 104.5	≤ 105.0
RBM Time Delay (t_{d2})	≤ 2.0 seconds	≤ 2.0 seconds

[a] RBM Operability requirements are not applicable:
(1) if MCPR ≥ 1.70; or
(2) if MCPR ≥ 1.45 and thermal power ≥ 90% Rated Thermal Power.

[b] Setpoints in percent of Rated Thermal Power.

[c] Setpoints relative to a full scale reading of 125.
For example, ≤ 114.1 means ≤ 114.1/125.0 of full scale.

This Table is referred to by Technical Specification 3.3.2.1 (Table 3.3.2.1-1).

11. S219000A205 001

Unit Two (2) is operating at rated power. The following sequence of events occurs:

@0200 RHR Suppression Pool Cooling Loop 2A is declared Inoperable due to Suppression Pool Cooling Isolation valve E11-F024A being closed with the stem and disk separated. (i.e. can not open valve)

@0600 Diesel Generator 4 is declared Inoperable

Which ONE of the following identifies when Unit Two (2) must enter a Tech Spec shutdown statement? (Assume that no repairs are made)

A. at 1000.

B. at 1400.

C✓ at 1800.

D. at 0600 the next day.

Feedback

REFERENCE - TS 3.8.1 D and 3.6.2.3 should be provided

In order to answer this question the applicant must be able to determine that the DG failure (electrical failure) will require cascading and declaring RHR B INOP resulting in a loss of safety function for RHR suppression pool cooling. Based upon this determination the correct action is in LCO 3.6.2.3 and it is to enter condition B. If applicant does not recognize that the loss of safety function is covered by the LCO he/she may select that a 3.0.3 entry is required after 4 hours.

DISTRACTOR ANALYSIS

A. INCORRECT - LCO 3.8.1 Action D.2 requires declaring required supported features by the DG inoperable when the redundant required features are inoperable (to prevent loss of safety function in event of a LOOP) four hours from discovery of concurrent inoperability. This includes RHR Suppression pool cooling. Four hours from discovery of would be at 1000. At this time, RHR 2B would have to be declared inop, since this would be loss of safety function for RHR suppression pool cooling. LCO 3.6.2.3 Condition B would be required at 1000. 3.6.2.3 C requires entry into a shutdown statement after 8 hour completion time for Condition B.

B. INCORRECT - LCO 3.6.2.3 has a completion time of 8 hours but this time would not start until after 4 hours of 3.8.1 D.

C. CORRECT - 4 hours from discovery 3.8.1 D plus 8 hours for 3.6.2.3 B

D. INCORRECT - Distractor based upon an additional 12 hour required to be in MODE 3. However question states when must Unit 2 enter into a shutdown statement not the completion time.

Notes

SYSTEM: 219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode

A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL ~~COOLING 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) no 43 reference

A2.05 A.C. electrical failures 3.3 3.5

This question matches the k/a since it measures the SRO's ability to determine the impact that AC electrical failures has on Suppression Pool Cooling and to use Tech Specs. to mitigate the consequences of the loss of power.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	SRO 3.5	Facility Objective:	CLS-LP-017*022
Ref Req'd Y or N:	Y TS 3.8.1 & 3.6.2.3	Technical Ref.:	TS 3.8.1 & 3.6.2.3
? Cognitive Level:	C/A	? Source:	NEW

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	---NOTE--- LCO 3.0.4 is not applicable. ---		
	A.1	Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days
B. Two RHR suppression pool cooling subsystems inoperable.	B.1	Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.3.2	Verify each RHR pump develops a flow rate ≥ 7700 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	92 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Four diesel generators (DGs); and

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

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE-------------- Only applicable when Unit 1 is in MODE 4 or 5. ---------------------------------- One Unit 1 offsite circuit inoperable.	A.1 Restore Unit 1 offsite circuit to OPERABLE status.	45 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ------------NOTES------------ 1. Only applicable when Unit 1 is in MODE 4 or 5. 2. Condition B shall not be entered in conjunction with Condition A. -------------------------------- Two Unit 1 offsite circuits inoperable due to one Unit 1 balance of plant circuit path to the downstream 4.16 kV emergency bus inoperable for planned maintenance. <u>AND</u> DG associated with the affected downstream 4.16 kV emergency bus inoperable for planned maintenance.	B.1 Declare required feature(s) with no power available inoperable when the redundant required feature(s) are inoperable. <u>AND</u> B.2 Perform SR 3.8.1.1 for OPERABLE offsite circuit(s). <u>AND</u> B.3 Restore both Unit 1 offsite circuits and DG to OPERABLE status.	Immediately from discovery of Condition B concurrent with inoperability of redundant required feature(s) 2 hours <u>AND</u> Once per 12 hours thereafter 7 days <u>AND</u> 10 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One offsite circuit inoperable for reasons other than Condition A or B.	C.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	C.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one 4.16 kV emergency bus concurrent with inoperability of redundant required feature(s)
	AND		
	C.3	Restore offsite circuit to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One DG inoperable for reasons other than Condition B.	D.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	D.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition D concurrent with inoperability of redundant required feature(s)
	AND		
	D.3.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	D.3.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		
	D.4	Restore DG to OPERABLE status.	7 days AND 10 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two or more offsite circuits inoperable for reasons other than Condition B.	E.1	Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s)
	AND		
	E.2	Restore all but one offsite circuit to OPERABLE status.	24 hours
F. One offsite circuit inoperable for reasons other than Condition B. AND One DG inoperable for reasons other than Condition B.	---NOTE--- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition F is entered with no AC power source to any 4.16 kV emergency bus. ---		
	F.1	Restore offsite circuit to OPERABLE status.	12 hours
	OR		
	F.2	Restore DG to OPERABLE status.	12 hours
G. Two or more DGs inoperable.	G.1	Restore all but one DG to OPERABLE status.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Required Action and associated Completion Time of Condition A, B, C, D, E, F or G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. One or more offsite circuits and two or more DGs inoperable. OR Two or more offsite circuits and one DG inoperable for reasons other than Condition B.	I.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days

(continued)

12. S262001G2.2.25 001

~~At Brunswick each Class 1E AC distribution system redundant load group has access to two offsite power supplies (one preferred and one alternate).~~ Teaching in stem.

← Bold

Which ONE of the following correctly identifies the preferred and alternate sources for emergency buses E1 and E2 as described in Technical Specifications?

The preferred source of power to emergency buses E1 and E2 is provided by the:

bold

A. Unit 1 UAT aligned to the main generator and the alternate source of power is provided by the Unit 1 SAT.

B.✓ Unit 1 SAT and the alternate source of power is provided by the Unit 1 UAT aligned in backfeed mode.

C. Unit 1 UAT aligned in backfeed mode and the alternate source of power is provided by the Unit 1 SAT.

D. Unit 1 SAT and the alternate source of power is provided by the Unit 1 UAT aligned to the main generator.

Feedback

REFERENCE - Unit One (1) Technical Specifications Bases B3.8.1-1&2 Rev. 31

DISTRACTOR ANALYSIS
A. INCORRECT - This describes the Normal Full power lineup. Per TS the preferred source of power to emergency buses E1 and E2 is provided by the Unit 1 SAT and the alternate source of power is provided by the Unit 1 UAT aligned in backfeed mode.
B. CORRECT
C. INCORRECT - Sources are reversed.
D. INCORRECT - Normal Full Power lineup reversed.

Comment - Ops validation comment suggested that this question was tricky due to wording (preferred and alternate) and inconsistent with AP-22. AP-22 specifically addresses when EDG or E-buses are scheduled to be unavailable, consideration may be given to aligning power to a UAT backfeed. This will facilitate a quick automatic transfer to the SAT if power were interrupted to the UAT. This question is only intended to test the knowledge of the electrical distribution LCO and does not address specific case described in AP-22. Discussed this with validator and he concurred the question was fine as written.

Notes

SYSTEM: 262001 A.C. Electrical Distribution

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
(CFR: 43.2)
IMPORTANCE RO 2.5 SRO 3.7

55.43 (2) Facility operating limitations in the technical specifications and their bases.

This question matches the k/a since it measures the SRO's knowledge of preferred and alternate sources listed in Tech Specs. for the Class 1E AC distribution system.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	SRO 3.7	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	TS 3.8.1 BASES
? Cognitive Level:	M OR FK	? Source:	NEW

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources—Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred and alternate power sources), and the onsite standby power sources (diesel generators (DGs) 1, 2, 3, and 4. Per the UFSAR (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The Class 1E AC distribution system is divided into redundant load groups, so loss of any one group does not prevent the minimum safety functions from being performed. Each load group has access to two offsite power supplies (one preferred and one alternate) via a balance of plant (BOP) circuit path. This BOP circuit path consists of the BOP bus and the associated circuit path (master/slave breakers and interconnecting cables) to a 4.16 kV emergency bus. Each load group can also be connected to a single DG.

Offsite power is supplied to the 230 kV switchyards from the transmission network by eight transmission lines. From the 230 kV switchyards, two qualified electrically and physically separated circuits provide AC power, through either a startup auxiliary transformer (SAT) or backfeeding via a unit auxiliary transformer (UAT), to 4.16 kV BOP buses. A single circuit path (master/slave breakers and interconnecting cables) from each BOP bus provides offsite power to its associated downstream 4.16 kV emergency bus. A detailed description of the offsite power network and circuits to the onsite Class 1E emergency buses is found in the UFSAR, Sections 8.2 and 8.3 (Ref. 2).

A qualified offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from either 230 kV bus (bus A or B) to the onsite Class 1E emergency buses.

The Unit 1 main generator provides the normal source of power to 4.16 kV emergency buses E1 and E2 via its respective UAT. The Unit 2 main generator provides the normal source of power to 4.16 kV emergency buses E3 and E4 via its respective UAT. In the event of a

(continued)

BASES

BACKGROUND (continued)

unit trip, an automatic transfer from the normal circuit (main generator output via the UAT) to the respective unit SAT occurs resulting in the SAT supplying power to two 4.16 kV emergency buses. As such, the Unit 1 SAT provides the preferred source of power to emergency buses E1 and E2 and the Unit 1 UAT (backfeed mode) is the alternate source of power to emergency buses E1 and E2. The Unit 2 SAT provides the preferred source of power to emergency buses E3 and E4 and the Unit 2 UAT (backfeed mode) is the alternate source of power to emergency buses E3 and E4. Each UAT can only be considered a qualified offsite source if it is capable of being powered from the 230 kV switchyard (Ref. 3).

The SATs and UATs are sized to accommodate the load sequence starting of all emergency loads on receipt of an accident signal.

The onsite standby power source for 4.16 kV emergency buses E1, E2, E3, and E4 consists of four DGs. Each DG is dedicated to its associated emergency bus. A DG starts automatically on a loss of coolant accident (LOCA) signal from either Unit 1 or Unit 2 or on an emergency bus degraded voltage or undervoltage signal (refer to LCO 3.3.8.1, "Loss of Power (LOP) Instrumentation"). After the DG has started, it automatically ties to its respective bus after offsite power is tripped as a consequence of emergency bus undervoltage or degraded voltage, independent of or coincident with a LOCA signal. The DGs also start and operate in the standby mode without tying to the emergency bus on a LOCA signal alone. Following the trip of offsite power, all loads are stripped from the emergency bus except the 480 V emergency bus. When the DG is tied to the emergency bus, select safety related loads are then sequentially connected to their respective emergency bus by individual timers associated with each auto-connected load following a permissive from a voltage relay monitoring each emergency bus.

In the event of a loss of preferred power, the emergency electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a LOCA.

(continued)

13. S264000G2.4.22 001

Following a dual unit loss of off-site power, electrical distribution status is:

DG1 tripped
DG2 tied to ~~E2~~ its associated emergency bus
DG3 under clearance
DG4 tripped

Which ONE of the following identifies why AOP-36.2 establishes cross-tie operation to energize all 480 VAC E Buses within one hour as the SCO's highest priority?

To restore power to:

A. critical RPV level and pressure instrumentation to prevent an unnecessary entry into the Reactor Flooding procedure.

B. drywell coolers to prevent Primary Containment temperature from exceeding the design limit.

C. drywell coolers to prevent an unnecessary LOCA signal (high drywell pressure coincident with low reactor pressure).

D. battery chargers to extend the battery capacity to the complete four hour coping duration required by analysis.

Feedback

REFERENCE - 0AOP-36.2 Rev. 25 page 148
Top priority is to cross-tie within 1 hour to maintain DC power for SBO coping loads on blacked out unit.

DISTRACTOR ANALYSIS
A. INCORRECT - Critical instruments with DC power are listed in Attachments of AOP-36.2. Cross-tie is not performed for this reason.
B. INCORRECT - Cooldown at greater than 100F/hr initiated to ensure PC temp below design.
C. INCORRECT - Nonblacked out unit maintained >500 psig to prevent LOCA signal.
D. CORRECT- Top priority is to cross-tie within 1 hour to maintain DC power for SBO coping loads on blacked out unit.

Notes

SYSTEM: 264000 Emergency Generators (Diesel/Jet)
2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.
(CFR: 43.5 / 45.12)
IMPORTANCE RO 3.0 SRO 4.0

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's knowledge of the bases of prioritizing station blackout procedure actions.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	SRO 4.0	Facility Objective:	CLS-LP-303-A*06C
Ref Req'd Y or N:	NO	Technical Ref.:	AOP-36.2
? Cognitive Level:	M OR FK	? Source:	NEW

4.0 GENERAL DISCUSSION

This procedure partially fulfills the requirements of Regulatory Guide 1.155, Station Blackout, and enforces the criteria set forth in NUMARC 87-00 Rev 1, and the Brunswick Nuclear Project Station Blackout Coping Analysis (SBOCA) Report. The term Station Blackout refers to the complete loss of AC power to the essential (emergency) and nonessential (BOP) switchgear busses in a nuclear power plant. Station Blackout therefore involves a Loss of Off-Site Power concurrent with a turbine trip, and the failure of the On-Site Diesel Generator Emergency Power System, but not the loss of available AC power to busses fed by 125 VDC batteries through inverters, or the loss of power from alternate AC power sources.

The initiating event for a Station Blackout is assumed to be a Loss of Off-Site Power (LOOP) at the plant site resulting from a switchyard related event due to random faults, or an external event, such as a grid disturbance, or a weather event that affects the Off-Site Power System either throughout the grid or at the plant. LOOPs caused by fire, flood, or seismic activity are not expected to occur with sufficient frequency to require explicit criteria, and are not considered in this procedure.

For BNP, with its normally dedicated emergency AC power sources, a Station Blackout is assumed to occur on only one unit. The non Blacked Out unit is capable of sharing a single Diesel Generator with the Blacked Out unit, after assuming a single failure on the non Blacked Out unit. BNP is required to cope for four hours after a Station Blackout.

During a Station Blackout, the most significant requirement is to quickly restore AC power to the 125 VDC battery chargers. The 125 VDC batteries have adequate capacity for only 1 to 2 hours of operation without DC load stripping during a Station Blackout. To extend the battery capacity to the complete four hour coping duration, it is necessary to cross-tie emergency busses with the non Blacked Out unit. If only one Diesel Generator is available, the cross-ties would include one 4160V Emergency Bus cross-tie and two 480V Substation cross-ties. In accordance with the requirements set forth in the SBOCA Report, these cross-ties must be completed within the first hour of the start of the Station Blackout.

14. S295007AA203 001

The Emergency Operating Procedures have been entered due to a Group One (1) isolation resulting in a high reactor pressure condition. ~~The Safety Relief Valves (SRVs) will only operate on lift pressure due to a loss of solenoid power and~~ The condenser is not available as a heat sink. HPCI is inoperable due to a failed auxiliary oil pump breaker with repairs ~~are~~ in progress.

As SCO consider the options available to stabilize reactor pressure per EOP-01-RVCP, Reactor Vessel Control Procedure.

Which ONE of the following pressure control methods will result in the least amount of reactor vessel inventory loss?

A.✓ RWCU (Recirc. mode).

B. RWCU (Blowdown mode).

C. RCIC in pressure control mode.

D. ~~Allow SRVs to cycle on lift setpoint~~. Cycle SRV's as necessary to stabilize pressure.

NO power

Feedback

REFERENCE 0OI-37.4 Rev. 4 pages 67-70

SRO Only: Mitigating actions during an emergency.10CFR55.43(b)(5)

DISTRACTOR ANALYSIS
A. CORRECT - Does not remove any coolant inventory.
B. INCORRECT - Inventory lost to radwaste or condenser.
C. INCORRECT - Does not draw a significant amount of steam but it is more than RWCU recirc..
D. INCORRECT - Significant loss of inventory.

Notes

APE: 295007 High Reactor Pressure
AA2. Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE :
(CFR: 41.10 / 43.5 / 45.13)
AA2.03 Reactor water level.................................... 3.7 3.7

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to interpret which reactor pressure control method will have the least effect on RPV level when operating in a high RPV pressure condition.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	SRO 3.7	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.4
? Cognitive Level:	C/A	? Source:	NEW

STEP RC/P-38

STABILIZE REACTOR PRESS BELOW 1050 PSIG WITH ONE OR MORE OF THE FOLLOWING SYSTEMS:

* MAIN TURBINE BYPASS VALVES
* SRVs ONLY WHEN SUPPRESSION POOL WATER LEVEL IS ABOVE -8 FEET.
  USE OPENING SEQUENCE.
  RESTORE CONTINUOUS PNEUMATIC SUPPLY IF NECESSARY PER **TABLE 2**.
* HPCI WITH SUCTION FROM CST
* RCIC WITH SUCTION FROM CST
* RWCU (RECIRC MODE) BYPASS FILTER DEMINS PER EOP-01-SEP-07
  DEFEAT SLC AND LL-2 ISOL INTERLOCKS IF NECESSARY PER "CIRCUIT ALTERATION PROCEDURE" (EOP-01-SEP-10)
* MAIN STEAM LINE DRAINS
* RWCU (BLOWDOWN MODE) ONLY IF NO BORON HAS BEEN INJECTED (OP-14)
  NOTIFY E&RC TO SAMPLE PRIOR TO BLOWDOWN
* RFP
* SJAE

RC/P-38

TABLE 2

SRV PNEUMATIC SUPPLY RESTORATION

SWITCH	REQUIRED SWITCH POSITION
DIV I NON-INTRPT RNA-SV-5262	OVERRIDE RESET
DIV II NON-INTRPT RNA-SV-5261	OVERRIDE RESET

STEP BASES:

This step stabilizes reactor pressure below 1050 psig to avoid SRV actuation and to permit the scram logic to be reset (if no other scram signal exists). No minimum value is specified since the reactor pressure at which the EOPs are entered cannot be predefined and the instruction must provide appropriate guidance for all events. The actual pressure band should be selected close to the initial value upon entry but below 1050 psig to permit use of available injection systems. An initial adjustment to establish an appropriate target pressure is permitted, provided the target can be reached expeditiously and the Technical Specification cooldown rate LCO is not exceeded.

STEP RC/P-38 (continued)

A pressure reduction from an intermediate pressure (e.g., 400 psig) to a pressure as low as the shutdown cooling reactor vessel pressure interlock (e.g., 75 psig) is not considered a permissible initial pressure adjustment even if the depressurization is within the LCO cooldown rate. To maintain reactor water level at intermediate or low reactor pressures, the available injection systems should not require such a pressure reduction

"Stabilize" means to hold reactor pressure as constant as practicable within the constraints imposed by the nature of the event, the degree of control afforded by the systems used, and the availability of personnel to perform manual control functions. The intent is that pressure be held as constant as is practicable. The specific actions required and the degree to which the ideal of a constant pressure can be approached will vary according to these constraints. For example:

a. If flow through a pressure control system is automatically regulated, the reactor is shutdown, and there is no break in the primary system, reactor pressure can usually be held within a fairly narrow control band with little operator action.

b. If flow through a large capacity pressure control system cannot be throttled, the reactor pressure control band will necessarily be relatively wide.

c. If pressure is decreasing, it may be necessary to close SRVs, main turbine bypass valves, or the MSIVs. In some scenarios, however, it may not be possible to terminate depressurization. Inventory loss through a break or operation of steam-driven injection systems may cause pressure to decrease.

d. Changing plant conditions may necessitate adjustments to the value at which pressure is initially stabilized. If the pressure control system in use becomes unavailable, the alternate system used in its place may not be capable of maintaining the same control band. Similarly, if an injection system in use becomes unavailable, pressure may need to be reduced to permit use of lower head systems.

Both the rate and the magnitude of reactor pressure changes must be considered. A pressure that is slowly decreasing over a relatively wide control band may be more "stable" than short-period oscillations within a narrower control band. In general, the adequacy of steps taken to stabilize reactor pressure must be judged by the effect of any continuing pressure variations on reactor water level and by whether additional actions are possible or likely to afford better control capability. If pressure variations are not interfering with reactor water level control actions, or cannot be stopped, pressure may be considered stabilized. If continuing pressure oscillations are complicating efforts to control reactor water level, or if the existing pressure prevents use of available injection systems, additional effort is warranted.

STEP RC/P-38 (continued)

The main turbine bypass valves are the preferred means of controlling reactor pressure, since they provide good control capability, are of relatively large capacity, and do not add heat to the suppression pool. The direction to use the bypass valves implicitly permits opening the MSIVs and placing the main condenser in service if such actions are necessary and conditions permit. It does not, however, constitute authorization to defeat any MSIV isolation interlocks.

If the main turbine bypass valves cannot be used to control reactor pressure or when the available capacity of the main turbine bypass valves (and main condenser) is less than that required to control reactor pressure below the high reactor pressure scram setpoint, additional systems must be employed to augment reactor pressure control.

Since symptom-oriented procedures must accommodate a full spectrum of initial plant conditions and event scenarios, no prioritization regarding the use of the listed reactor pressure control systems is specified in this step.

If Suppression Pool water level is not above the top of the SRV discharge device (-8 feet), steam discharged through the SRVs passes directly into the Suppression Chamber airspace. The magnitude of the resultant Primary Containment pressure increase could potentially exceed Primary Containment pressure limits. Therefore, use of the SRVs is prohibited when the discharge devices are not submerged.

Loss of the continuous pneumatic supply to the SRVs limits the number of times that an SRV can be cycled manually since pneumatic pressure is required for this mode of valve operation. Even though the SRV accumulators contain a reserve pneumatic supply, leakage through in-line valves, fittings and actuators may deplete the reserve capacity. Thus, subsequent to the loss of the continuous SRV pneumatic supply, there is no assurance as to the number of SRV operating cycles remaining. For these reasons, if SRVs must be used to augment reactor pressure control the continuous SRV pneumatic supply must be available. The step authorizes the use of the Group 10 override switches to allow restoration of the continuous pneumatic supply to the SRVs when they are being used for pressure control. This action may be taken prior to or after the system isolation dependent upon time, manpower, and the need or anticipated need for SRV use.

The requirement to have the SRV continuous pneumatic supply available limits the number of cycles on the valves and conserves pneumatic pressure so that, if Emergency Depressurization is subsequently required, the valve will be available for this purpose. If other pressure control systems are not capable of maintaining reactor pressure below the lowest SRV lifting pressure, SRVs will still open when the lifting pressure is reached.

STEP RC/P-38 (continued)

When manual SRV actuation is required for reactor pressure control, an opening sequence is preferred which distributes heat uniformly throughout the Suppression Pool to avoid high local pool temperatures which may result in inefficient pool cooling. The opening sequence also uniformly distributes the total number of SRV actuations among the total number of SRVs.

Use of steam driven pumps (i.e., HPCI, RCIC, and RFP) to augment reactor pressure control may be required. These systems do not draw a significant amount of steam but may be sufficient to control reactor pressure increases, or in conjunction with other systems may assist in controlling reactor pressure. Suction for HPCI and RCIC, in the pressure control mode, is always to be aligned to the condensate storage tank (CST). Use of auxiliary systems and lineups may be required to keep water in the CST.

The filter/demineralizers are bypassed when operating RWCU in the recirculation mode to minimize boron depletion if boron injection is required and to prevent chemical breakdown of the demineralizer resins while operating with elevated temperature to the nonregenerative heat exchanger.

Since operation of RWCU in the recirculation mode does not remove coolant inventory from the reactor vessel, overriding interlocks and operation of the system after boron injection (if the filter/demineralizers are bypassed) is authorized.

Operation of RWCU in the blowdown mode when boron has been injected into the reactor vessel is not permitted in order to maintain the required concentration of boron in the reactor vessel.

Reactor coolant must be sampled and analyzed for activity as prescribed by existing plant sampling procedures. Failure to determine coolant activity might result in discharge of radioactivity to the environment beyond allowable limits.

15. S295016G2.1.32 001

Consider the following normal RHR suction valve interlocks for control of Reactor Water level:

1. Shutdown Cooling pump suction valve (F006) cannot be opened unless Torus common suction valve (F020) is closed.
2. Shutdown Cooling suction isolation valves (F008/F009) will automatically isolate on low Reactor water level.

During Shutdown Cooling operation from Remote Shutdown stations per AOP-32.0, which of the above (if any) interlocks are functional?

A.✓ 1 only.

B. 2 only.

C. Both 1 and 2.

D. Neither 1 nor 2.

Feedback

Randomly selected from the bank - Q# LOI-CLS-LP-017-A*009 001

REFERENCE - AOP-32 Rev. 36 page 19

DISTRACTOR ANALYSIS
A. CORRECT - Since PCIS isolation logic originates from control room panels, the isolations are defeated by placing the valve MCC controls to Local. The F006/F020 interlock wiring does not go through the control room and is therefore functional.
B AND C - INCORRECT - PCIS is not functional
D. INCORRECT - F006/F020 is functional.

Notes

APE: 295016 Control Room Abandonment
2.1.32 Ability to explain and apply system limits and precautions.
(CFR: 41.10 / 43.2 / 45.12)
IMPORTANCE RO 3.4 SRO 3.8

55.43 (2) Facility operating limitations in the technical specifications and their bases.

This question matches the k/a since it measures the SRO's ability to explain precautions listed in AOP-32 for interlock operation while operating from remote locations.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SRO 3.8	Facility Objective:	CLS-LP-017-A*009
Ref Req'd Y or N:	NO	Technical Ref.:	AOP-32
? Cognitive Level:	C/A	? Source:	BANK LOI

CAUTION

When the Normal/Local switches for remote shutdown are placed in LOCAL, all PCIS functions are bypassed.

NOTE: Notify the TSC to sample reactor coolant if fuel failure is suspected.

(b) Station 2, CLOSE RHR Pumps B and D Suppression Pool Suction Valve, E11-F020B, at MCC 1(2)XB Compt DN6, Row G1.

(c) Station 2, CLOSE RHR Pump B Suppression Pool Suction Valves, E11-F004B, at MCC 1(2)XB Compt DK9, Row M1.

(d) Station 2, CLOSE RHR Pump D Suppression Pool Suction Valve, E11-F004D, at MCC 1(2)XB Compt DLO, Row M3.

CAUTION

R1 Interlocks associated with the E11-F007B that would normally close this valve on increased flow to prevent inadvertent draining of the reactor coolant to the suppression pool are bypassed during performance of this procedure.

(e) Station 2, IF the RHR Loop B Minimum Flow Bypass Valve, E11-F007B, at Mcc 1(2)XB Compt DL3, Row M2 is open, THEN

1) PLACE the normal/local keylock control switch to the LOCAL position and CLOSE E11-F007B at MCC 1(2)XB Compt DL3, Row M2.

(f) Station 2, VERIFY CLOSED RHR Minimum Flow Bypass Valve, E11-F007B, at MCC 1(2)XB Compt DL3, Row M2, THEN

1) PLACE the circuit breaker control switch to the OFF position at 1(2)XB Compt DL3, Row M2.

(g) CLOSE or VERIFY CLOSED the breakers for:

Station 2: E11-F008 at MCC 1(2)XDB Compt B50, Row A4
E11-F006B at MCC 1(2)XB Compt DL1, Row L2
E11-F006D at MCC 1(2)XB Compt DL2, Row L3
Station 3: E11-F009 at MCC 1(2)XA Compt DH3, Row C1

16. S295018G2.4.30 001

Unit One (1) is operating at rated power when a partial loss Reactor Building Closed Cooling Water (RBCCW) flow to the Reactor Water Cleanup (RWCU) System Non-Regenerative Heat Exchanger resulted in an automatic isolation of the RWCU Inlet Outboard Isolation Valve, G31-F004, due to RWCU Non-Regenerative Heat Exchanger discharge high temperature isolation signal. No other isolation valves were actuated. The plant remains stable at rated power.

The Shift Superintendent is assessing the need for ~~immediate~~ notifications to the NRC per 0OI-01.07, Notifications and 10CFR50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors.

Which ONE of the following identifies a proper assessment of 0OI-01.07, Notifications?

This event is:

A. ✓ not reportable per 10CFR50.72.

B. reportable per 10CFR50.72~~(b)(3)(iv)(A), ONLY.~~ within 1 hr

C. reportable per 10CFR50.72~~(b)(3)(iv)(B)(2), ONLY.~~ within 4 hrs

D. reportable per 10CFR50.72~~(b)(3)(iv)(A) and 10CFR50.72(b)(3)(iv)(B)(2).~~ within 8 hrs.

Feedback

REFERENCE - 0OI-01.07 Rev. 14 Att. 1 pages 18-24 provided as reference.

SRO candidates must be able to use 0OI.01.07 to assess need for ENS reports and must know 10CFR50.72 reference to fill ENS report.

DISTRACTOR ANALYSIS

A. CORRECT - This event is not reportable per 10CFR50.72 as item# 3.3.2 requires isolation signals affecting containment isolation valves in more than one system.

B. INCORRECT - This item# 3.3 is plausible because someone unfamiliar with using 0OI-01.07 may interpret words "Did the event result in a valid actuation of any of the systems listed below" as YES because RWCU is listed below however actual wording in 10CFR50.72 clearly states more than one general containment isolation system.

C. INCORRECT - item# 3.3.2 requires isolation signals affecting containment isolation valves in more than one system.

D. INCORRECT - item# 3.3.2 requires isolation signals affecting containment isolation valves in more than one system.

Notes

APE: 295018 Partial or Complete Loss of Component Cooling Water

2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies.

(CFR: 43.5 / 45.11)

IMPORTANCE RO 2.2 SRO 3.6

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to determine reportability requirements following an event involving a partial loss of RBCCW.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SRO 3.6	Facility Objective:	CLS-LP-200-D*019
Ref Req'd Y or N:	Y OI-01.07 ATT 1	Technical Ref.:	0OI-01.07
? Cognitive Level:	C/A	? Source:	NEW

Reportability Evaluation Checklist

NOTE:	NUREG-1022, Rev. 2 should be referenced to provide additional guidance on reportability.
NOTE:	If the answer to any of the following questions is YES, then the event is reportable within one hour.
NOTE:	If all answers to the following questions are NO, the event is not reportable within one hour. Section 2.0 of this attachment contains the guidance for making four-hour reportability determinations.

1.0 ONE-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
1.1			Is the event a deviation from technical specifications as per 10CFR50.54(X)? [50.72(b)(1)]
1.2			Has any licensed material been lost, stolen, or missing in an aggregate quantity equal to or greater than 1,000 times the quantity specified in 10CFR20 Appendix C under such circumstances that it appears that an exposure could result to persons in unrestricted areas? [10CFR20.2201(a)(i)]
			NOTE: For further information related to this item, refer to SEC-NGGC-2147, Reporting of Safeguards and Fitness for Duty Events.
1.3			Does the event involve by-product, source or special nuclear material possessed by the licensee that might have or threatens to cause:
1.3.1			Any individual's exposure to reach or exceed 25 Rems total effective dose equivalent (TEDE); 75 Rems eye dose equivalent; or 250 Rads shallow-dose equivalent to the skin or extremities? [10CFR20.2202(a)(1)]
1.3.2			The release of radioactive material inside or outside of a restricted area, such that, had an individual been present for 24 hours, the individual could have received an intake 5 times the occupational annual limit on intake? [10CFR20.2202(a)(2)]
1.4			Has any safety/relief valve failed to close? (NUREG 0626 and NUREG 0660)

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, the event is reportable within four hours.
NOTE:	If all answers to the following questions are NO, then the event is not reportable within four hours. Section 3.0 of this attachment contains the guidance for making Eight-hour reportability determinations.

2.0 FOUR-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
2.1			Is plant shutdown required by technical specifications being initiated? [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? [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? [50.72(b)(2)(iv)(B)]
2.4			Is the event a situation, as related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made? [50.72(b)(2)(xi)]
			NOTE: Such an event may include an on-site fatality or an inadvertent release of radioactively contaminated materials.

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, the event is reportable within eight hours.
NOTE:	If all the answers to the following questions are NO, the event is not reportable within eight hours. Section 4.0 of this attachment contains the guidance for making 24-hour reportability determinations.

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? [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? [50.72(b)(3)(ii)(B)]
3.3 3.3.1			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. [50.72(b)(3)(iv)(A)] These systems are: Reactor protection system (RPS) including: reactor scram and reactor trip. [50.72(b)(3)(iv)(B)(1)]
3.3.2			General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). • Main Steam Isolation. • Main Steam Line Drain Isolation. • HPCI Steam Line Isolation. • RCIC Steam Line Isolation. • RWCU Suction Isolation. • Primary Containment Isolation. • Secondary Containment Isolation. • SGTS Actuation. • Combustible Gas Control (CAD). [50.72(b)(3)(iv)(B)(2)]

Reportability Evaluation Checklist

ITEM #	YES	NO	DESCRIPTIVE QUESTION
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 [50.72(b)(3)(iv)(B)(4)]
3.3.4			Reactor Core Isolation Cooling (RCIC) [50.72(b)(3)(iv)(B)(5)]
3.3.5			Containment heat removal and depressurization systems including containment spray and fan cooler systems. • RHR Suppression Pool Cooling. • Drywell Spray System Actuation. [50.72(b)(3)(iv)(B)(7)]
3.3.6			Emergency Diesel Generators (DGs) [50.72(b)(3)(iv)(B)(8)]

Reportability Evaluation Checklist

ITEM #	YES	NO	DESCRIPTIVE QUESTION
3.4			Could the event or condition at the time of discovery have prevented the fulfillment of the safety function of structures or systems that are needed to: [50.72(b)(3)(v)] Events covered in this section may include: One or more procedural errors, Equipment failures, Discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported if redundant equipment in the same system was operable and available to perform the required safety function. [50.72(b)(3)(vi)]
3.4.1			Shut down the reactor and maintain it in a safe shutdown condition? [50.72(b)(3)(v)(A)]
3.4.2			Remove residual heat? [50.72(b)(3)(v)(B)]
3.4.3			Control the release of radioactive material? [50.72(b)(3)(v)(C)]
3.4.4			Mitigate the consequences of an accident? [50.72(b)(3)(v)(D)]
3.5			Does the event require the transport of a radioactively contaminated person to an off-site medical facility for treatment? [50.72(b)(3)(xii)]

Reportability Evaluation Checklist

NOTE:	Additional reportability guidance concerning loss of emergency preparedness capabilities is contained in NUREG-1022, Rev 2, Pages 75 through 79. It is also advisable to consult with an Emergency Preparedness representative when assessing the significance of the loss of capability.

ITEM #	YES	NO	DESCRIPTIVE QUESTION
3.6			Has the event resulted in a major loss of emergency assessment capability, off-site response capability, or communications capability (i.e., significant portion of the Main Control Room indication, emergency notification system, or off-site notification system)? [50.72(b)(3)(xiii)] Major loss of emergency or off-site notification system is considered to be/but not limited to: a. Loss of: 1) Selective signaling; OR NRC Emergency Notification System (ENS); AND 2) Commercial telephone network. b. Inoperability for ≥ one hour of: 1) Seven or more off-site sirens; OR 2) All off-site sirens in one county.

Reportability Evaluation Checklist

NOTE:	If the answer to any of the following questions is YES, then the event is reportable within 24 hours.

4.0 24-HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
4.1			Does the incident involve the loss of control of licensed material possessed by BNP which might have caused or threatens to cause:
4.1.1			Any individual's exposure in a period of 24 hours to exceed: 5 Rems total effective dose equivalent (TEDE); or 15 Rems eye dose equivalent; or 50 Rems shallow-dose equivalent to the skin or extremities? [10CFR20.2202(b)(1)]
4.1.2			The release of radioactive material inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake? [10CFR20.2202(b)(2)]

17. S295021AA203 001

Unit Two (2) is operating in MODE 3 and has established Alternate Shutdown Cooling through Safety Relief Valve (SRVs) per AOP-15 due to a loss of normal shutdown cooling. The Reactor Water Cleanup System and both Reactor Recirculation Loops are isolated. The time is now 0200 and as WCC SRO you are asked to determine the cooldown rate from the following data:

Time	0100	0200
Reactor Pressure	153 psig	67 psig
SRV tailpipe temperature	352°F	282°F
Bottom Head metal temp.	370°F	330°F
Recirc loop suction temperature	342°F	292°F

Which ONE of the following correctly identifies the approximate RPV cooldown rate over the past hour that should be used to ensure Technical Specification compliance per AOP-15.0?

A. 40 °F/hr

B. 50 °F/hr

C. 60 °F/hr

D✓ 70 °F/hr

Feedback

REFERENCE - AOP-15 Rev. 16 page 13

DISTRACTOR ANALYSIS
A. INCORRECT - Corresponds to using bottom head metal temperature
B. INCORRECT - Corresponds to using recirc suction temperature
C. INCORRECT - Corresponds to using Reactor Pressure not SRV tailpipe.
D. CORRECT - Per AOP-15 - Calculate cooldown rate with SRV tailpipe temperature

Notes

APE: 295021 Loss of Shutdown Cooling
AA2. Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING :
(CFR: 41.10 / 43.5 / 45.13)
AA2.01 Reactor water heatup/cooldown rate 3.5 3.6

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to determine RPV cooldown rate while operating in an alternate decay removal lineup following a loss of SDC.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SRO 3.6	Facility Objective:	CLS-LP-302-L*006
Ref Req'd Y or N:	Y STEAM TABLES	Technical Ref.:	AOP-15
? Cognitive Level:	C/A	? Source:	NEW

3.0 OPERATOR ACTIONS

NOTE:	Raising RPV water level slowly using CRD is preferred to reduce stresses induced in the RPV vessel and piping when injecting cold water. RHR and CS may be used if necessary, but should be considered only after determining other methods are not effective.

10. **RAISE AND MAINTAIN** reactor water level greater than 254 inches. ☐

NOTE:	The RHR pumps are preferred for injection.
NOTE:	Monitoring T_{SAT} in accordance with 1(2)PT-01.7 may **NOT** be valid under these special conditions due to reactor pressure **NOT** necessarily relating to T_{SAT}. Therefore, SRV tailpipe temperature recorder *B21-TR-R614* on Panel H12-P614, and/or ERFIS trending should be utilized for monitoring reactor coolant cool down rate.

11. **IF** any low pressure injection system, other than the RHR Loop operating in Suppression Pool Cooling, is available, **THEN PERFORM** the following:

    a. **START** one RHR or Core Spray Pump. ☐

    b. **THROTTLE OPEN** the injection valve on the affected pump until the SRV opens. ☐

12. **IF** the only low pressure injection system available is the RHR Loop in Suppression Pool Cooling, **THEN PERFORM** the following:

    a. **CLOSE** *TORUS COOLING ISOL VLV, E11-F024A(B.* ☐

    b. **THROTTLE OPEN** *OUTBOARD INJECTION VLV, E11-F017A(B),* until the selected SRV opens. ☐

13. **IF** reactor pressure can **NOT** be maintained less than 164 psig above Suppression Chamber pressure, **THEN PLACE** another SRV control switch to *OPEN.* ☐

18. S295024G2.2.22 001

With regards to the following instrumentation Limiting Conditions for Operation (LCOs):

LCO 3.3.6.1 Primary Containment Isolation Instrumentation - Group 6
LCO 3.3.6.2 Secondary Containment Isolation Instrumentation

Which ONE of the following parameters is a function required by both of the above listed LCOs?

A.✓ Drywell Pressure - High

B. Reactor Vessel Water Level - Low Level 1

C. Reactor Vessel Water Level - Low Level 2

D. Reactor Building Ventilation Temperature - High

Feedback

REFERENCE TS 3.3.6.1, 3.3.6.2

DISTRACTOR ANALYSIS
A. CORRECT - required by both
B. INCORRECT - not required by 3.3.6.2
C. INCORRECT - not required by 3.3.6.1
D. INCORRECT - not required by 3.3.6.1 or 3.3.6.2 but is an isolation signal

Notes

EPE: 295024 High Drywell Pressure
2.2.22 Knowledge of limiting conditions for operations and safety limits.
(CFR: 43.2 / 45.2)
IMPORTANCE RO 3.4 SRO 4.1

55.43 (2) Facility operating limitations in the technical specifications and their bases.

This question matches the k/a since it measures the SRO's knowledge of LCOs involving High Drywell Pressure.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SRO 4.1	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	TS
? Cognitive Level:	M OR FK	? Source:	NEW

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level—Low Level 3	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≥ 13 inches
b. Main Steam Line Pressure—Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 825 psig
c. Main Steam Line Flow—High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ 138% rated steam flow
d. Condenser Vacuum—Low	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 7.5 inches Hg vacuum
e. Main Steam Isolation Valve Pit Temperature—High	1,2,3	2	D	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 197°F
f. Main Steam Line Flow—High (Not in Run)	2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ 33% rated steam flow
2. Primary Containment Isolation					
a. Reactor Vessel Water Level—Low Level 1	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 153 inches
b. Drywell Pressure—High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 1.8 psig

(continued)

(a) With any turbine stop valve not closed.

Table 3.3.6.1-1 (page 2 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation (continued)					
c. Main Stack Radiation—High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	(b)
d. Reactor Building Exhaust Radiation—High	1,2,3	1	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 16 mR/hr
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 275% rated steam flow
b. HPCI Steam Line Flow—High Time Delay Relay	1,2,3	1	F	SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.9	≥ 4 seconds and ≤ 12 seconds
c. HPCI Steam Supply Line Pressure—Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 104 psig
d. HPCI Turbine Exhaust Diaphragm Pressure—High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 9 psig
e. Drywell Pressure—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 1.8 psig
f. HPCI Steam Line Area Temperature—High	1,2,3	1	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 200°F
g. HPCI Steam Line Tunnel Ambient Temperature—High	1,2,3	1	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 200°F
h. HPCI Steam Line Tunnel Differential Temperature—High	1,2,3	1	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 50°F

(continued)

(b) Allowable Value established in accordance with the methodology in the Offsite Dose Calculation Manual.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

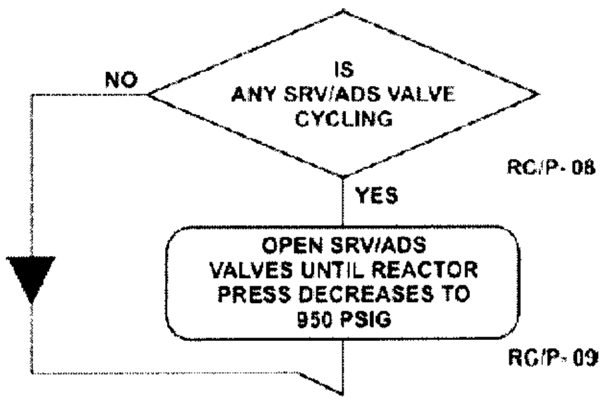
FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level—Low Level 2	1,2,3,	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 101 inches
2. Drywell Pressure—High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.8 psig
3. Reactor Building Exhaust Radiation—High	1,2,3, (a),(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 16 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in secondary containment.

19. S295025G2.4.6 001

In reference to the following steps of EOP-01-LPC, Level/Power Control Procedure:



Doesn't match K/A.

Prior to power uprate, Step RC/P-09 stated "OPEN SRV/ADS valves until reactor press decreases to 970 psig."

Which ONE of the following identifies the bases for changing the pressure from 970 psig to 950 psig?

A. The lower pressure is consistent with the new lower setpoint for the Group 1 isolation on low steam line pressure required by power uprate.

B.✓ The pressure at which all the turbine bypass valves are fully open is lower because of the higher steam line losses associated with uprated steam flow.

C. The uprated reactor has a greater heat potential thus requiring a greater margin to the opening setpoints for the SRVs .

D. The uprated reactor operates with higher fuel temperatures that require the lower coolant temperature associated with a lower steam pressure.

Feedback

REFERENCE - 0OI-37.5 Rev. 6 PAGE 47 and 89

DISTRACTOR ANALYSIS

A. INCORRECT - The change to 950 psig (100% bypass valve capacity) was made on both Unit One and Unit Two while the Group 1 setpoint change was only made on Unit Two to minimize Group 1 isolations on a SCRAM.
B. CORRECT- see reference
C. INCORRECT - 950 psig is based on 100% bypass capability and has nothing to do with a greater margin to the opening setpoints for the SRVs.
D. INCORRECT - Not based on cooling of fuel but rather minimizing heat to torus.

Notes

EPE: 295025 High Reactor Pressure
2.4.6 Knowledge symptom based EOP mitigation strategies.
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 3.1 SRO 4.0

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's knowledge of EOP symptom based steps used to prevent SRV cycling under high RPV pressure conditions.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SRO 4.0	Facility Objective:	CLS-LP-300-E*011
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.5
? Cognitive Level:	M OR FK	? Source:	NEW

STEPS RC/P-08 and RC/P-09 (continued)

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.

As SRVs are opened, reactor pressure will decrease and approach some equilibrium value dependent upon the thermal power being generated. Since the SRVs are of relatively large capacity, it is unlikely that this final pressure will exactly correspond to the pressure at which all turbine bypass valves are fully open; it will either be higher or lower. Opening SRVs "until reactor pressure drops to" the specified value will thus most likely require a pressure reduction below the target. The requisite number should still be opened, even if it results in temporary closure of some bypass valves. Since a prompt reduction in reactor pressure is desired as soon as possible to discontinue SRV cycling, adherence to a specific SRV opening sequence is unwarranted in this step.

No pressure control band or SRV opening sequence is specified here, since the purpose of the instruction is simply to reduce reactor pressure quickly and effect direct, positive control of the SRVs. Nor is any direction to close the SRVs after they are manually opened intended; the control and stabilization of reactor pressure after SRV cycling is terminated is addressed in the following steps.

REVISION SUMMARY

Revision 6 incorporates the Unit 1 MSBWP position of 00 per EC 52354. This includes use of Table 5 for conditions indicative of the reactor being shutdown under all conditions without boron. EC 50516 changes associated with Unit 1 conversion to 47% enriched boron and the changes in HSBW and CSBW related values were incorporated. These changes were to steps RC/L-30, RC/P-25, RC/P-52 discussions. Added a new caution associated with HPCI injection with suction from the CST and a CST level of less than 5 feet per EC 54587. Revised step RC/L-23 to change 170°F to 140°F per EOP Self Assessment AR 79535. Steps RC/L-06 and RC/L-07 discussion was revised to address use of PCPL-A instead of 68.5 feet as the primary containment limit per EOP Self Assessment AR 79540. Step numbers were changed in the 1(2)EOP-01-LPC and those changes are reflected in this procedure. The discussion of steps RC/Q-1 and RC/Q-2 has been revised to replace 850 psig with 835 psig due to a setpoint change for the Group 1 isolation on low pressure per EC 50552. Discussion of steps beginning LPC-9 has been revised in two locations to replace "is shutdown" with "will remain shutdown". This is the intent of the steps.

Revision 5 incorporates the new Unit 2 HSBW and CSBW related values, and the new SLC tank minimum volume value associated with use of the 47% enriched boron required by EC 46810. This revision also incorporates the new APRM downscale value for Unit 2 of 2% resulting from the installation of EC 46730 and Tech Spec Change request 2001TSC05 (BNP Letter BSEP 01-0076) dated June 26, 2001. The Unit 2 MSBWP of position 02 was added per EC 49331. The crediting of containment pressurization for meeting NPSH requirements for the RHR and CS pumps on Unit 2 was addressed as required by EC 47907. Step RC/L-50 was reworded to agree with Step LPC-4 as required by AR 79546 Assignment 9. The new Unit 2 reactor pressure value of 950 psig that results in steam flow equivalent to 100% turbine bypass valve capacity (resulting from the uprated steam flow and increased steam line losses) as required by EC 47907, was also incorporated.

Revision 4 incorporates the new Unit 1 reactor pressure value of 950 psig that results in steam flow equivalent to 100% turbine bypass valve capacity (resulting from the uprated steam flow and increased steam line losses) as required by EC 46861, Implement Extended Power Uprate. The crediting of containment pressurization for meeting NPSH requirements for the RHR and CS pumps on Unit 1 (as per EC 46861) was addressed as required.

Revision 3 incorporates the new APRM downscale value for Unit 1 of 2% resulting from the installation of ESR 00-00442 (Replace U1 Power Range Neutron Monitoring System) and Tech Spec Change request 2001TSC05 (BNP Letter BSEP 01-0076) dated June 26, 2001. New Unit 1 Minimum Alternate Flooding Pressure values were incorporated that resulted from the new GE14 fuel being loaded by EC 47777 (B1C14 Reload Core Design) as determined in EOP calculation 0EOP-WS-10. Finally, administrative type changes were made for both the Hot and Cold Shutdown Boron Weights.

20. S295029EA202 001

Unit Two (2) has experienced a reactor scram and MSIV closure. EOP-01-RVCP, Reactor Vessel Control Procedure has been entered.

SRV/ADS	Being cycled per sequence for pressure control
HPCI	Pressure Control Mode to augment SRV/ADS valves
RCIC	Being utilized for RPV level control
Supp. Pool Temp.	106°F and rising
Supp. Pool Lvl.	approaching Supp. Chamber Lvl Hi-Hi (A-01 1-5) setpoint

While Suppression Pool Cooling is being placed in service the SCO is evaluating RPV pressure control.

Which ONE of the following identifies the impact that the Suppression Chamber Level High-High condition will have on RPV pressure control using HPCI and RCIC in pressure control mode?

When the Suppression Chamber Lvl Hi-Hi setpoint is exceeded:

A. SEP-10 allows jumpers to be installed to allow HPCI and RCIC to be aligned for operation in pressure control mode.

B.✓ HPCI and RCIC are no longer available for pressure control and SEP-10 jumpers are not allowed.

C. RCIC can be immediately aligned in pressure control mode but HPCI will require jumpers to be installed per SEP-10 to allow operation in pressure control mode.

D. RCIC can be immediately aligned in pressure control mode but HPCI is no longer available for pressure control since SEP-10 jumpers are not allowed.

Feedback

REFERENCE -
APP A-01 1-5 Rev. 45 page 11
2OP-19 Rev. 102 page 31-33
SD-19 Rev. 8 pages 66 and 99
SEP-10 Rev. 11 page 2

DISTRACTOR ANALYSIS
A. INCORRECT - HPCI and RCIC are not available for pressure control as E41-F011 is interlocked closed with HPCI torus suction valves OPEN. These valves open on hihi torus level. EOPs allow jumpering Hi-Hi suppression pool level suction swap but only if suppression pool temperature approached 140°F.
B. CORRECT
C. INCORRECT - HPCI jumpers are not authorized so HPCI and RCIC are not available for pressure control with Hi-Hi suppression pool level.
D. INCORRECT - RCIC is not available for pressure control with Hi-Hi suppression pool level.

Notes

EPE: 295029 High Suppression Pool Water Level
EA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL :
(CFR: 41.10 / 43.5 / 45.13)
EA2.02 Reactor pressure..................................... 3.5 3.6

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to determine the impact that high suppression pool level will have on reactor pressure control systems.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	SRO 3.6	Facility Objective:	CLS-LP-019-A*021
Ref Req'd Y or N:	NO	Technical Ref.:	SD-19
? Cognitive Level:	C/A	? Source:	NEW

A. TITLE - Circuit Alteration Procedure

B. ENTRY CONDITION

1. As directed by "Reactor Vessel Control Procedure" (EOP-01-RCVP), OR

2. As directed by "Level/Power Control" (EOP-01-LPC), OR

3. As directed by the DW/T section of the "Primary Containment Control Procedure" (EOP-02-PCCP), OR

4. As directed by "Steam Cooling Procedure" (EOP-01-STCP), OR

5. As directed by "Alternate Emergency Depressurization Procedure" EOP-01-AEDP) OR

6. As directed by "SAMG Primary Containment Flooding", (0SAMG-01), OR

7. As directed by "Containment and Radioactivity Release Control Procedure", (0SAMG-02).

C. OPERATOR ACTIONS

CO: __ 1. IF directed to defeat the HPCI high Suppression Pool level suction transfer logic, AND Suppression Pool average water temperature is approaching 140°F, THEN PERFORM Section 1 on page 4 of this procedure.

CO: __ 2. IF directed to defeat the RWCU isolation logic, THEN PERFORM Section 2 on page 5 of this procedure.

CO: __ 3. IF directed to defeat the RCIC low reactor pressure isolation logic due to low reactor vessel pressure, AND RCIC injection is required, THEN PERFORM Section 3 on page 6 of this procedure.

4. IF:

CO: __ a. Directed to defeat the drywell cooler LOCA lockout logic due to low reactor water level, AND

CO: __ b. Actual LOCA conditions do not exist in the Drywell, AND

CO: __ c. RBCCW is operating and supplying the Drywell,

CO: __ d. THEN PERFORM Section 4 on page 7 of this procedure.

SUPPRESSION CHAMBER LVL HI-HI

AUTO ACTIONS

1. If closed, the Suppression Pool Suction Valve, E41-F042, opens.
2. If closed, the Suppression Pool Suction Valve, E41-F041, opens.
3. If open, the Condensate Storage Tank Suction Valve, E41-F004, closes.

CAUSE

1. Suppression pool water level high (-25 inches)
2. Circuit malfunction.

OBSERVATIONS

1. Suppression pool water level (CAC-LI-2601-1, CAC-LR-2602).
2. Suppression Pool Suction Valve, E41-F042, open.
3. Suppression Pool Suction Valve, E41-F041, open.
4. Condensate Storage Tank Suction Valve, E41-F004, closed.

ACTIONS

1. If the suppression pool level is actually high and the HPCI suction did not automatically transfer, perform the following:
    a. Open the Suppression Pool Suction Valve, E41-F042.
    b. Open the Suppression Pool Suction Valve, E41-F041.
    c. Close the Condensate Storage Tank Suction Valve, E41-F004.
2. Determine the cause of addition of water to the suppression pool.
3. Minimize evolutions which add water to the suppression pool as much as possible.
4. If the suppression pool level is high, drain the suppression pool to radwaste per OP-17, RHR System.
5. If a circuit malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

HPCI Auxiliary Relay E41-K19 (actuated from E41-LSH-N015A or E41-LSH-N015B)	Energized
Suppression Pool Level Switch E41-LSH-N015A	-25 inches
Suppression Pool Level Switch E41-LSH-NO15B	-25 inches

8.3 Reactor Pressure Control With HPCI in Operation

8.3.1 Initial Conditions

1. The HPCI System has automatically initiated **AND:** ☐

   a. The system is no longer required to maintain reactor water level,

   OR

   b. The automatic initiation signal is **NOT** valid.

   OR

2. The HPCI System was manually started and is no longer required to maintain reactor water level. ☐

8.3.2 Procedural Steps

NOTE: The following step isolates the RCIC CST return line.

CAUTION

After an automatic initiation, an ECCS subsystem shall **NOT** be shut down **OR** placed in manual until at least two independent indications are verified for one of the following conditions:

1. Adequate core cooling is ensured.
2. The initiation signal was **NOT** valid.
3. The system is **NOT** functioning properly in the automatic mode.

CAUTION

WHEN evolutions are in progress that have the potential to change suppression pool water level, **THEN** a second operator must be present to monitor suppression pool water level.

8.3.2 Procedural Steps

CAUTION

Suppression pool cooling should be in service whenever the HPCI System is discharging steam into the suppression pool. It is required to be placed in service if the suppression pool average water temperature increases to 95°F, in accordance with 0AOP-14.0.

CAUTION

WHEN suppression pool temperature is greater than 95°F, **THEN** suppression pool temperature is required to be logged at 30 minute intervals, in accordance with 0AOP-14.0.

CAUTION

R25 During HPCI turbine operation using reactor steam, area dose rates increase significantly. Proper ALARA precautions should be observed when entering the HPCI room.

1. **ENSURE** *BYPASS TO CST VLV, E51-F022*, is closed. ☐

2. **TRANSFER** HPCI *FLOW CONTROL* from automatic (*A*) to manual (*M*) on the controller. ☐

3. **REDUCE** HPCI turbine speed to between 3000 and 3300 rpm using the HPCI *FLOW CONTROL* manual lever. ☐

4. **OPEN** *REDUNDANT ISOL TO CST VLV, E41-F011.* ☐

8.3.2 Procedural Steps

> **NOTE:** *MIN FLOW BYPASS TO TORUS VLV, E41-F012*, will open when HPCI flow reaches 400 gpm decreasing coincident with high HPCI pump discharge pressure.

R18

> **CAUTION**
>
> The following two steps should be performed expeditiously to minimize the time *MIN FLOW BYPASS TO TORUS VLV, E41-F012*, is open. This creates a flow path from the condensate storage tank to the suppression pool.

5. **CLOSE** *HPCI INJECTION VLV, E41-F006.* ☐

> **NOTE:** **WHEN** HPCI flow reaches 800 gpm, **THEN** *MIN FLOW BYPASS TO TORUS VLV, E41-F012*, will close.

6. **WHEN** *HPCI INJECTION VLV, E41-F006*, is closed, **THEN THROTTLE OPEN** *BYPASS TO CST VLV, E41-F008*, until flow is greater than 1000 gpm. ☐

7. **ENSURE** *MIN FLOW BYPASS TO TORUS VLV, E41-F012*, has closed. ☐

8. **ADJUST** setpoint, if necessary, **AND TRANSFER** HPCI *FLOW CONTROL* from manual (*M*) to automatic (*A*) on the controller. ☐

9. **MONITOR** HPCI System operation in accordance with Section 6.0. ☐

10. **IF** desired, **THEN ADJUST** HPCI parameters as needed using guidance in Section 6.0. ☐

11. **WHEN** HPCI System operation is no longer needed, **THEN SHUT DOWN** HPCI in accordance with Section 7.0. ☐

By design, the HPCI System is required to operate independently of any external cooling. Therefore, if a loss of all cooling should occur, the system is still operable. Realistically, there will be a limit on area temperature; i.e., that temperature when the electrical components begin to breakdown and fail.

4.3.16 Standby Liquid Control

Should the Standby Liquid Control System be inoperable, the HPCI System may be used to inject liquid poison (sodium pentaborate) into the core in accordance with EOP-01-LEP-03. This procedure directs that liquid poison be directed to the CST suction vent for the HPCI System. The HPCI Pump is then started, pumping the poison into the Reactor Vessel.

4.3.17 Reactor Core Isolation Cooling

The RCIC test return line is shared with the HPCI System via redundant test return valve E41-F011. RCIC cannot be operated in the flow test (pressure control) mode if HPCI has an initiation signal or if either HPCI suppression pool suction valve is open.

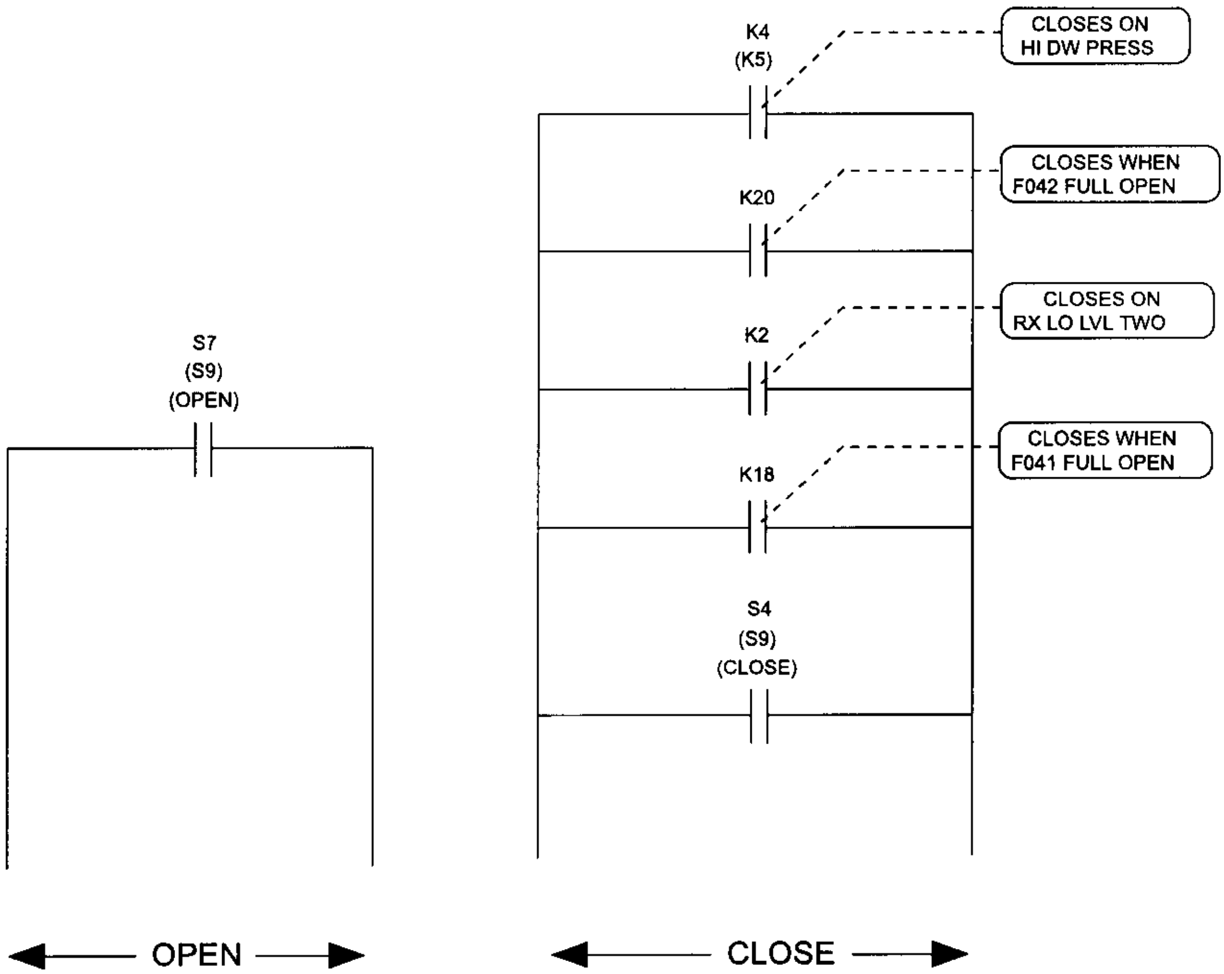
The RCIC Barometric Condenser Discharge Line was rerouted to the SBGT to prevent air injection to the containment during an emergency response. This reroute was made by connecting a tee to the HPCI Vacuum Pump discharge to the SBGT. A check valve was installed in the HPCI Vacuum Pump discharge line to allow isolation of the HPCI Barometric Condenser from the RCIC Vacuum Pump exhaust.

4.3.18 Reactor Water Level Instrumentation

The HPCI initiation and trip logic rely on nuclear boiler instruments. The initiation logic is single failure resistant due to the one out of two taken twice logic. The trip logic on high vessel water level requires both instruments to trip preventing a single sensor failure from causing a spurious trip. However, a single sensor failure can disable the trip function and would require operator action to prevent vessel overfill.

FIGURE 19-15

Test Return Isolation Valve, E41-F008 (E41-F011) Control Logic



21. S295030EA201 001

During an accident on Unit Two (2), plant conditions are:

RPV pressure	800 psig
RPV water level	+100 inches
Drywell pressure	19 psig
Suppr Pool water level	-4 ft
Suppr Chamber press	18 psig

Which ONE of the following identifies the action that is required based upon the above conditions?

A.✓ Perform emergency depressurization.

B. Initiate a reactor cooldown not to exceed 100°F/hour.

C. Rapidly depressurize the reactor to the main condenser. may be allowed to anticipate blowdown if follow EOP in sequence.

D. Cycle SRVs in sequence to initiate a reactor cooldown at a rate >100°F/hour.

Feedback

REFERENCE - EOP-UG Rev. 43 page 74. OI-37.8 Rev. 4 page 30. Unsafe region of PSPL. EOP-01-UG Attachment 5 figures to be provided as reference

DISTRACTOR ANALYSIS

A. CORRECT - UNSAFE PSPL due to combination of low suppression pool level and high suppression chamber pressure EOP-02-PCCP requires emergency depressurization.

B. INCORRECT - Meets ED requirements

C. INCORRECT - suppression pool level is low but not low enough to require alternate emergency depressurization to the main condenser.

D. INCORRECT - Must ED per procedure and OPEN 7 ADS valves.

Notes

EPE: 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)

EA2.01 Suppression pool level................................ 4.1* 4.2*

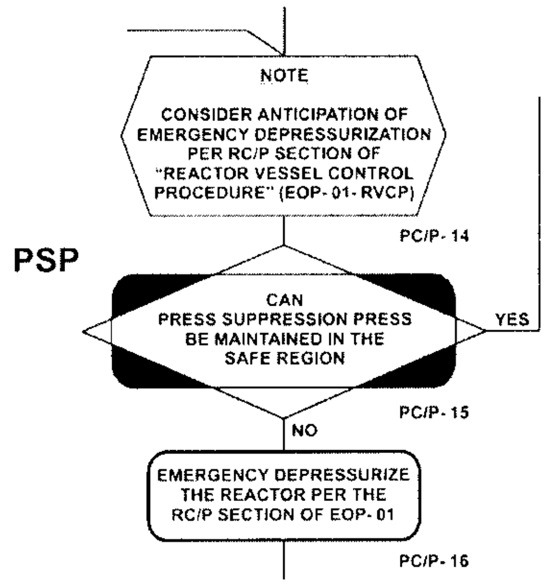
55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to interpret that suppression pool level is low enough to cause the suppression pool pressure suppression function to be inoperable thus requiring an ED.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SRO 4.2	Facility Objective:	CLS-LP-300-L*20C
Ref Req'd Y or N:	Y ATT 5 PSPL	Technical Ref.:	OI-37.8
? Cognitive Level:	C/A	? Source:	BANK LOI

STEPS PC/P-14 through PC/P-16



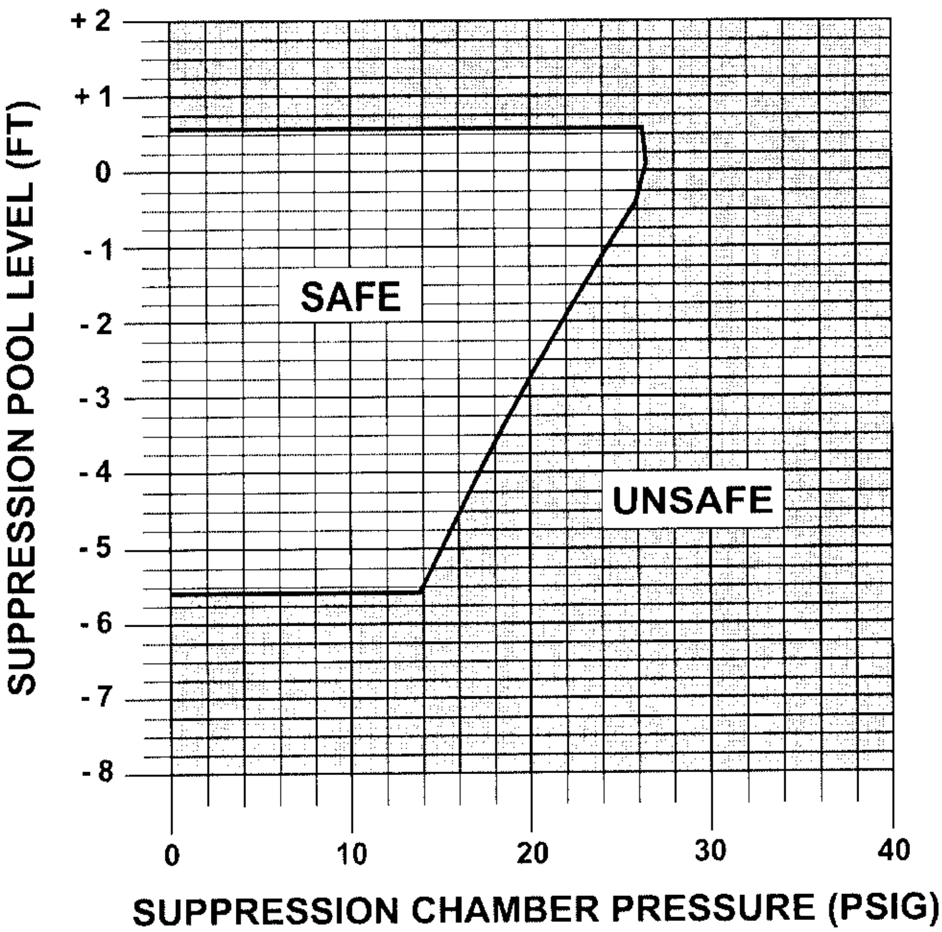
STEP BASES:

If suppression pool and/or drywell sprays could not be initiated or if operation was not effective in reversing the rising trend of primary containment pressure, as evidenced by not being able to maintain suppression chamber pressure below the Pressure Suppression Pressure, the reactor is depressurized to minimize further release of energy from the reactor vessel to the primary containment. This action serves to terminate, or reduce as much as possible, any continued primary containment pressure rise.

The Pressure Suppression Pressure is defined to be the lesser of either (1) the highest suppression chamber pressure which can occur without steam in the suppression chamber airspace or (2) the highest suppression chamber pressure at which initiation of reactor depressurization will not result in exceeding Primary Containment Pressure Limit A before reactor pressure drops to the Minimum Reactor Flooding Pressure, or (3) the highest suppression chamber pressure which can be maintained without exceeding the suppression pool boundary design load if SRVs are opened. This pressure is a function of primary containment water level, and it is utilized to assure the pressure suppression function of the containment is maintained while the reactor is at pressure. (For additional information about the Pressure Suppression Pressure see the EOP User's Guide.)

ATTACHMENT 5 (Cont'd)

FIGURE 7
PRESSURE SUPPRESSION PRESSURE



22. S295031EA201 001

Following a line break in the Unit Two (2) drywell, plant conditions are:

RPV water level	+10 inches, steady on N036/N037
RPV water level	+15 inches, steady on N026A/N026B
RPV pressure	25 psig
Drywell average temp	250°F
Drywell ref leg temp	300°F
Reactor Building 50' area	108°F

Which ONE of the following correctly describes RPV level determination under the conditions stated above?

RPV water level:

A.✓ may be determined from N036/N037 only.

B. may be determined from N026A/N026B only.

C. may be determined from either N036/N037 or N026A/N026B.

D. may NOT be determined from either N036/N037 or N026A/N026B.

If don't have Rx Bldg 50' temperature then they can rule out B, C and possibly D since this is required information per the CAUTION 1.

Feedback

REFERENCE - EOP UG CAUTION 1 should be provided as reference

Randomly selected from LOI BANK Q# LOI-CLS-LP-300-B*016 003

DISTRACTOR ANALYSIS

A. CORRECT - Conditions in unsafe region of RPV saturation limit requiring N026A/B to read above +20 inches, Fuel zone can still be used if no indications of reference leg flashing.
B. AND C INCORRECT - Conditions in unsafe region of RPV saturation limit requiring N026A/B to read above +20 inches.
D. INCORRECT - Fuel zone can still be used if no indications of reference leg flashing.

Notes

EPE: 295031 Reactor Low Water Level

EA2. Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL :

(CFR: 41.10 / 43.5 / 45.13)

EA2.01 Reactor water level............................ 4.6* 4.6*

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to interpret that reactor water level is too low for the N026 to be valid even if it is on-scale. Measure the SRO's ability to evaluate level instrument operability per CAUTION 1.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SRO 4.6	Facility Objective:	
Ref Req'd Y or N:	Y CAUTION 1	Technical Ref.:	CAUTION 1
? Cognitive Level:	C/A	? Source:	BANK LOI

ATTACHMENT 6
REACTOR WATER LEVEL CAUTION
(Caution 1)

A reactor water level instrument may be used to determine reactor water level only when the conditions for use as listed in Table 1 are satisfied for that instrument.

TABLE 1

CONDITIONS FOR USE OF REACTOR WATER LEVEL INSTRUMENTS

NOTE

Reference leg area drywell temperature is determined using Figure 13, ERFIS, or Instructional Aid based on Figure 13.

NOTE

If the temperature near any instrument run is in the UNSAFE region of the REACTOR SATURATION LIMIT (Figure 14), the instrument may be unreliable due to boiling in the run.

NOTE

Immediate reference leg boiling is not expected to occur for short duration excursions into the unsafe region due to heating of the drywell. The thermal time constant associated with the mass of metal and water in the reference leg will prohibit immediate boiling of the reference leg. Reference leg boiling is an obvious phenomenon. Large scale oscillations of all water level instruments associated with the reference leg that is boiling will occur. This occurrence will be obvious and readily observable by the operator. Additionally, if the operator is not certain whether boiling has occurred, he can refer to plant history as provided on water level recorders or ERFIS. Reference leg boiling is indicated by level oscillations without corresponding pressure oscillations.

Instrument	Conditions for Use
Narrow Range Level Instruments C32-LI-R606A, B, C (N004A, B, C) C32-LPR-R608 (N004A, B) Indicating Range 150-210 Inches Cold Reference Leg	Unit 1 Only: The indicated level is in the SAFE region of Figure 15. Unit 2 Only: The indicated level is in the SAFE region of Figure 15A.
Shutdown Range Level Instruments B21-LI-R605A, B (N027A, B) Indicating Range 150-550 Inches Cold Reference Leg	The indicated level is in the SAFE region of Figure 16. NOTE To determine reactor water level at the Main Steam Line Flood Level (MSL), see Figure 21. NOTE Figure 21 has two curves: The upper curve is for reference leg area drywell temperature equal to or greater than 200°F. The lower curve is for reference leg area drywell temperature less than 200°F.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	1. Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, <u>OR</u> B21TA103) <u>AND</u> 2. <u>IF</u> the reference leg area drywell temperature is in the UNSAFE region of the Reactor Saturation Limit (Figure 14), <u>THEN</u> the indicated level is greater than 20 inches <u>OR</u> <u>IF</u> the reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), <u>THEN</u> the indicated level is greater than 10 inches.

require this info

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Fuel Zone Level Instruments B21-LI-R610 (N036) B21-LR-R615 (N037) Indicating Range -150 - +150 Inches Cold Reference Leg	1. IF the reference leg area drywell temperature is less than 440°F, THEN the indicated level is greater than -150 inches OR IF the reference leg area drywell temperature is greater than or equal to 440°F, THEN the indicated level is greater than -130 inches. AND 2. Reactor Recirculation Pumps are shutdown. NOTE To determine reactor water level at TAF, see Unit 1 Only: Figure 17 and Unit 2 Only: Figure 17A To determine reactor water level at the minimum steam cooling level (LL-4), see Unit 1 Only: Figure 18 and Unit 2 Only: Figure 18A To determine reactor water level at the minimum zero injection level (LL-5), see Unit 1 Only: Figure 19 and Unit 2 Only: Figure 19A To determine reactor water level at 90 inches, see Figure 20. Continued on next page.

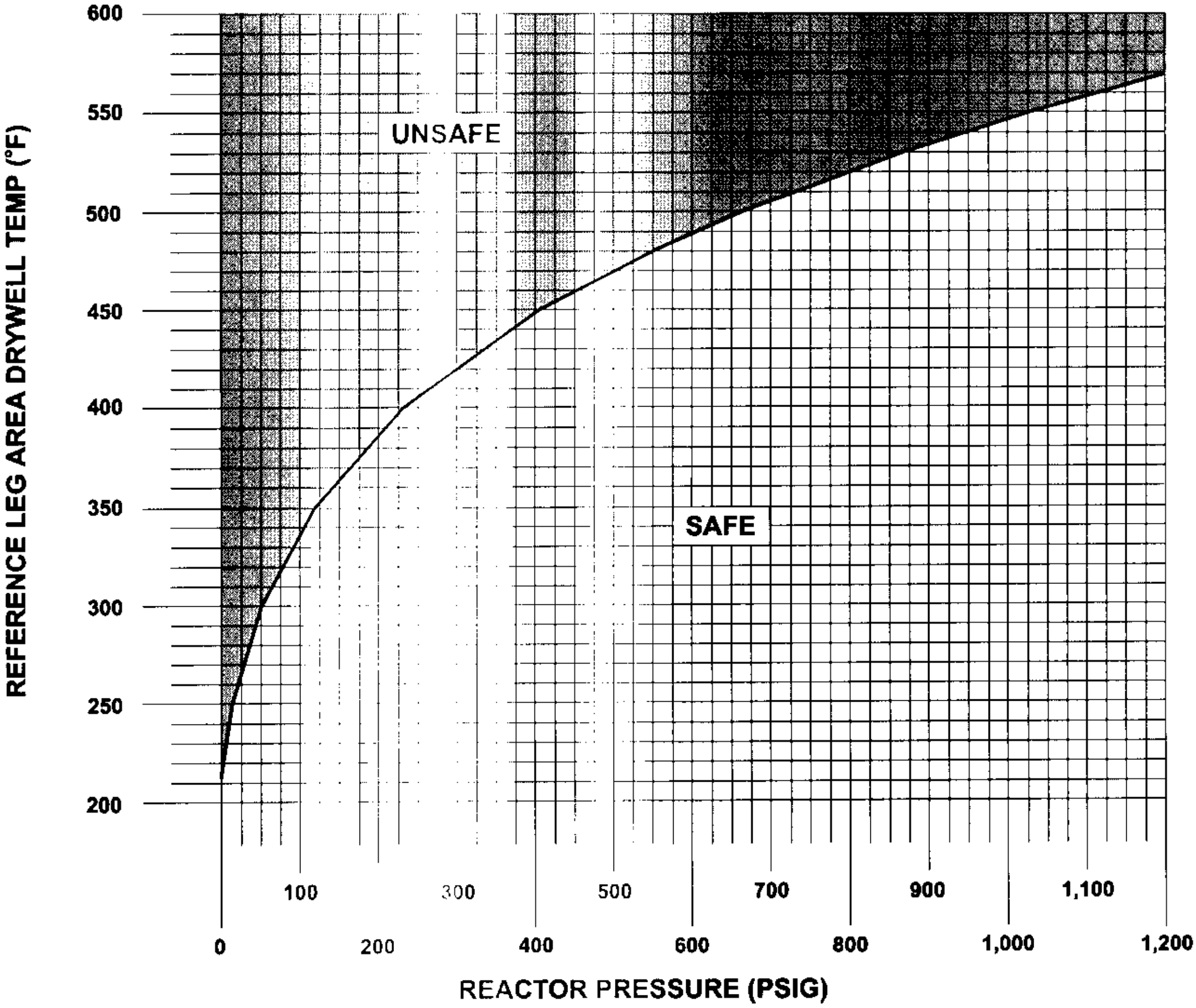
ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
	NOTE Each figure has two curves: The upper curve for reference leg area drywell temperature greater than 200°F. The lower curve for reference leg area drywell temperature less than or equal to 200°F. If containment conditions are such that reference leg area temperatures could not be controlled and maintained less than the 200°F requirement, then the upper lines on the graph should be utilized. NOTE These level instruments are valid for indication with RHR LPCI flow.

ATTACHMENT 6 (Cont'd)

FIGURE 14
REACTOR SATURATION LIMIT



23. S295034EA201 001

During accident conditions, Reactor Building HVAC has been placed in service per EOP-03-SCCP and EOP-01-SEP-04. Following restart ~~of~~ of the ventilation system, the operator notes the following:

Rx Bldg Vent Exhaust Rad Monitor A reads 2.0 mr/hr
Rx Bldg Vent Exhaust Rad Monitor B reads 3.5 mr/hr
Rx Bldg Vent Temp High (UA-03 6-2) alarm is sealed in

Which ONE of the following identifies the action that is required and the basis for this action.?

A. Continue to operate Reactor Building HVAC since the high radiation isolation function is defeated per SEP-04.

B.✓ Isolate Reactor Building HVAC and ensure SBGT is running because the rad monitor readings are not reliable.

C. Continue to operate Reactor Building HVAC until both of the rad monitor readings exceed the radiation isolation setpoint.

D. Isolate Reactor Building HVAC and ensure SBGT is running because the rad monitors read in excess of the isolation setpoint.

Feedback

REFERENCE - 0OI-37.9 Rev. 0 page 16
Randomly selected bank Q# LOI-CLS-LP-300-K*011 003

DISTRACTOR ANALYSIS
A. INCORRECT - Cannot run HVAC with INOP rad. monitors
B. CORRECT - Must isolate RBHVAC with valid signal
C. INCORRECT - Cannot run HVAC with INOP rad. monitors
D. INCORRECT - correct action wrong reason, setpoint is 4mr/hr

SRO ONLY - SRO must evaluate reliability of vent radiation monitors and interpret that readings are not reliable as Rx Building vent Temp Hi alarm indicates temp in the exhaust duct >135 deg. This invalidates the rad monitors due to EQ based on temp in vicinity of the rad monitors. SEP-04 defeats lo RPV level, hi DW press and main stack rad, but does not defeat Rx Bldg rad or vent temp, and requires isoaltion of the HVAC if either condition exists.

Notes

EPE: 295034 Secondary Containment Ventilation High Radiation
EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION :
(CFR: 41.10 / 43.5 / 45.13)
EA2.01 Ventilation radiation levels.......................... 3.8 4.2

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to evaluate rad monitor information and interpret the validity of radiation monitors under high rad. and temperature conditions.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	SRO 4.2	Facility Objective:	CLS-LP-300-K*011
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.9
? Cognitive Level:	C/A	? Source:	BANK LOI

STEPS SCCP-11 and 12 (continued)

Defeating high drywell pressure and low reactor water level isolation interlocks is appropriate, if needed, since application of these isolations to Reactor Building HVAC is for the sole purpose of limiting radioactivity release to the environment. Once assurance is provided that an excessive release of radioactivity will not occur, these two isolation interlocks become dispensable. The Reactor Building HVAC Restart Procedure (EOP-01-SEP-04) provides detailed instructions on bypassing these interlocks. The reactor building ventilation exhaust high temperature isolation is not bypassed since the radiation detectors are not qualified for operation at high temperatures which could exist following a high energy line break in the Reactor Building. If the high temperature isolation was defeated, the radiation monitors could not be relied upon to isolate the building to secure a subsequent release.

RX BLDG VENT TEMP HIGH

AUTO ACTIONS

1. Reactor Building ventilation system trips and isolates.
2. Standby gas treatment trains start.
3. If open, the inboard and outboard primary containment purge and vent valves close.
4. PASS sample valves to torus close.

CAUSE

1. High temperature in the Reactor Building exhaust plenum, 135°F.
2. Circuit malfunction.

OBSERVATIONS

1. Reactor Building air temperature monitor indicates greater than 135°F on Panel XU-3.
2. RX BLDG ISOLATED (2APP-UA-05 6-10) alarm.

ACTIONS

1. Verify auto actions.
2. If entry conditions are met, enter 0EOP-03-SCCP, Secondary Containment Control.
3. If entry conditions are met, enter 0EOP-04-RRCP, Radiological Release Control.
4. Refer to 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity.
5. If a circuit malfunction is suspected, ensure that a DLE is submitted.

DEVICE/SETPOINTS

Rad Monitor D12-TS-N010A/B 135°F

POSSIBLE PLANT EFFECTS

Possible release to environs in excess of ODCM limits.

REFERENCES

1. LL-9353-30
2. 0AOP-05.0, Radioactive Spills, High Radiation and Airborne Activity
3. 0EOP-03-SCCP, Secondary Containment Control Procedure
4. 0EOP-04-RRCP, Radiological Release Control Procedure
5. ODCM 7.3.7
6. APP UA-05 6-10, Rx Bldg Isolated

24. S295035G2.4.4 001

Unit Two (2) is operating at rated power when simultaneous and multiple fire alarms within the Reactor Building actuate. In addition the following alarms have been received:

UA-05 6-7 Rx Bldg Static Press Diff-Low
UA-12 3-3 Rx Bldg Diff Press High/Low

The CO has verified system actuations and started the Standby Gas Treatment System (SBGT) per OP-10.

Reactor Building Static Pressure indicates +0.1 inches of water on panel XU-3. Security reports that steam can be seen exiting the 117' Reactor Building siding. No radiation related annunciators have alarmed.

Which ONE of the following corrrectly identifies the emergency operating procedure(s) that the SCO is required to enter (if any) based upon these indications?

Based upon these indications the SCO is:

A. NOT required to enter any emergency operating procedure at this time as no radiation alarms have alarmed yet.

B. required to enter EOP-03-SCCP, Secondary Containment Control Procedure ONLY.

C. required to enter EOP-04-RRCP, Radiation Release Control Procedure ONLY.

D✓ required to enter EOP-03-SCCP, Secondary Containment Control Procedure and EOP-04-RRCP, Radiation Release Control Procedure.

Feedback

REFERENCES -
APP UA-05 6-7 Rev. 37 page 75
APP UA-12 3-3 Rev. 23 page 32
SD-04.1 Rev. 3 page 11
OI-37.9 Rev. 0 page 11 and 17
OI-37.10 Rev. 3 page 5

Multiple and simultaneous fire alrms indicates a HELB. The result is a loss of sec. ctmt. diff press. control which requires entry into SCCP. Steam exiting siding is an indication that blow out panels lifted on high Rx. bldg press. Once lifted secondary ctmt. is lost. With positive rx. bldg pressure and panels lifted this will result in an unmonitored release requiring entry in RRCP.

DISTRACTOR ANALYSIS
A, B, C - INCORRECT - Loss of secondary containment and unmonitored release path exist which requires entry into SCCP and RRCP regardless of radiation levels.
D - CORRECT

Notes

EPE: 295035 Secondary Containment High Differential Pressure
2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.
(CFR: 41.10 / 43.2 / 45.6)
IMPORTANCE RO 4.0 SRO 4.3

55.43 (2) Facility operating limitations in the technical specifications and their bases.

This question matches the k/a since it measures the SRO's knowledge of EOP entry conditions for secondary containment high D/P condition.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	SRO 4.3	Facility Objective:	CLS-LP-300-M*003
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.9 AND 10
? Cognitive Level:	C/A	? Source:	NEW

RX BLDG STATIC PRESS DIFF-LOW

AUTO ACTIONS

NONE

CAUSE

1. Low negative pressure differential in the Reactor Building.
2. High wind speeds
3. Circuit malfunction.

OBSERVATIONS

1. Reactor Building Static Pressure Indicator, VA-PI-1297, located on RTGB Panel XU3.

ACTIONS

1. If the low differential pressure is due to high wind speeds then refer to OP-37.1 for Supply/Exhaust fan configuration.
2. Start the standby Reactor Building exhaust fan.
3. If a standby exhaust fan is not available, stop a Reactor Building supply fan.
4. If negative pressure cannot be maintained, start SBGT System per OP-10, Standby Gas Treatment System.
5. If secondary containment integrity is required, and Reactor Building pressure cannot be maintained negative, enter EOP-03-SCCP, Secondary Containment Control.
6. Notify E&RC Counting Room that reactor building ventilation has been secured.
7. If positive pressure in the Reactor Building is indicated, (VA-PI-1297 on XU-3), trip all Reactor Building supply fans.

DEVICE/SETPOINTS

Pressure Differential Switch YA-PDS-1508	0.1 inches water

POSSIBLE PLANT EFFECTS

1. If static pressure in the Reactor Building increases to 4 inches water, the Reactor Building supply and exhaust fans trip.

REFERENCES

1. 9527-LL-9354 - 20
2. OP-10, Standby Gas Treatment System
3. OP-37.1, Reactor Building Heating and Ventilation System Operating Procedure
4. EOP-03-SCCP

RX BLDG DIFF PRESS HIGH/LOW
(Reactor Building Differential Pressure High/Low)

AUTO ACTIONS

1. Reactor Building supply and exhaust fans trip.

CAUSE

1. High or low differential pressure between the Reactor Building and atmospheric pressure.
2. Circuit malfunction.

OBSERVATIONS

1. Reactor Building Static Pressure Indicator, 2-VA-PI-1297, on RTGB Panel XU-3.

ACTIONS

1. If secondary containment integrity is required and differential pressure is low, enter 0EOP-03-SCCP, Secondary Containment Control, and execute concurrently with this procedure.
2. Inform E&RC Chemistry Reactor Building Ventilation is not in service.
3. Verify that the valve lineup is correct per 2OP-37.1, Reactor Building Heating and Ventilation System.
4. Start up the system per Section 5.1 of 2OP-37.1.
5. If a circuit malfunction is suspected, ensure that a W/O is submitted.

DEVICE/SETPOINTS

Reactor Building Static Press Hi Switch 2-VA-PDS-3779	+4.0 inches of water
Reactor Building Static Press Lo Switch 2-VA-PDS-3780	-4.0 inches of water

POSSIBLE PLANT EFFECTS

NONE

REFERENCES

1. LL-9354 - 31
2. 2OP-37.1, Reactor Building Heating and Ventilation System
3. 0EOP-03-SCCP

The dirty drains (Floor Drains) are collected by five individual sumps within the building. All sumps are installed into the -17' quadrant floor. One sump is installed in each RHR quadrant. The north RHR quadrant contains RB Sump Pump 20, the south quadrant contains RB Sump 23. The North Core Spray quadrant houses RB Sump Pump 21 while RB Sump Pump 24 is located in the South CS quadrant. RB Sump Pump 22 is in the HPCI Room. Each receives the drains from the respective areas and other floors of the Reactor Building. Each sump also contains a centrifugal pump controlled from Radwaste which pumps the liquid into the Radwaste Floor Drain Collection Tank (RW FDCT) when level reaches a preset value in the sump.

Power supplies are as follows:

COMPONENT	POWER SUPPLY
RBEDT Pump	2(1)XL
RB Sump Pumps 20, 21, 22	2(1)XL
RB Sump Pumps 23, 24	2(1)XM

Two timers attached to each of the sump and RBEDT pumps will initiate an alarm should the pump start twice prior to the timer timing out or the pumps run in excess of timer setting. Annunciators are also provided in the Control Room should quadrant water levels reach abnormally high levels.

2.4 Reactor Building Blowout Panels

In the event that a tornado should traverse the site, blowout panels are provided in the superstructure of the Reactor Buildings to relieve excess internal pressure due to the rapid depressurization of the air surrounding the structures. Stairways, hatches, and other openings are used to vent interior spaces. Blowout panels were designed to relieve at pressure of 0.75 psig (110psf/144 in^2/ft^2). This is accomplished by the use of necked-down fasteners which attach the siding to the structural steel and are sized and spaced to meet this requirement. The reminder of the siding is designed to fail at 125 psf through the use of necked-down fasteners. The failure of the siding reduces the tornado loading on the structural steel framing. The design of the structural steel framing is predicated on the failure of the siding on all sides except the windward side.

Credit is also taken in the High Energy Line Break (HELB) analysis for the 117' elevation blowout panels as well as blowout panels in the MSIV pits to relieve pressure due to a pipe break accident without affecting the structural integrity of the Reactor Building.

STEPS SCCP-1 and SCCP-2 (continued)

An area temperature or area differential temperature above its maximum normal operating level is an indication that steam from a primary system may be discharging into the Reactor Building. As temperatures continue to increase, the continued operability of equipment needed to carry out EOP actions may be compromised. High area temperatures also present a danger to personnel, a consideration of significance since access to the Reactor Building may be required by actions specified in the EOPs. The annunciator procedures associated with high ambient temperature and high differential temperatures (A-02 5-7, A-02 6-8, A-02 6-7, and A-06 6-7) require the operator to enter the SCCP as the setpoints for the annunciators are also entry conditions for this procedure.

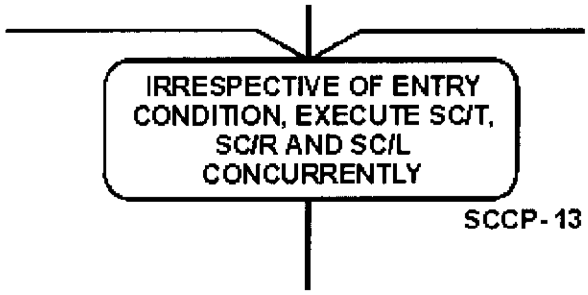
A high reactor building differential pressure is indicative of a potential loss of reactor building structural integrity and could result in uncontrolled release of radioactivity to the environment. Annunciator procedure UA-05 6-7 deals with loss of negative pressure and describes actions to remedy the situation. If the event is not caused by a malfunction of the Reactor Building HVAC System, the annunciator procedures will require entry into the SCCP. If an HVAC malfunction has caused the problem and negative pressure cannot be maintained by manipulating the Reactor Building HVAC System or starting SBGT, then the annunciator procedures will require entry into the SCCP. This will preclude entry into this procedure for problems with Reactor Building HVAC which can be immediately corrected by operator action or initiation of SBGT.

High reactor building ventilation exhaust radiation may indicate that radioactivity is being released to the environment when the system should have automatically isolated. The PROCESS RX BLDG VENT RAD HI annunciator procedure (UA-03 4-5) will direct the operator to the SCCP since the annunciator setpoint is the same as the EOP entry condition.

An area radiation level above its maximum normal operating level is an indication that water from a primary system (or from a primary to secondary system leak) may be discharging into the Reactor Building. The AREA RAD RX BLDG HIGH annunciator procedure (UA-03 2-7) provides for entry into the SCCP when entry condition levels for any areas are exceeded.

A HPCI, RHR, or Core Spray room water level above its maximum normal operating level (6 inches) is an indication that steam or water may be discharging into the Reactor Building. The annunciator procedures for HPCI, RHR, and Core Spray rooms FLOOD LEVEL HI and HI-HI (UA-12) will direct the operator to the SCCP. The HI level annunciator setpoints correspond to the entry conditions for this EOP.

STEP SCCP-13



STEP BASES:

The Secondary Containment Control Procedure (SCCP) is structured along three parallel action paths. Actions taken to control one parameter may directly affect control of the other parameters and thus all three sections are executed concurrently. Current values and trends of parameters and the status of plant systems and equipment during a transient dictate the order and priority with which specific actions are executed. This approach precludes the assignment of any particular priority to the execution of any of the individual sections.

Although reactor building pressure is an entry condition to this procedure, it is not a parameter which has any directly associated operator actions since only a small operating margin exists between its normal operating value and the pressure at which the Reactor Building (secondary containment) blowout panels release. Once secondary containment integrity has been lost by the blowout panels functioning to relieve high reactor building pressure, no mechanism is available through which reactor building pressure can any longer be effectively controlled. When available and while radiation levels permit, direction to operate the Reactor Building HVAC is specified in the procedure. This adequately addresses reactor building pressure control to the extent that procedural instructions are appropriate.

STEPS RR-1 and RR-2 (continued)

STEP BASES:

These entry conditions are more conservative than PSTG entry conditions so that guidance of several AOPs can be incorporated into this EOP. This provides a more integrated response by the operator to off-normal conditions that may escalate into emergency plan events. The PSTG entry conditions are radioactivity release rates above the off-site release rates requiring an Abnormal Radiological Effluent or Radiation Levels Alert classification. The entry conditions of this procedure are levels which are at or below instantaneous release rate limits for monitored releases or any unmonitored release. These values are sufficiently high to prevent unnecessary entry into the procedure, but low enough to provide for operator action prior to an emergency condition. Use of annunciator setpoints as entry conditions allows easy operator alerting that an entry condition has been exceeded for a parameter that may only be recorded on a back panel instrument. Entry when exceeding E&RC-2020 dose rate limits provides operator response to a combined release (such as would be experienced with fuel damage) which exceeds instantaneous gaseous release limits.

1.0 SYMPTOMS

1.1 *AREA RAD REFUEL FLOOR HIGH* (UA-03 3-7) is in alarm.

1.2 *AREA RAD NEW FUEL STORAGE HIGH* (UA-03 4-7) is in alarm.

1.3 *PROCESS RX BLDG VENT RAD HI* (UA-03 4-5) is in alarm.

1.4 Area Radiation Monitor (ARM) is in alarm.

1.5 Continuous Air Monitor (CAM) is in alarm.

1.6 Routine surveys indicate high radiation, contamination and/or airborne activity.

1.7 Report of spill, leak, or potential damage to new or spent fuel.

2.0 AUTOMATIC ACTIONS

2.1 **IF** *PROCESS RX BLDG VENT RAD HI-HI* (UA-03 3-5) is in alarm, **THEN** the following actions occur:

- Reactor Building Ventilation isolation ☐
- SBGTS auto start ☐
- Group 6 Isolation. ☐

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

R1 3.1.1 **IF** a fuel assembly was dropped or damaged, **THEN** **ENSURE** the Control Room Emergency Ventilation System (CREVS) is in operation. ☐

3.0 OPERATOR ACTIONS

3.2 Supplementary Actions

3.2.1 **EVACUATE** unnecessary personnel from the affected area. ☐

NOTE:	Simultaneous or multiple actuations of fire alarms within the Reactor Building may provide additional indication of a High Energy Line Break (HELB).
NOTE:	The Reactor Building Sprinkler System is required to be isolated within 15 minutes of indication of a HELB. Location of the system isolation valves has been provided below to expedite isolation: - *2-FP-PIV45*, Near Radwaste Building – Northeast Corner - *2-FP-PIV33*, East end of the Unit 2 Reactor Building - *1-FP-V214*, Deluge Valve Pit No. 1 next to Unit 1 Reactor Building - *2-FP-V214*, Deluge Valve Pit No. 2 next to Unit 2 Reactor Building

3.2.2 **IF** a HELB is indicated in the Reactor Building, **THEN PERFORM** the following:

1. **UNLOCK AND CLOSE** *UNIT 1(2) REACTOR BUILDING SPRINKLER SHUTOFF VALVE, 2-FP-PIV45(33).* ☐

2. **IF** additional sprinkler system isolation is required, **THEN UNLOCK AND CLOSE** *DELUGE VALVE 1(2)FP-DV20 SHUTOFF VALVE, 1(2)FP-V214.* ☐

3.2.3 **IF** new or spent fuel damage is suspected, **THEN PERFORM** the following:

1. **PLACE** any fuel that is being moved in a safe condition. ☐

2. **SECURE** further fuel movement. ☐

3. **EVACUATE** personnel from the refueling floor. ☐

4. **ISOLATE** Secondary Containment. ☐

5. **START** Standby Gas Trains. ☐

25. S2950370EA202 001

A high power ATWS occurs on Unit One (1). With reactor water level at +170 inches, injection to the reactor was terminated and prevented. Present plant conditions are:

APRM power indicates 3%
RPV water level indicated (N026) is +110 inches
Suppression Pool Temperature is 105°F
MSIVs are closed
HPCI Aux Oil Pump in Pull-To-Lock
One (1) SRV is open for pressure control

Which ONE of the following identifies the required operator action for HPCI per EOP-01-LPC, Level Power Control Procedure?

Per EOP-01-LPC, Level Power Control, the required operator action for HPCI is to:

A.✓ maintain the HPCI Aux Oil Pump in Pull-To-Lock.

B. start HPCI and maintain RPV water level between LL4 and +110 inches.

C. start HPCI and maintain RPV water level between TAF and +110 inches.

D. start HPCI and restore RPV water level to between +170 and +200 inches

Feedback

Randomly selected from LOI bank - Q# CLS-LP-300E, 009

REFERENCE - OI-37.5 Rev. 6 pages 19 and 20

SRO Only: Mitigating actions during an emergency.10CFR55.43(b)(5)
The APRM downscale set point has changed from 4% to 2%. Level must be lowered to +90" once terminate and prevent actions are performed, even if APRM downscale prior to +90".

DISTRACTOR ANALYSIS

A. CORRECT - Level band is LL4 to +90 inches
B,C,D - INCORRECT - Level band is LL4 to +90 inches

Notes

EPE: 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

EA2. Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN :

(CFR: 41.10 / 43.5 / 45.13)
EA2.02 Reactor water level.................................. 4.1* 4.2*

55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

This question matches the k/a since it measures the SRO's ability to establish the correct reactor water level band when operating under ATWS conditions.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	SR0 4.2	Facility Objective:	CLS-LP-300-E*009
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.5
? Cognitive Level:	C/A	? Source:	BANK LOI

STEPS RC/L-14 through RC/L-17

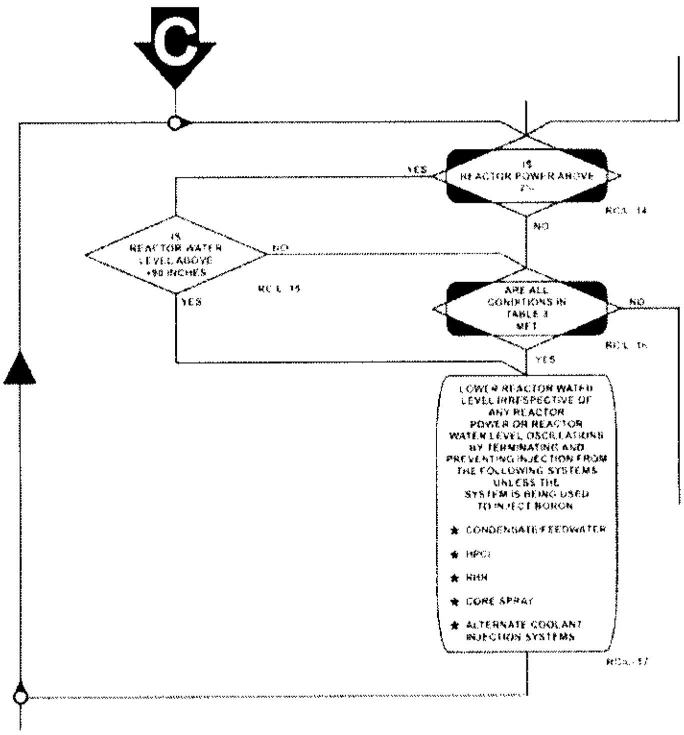


TABLE 3

LEVEL/POWER CONTROL
CRITICAL STEP RC/L-16

	YES	NO
REACTOR POWER ABOVE 2% OR CANNOT BE DETERMINED		
SUPPRESSION POOL TEMPERATURE ABOVE 110°F		
REACTOR WATER LEVEL ABOVE TAF		
ANY SRV OPEN OR BEING USED TO CONTROL PRESSURE OR DRYWELL PRESSURE ABOVE 1.7 PSIG		

STEP BASES:

To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, reactor water level is lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, the initiation and growth of oscillations is principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude.

If reactor power is at or below the APRM downscale trip setpoint (2%), it is highly unlikely that the core bulk boiling boundary would be below that which provides suitable stability margin for operation at high powers and low flows. (A minimum boiling boundary of 4 ft above the bottom of active fuel has been shown to be effective as a stability control because a relatively long two-phase column is required to develop a coupled neutronic/ thermal-hydraulic instability.) Furthermore, flow/density variations would be limited with reactor power this low since the core has a relatively low average void content.

STEPS RC/L-14 through RC/L-17 (continued)

Therefore, there is significant stability margin with power at or below the APRM downscale trip setpoint.

Twenty-four inches below the lowest nozzle in the feedwater sparger (i.e. 90 inches) has been selected as the upper bound of the reactor water level control band. This water level is sufficiently low that steam heating of the injected water will be at least 65% to 75% effective (i.e., the temperature of the injected water will be increased to 65% to 75% of its equilibrium value in the steam environment). This water level is sufficiently high that even without bypassing the low reactor water level MSIV isolation, reactor water level can be controlled with the feedwater pumps to preclude the isolation.

Lowering reactor water level is accomplished by terminating and preventing all injection into the reactor vessel, except from boron injection systems, RCIC, and CRD. Boron injection systems, RCIC, and CRD are relatively low flow systems. Boron injection systems and CRD may be needed to establish and maintain reactor shutdown conditions. When restoration of injection is subsequently required but other outside shroud injection systems are incapable of injection, continued RCIC operation (along with boron injection systems and CRD) may prevent reactor water level from dropping to the level that requires Emergency Depressurization. The marginal decrease in the rate of water level reduction resulting from continued RCIC operation has a negligible impact on lowering core inlet subcooling.

With reactor vessel injection terminated, reactor water level and reactor power decrease at the maximum possible rate allowed by boiloff. Failure to completely stop reactor vessel injection flow (with the exception of CRD, RCIC, and SLC) would delay the reduction in core inlet subcooling, thus increasing the potential for flux oscillations.

The combination of high reactor power (above the APRM downscale trip), high Suppression Pool temperature (above the Boron Injection Initiation Temperature), and an open SRV or high drywell pressure (above the scram setpoint), are symptomatic of heat being rejected to the Suppression Pool at a rate in excess of that which can be removed by the Suppression Pool Cooling system. Unless mitigated, these conditions ultimately result in loss of NPSH for ECCS pumps taking suction on the Suppression Pool, containment overpressurization, and (ultimately) loss of Primary Containment integrity, which in turn could lead to a loss of adequate core cooling and uncontrolled release of radioactivity to the environment.

The Boron Injection Initiation Temperature (BIIT) is a function of reactor power and is utilized in establishing the Suppression Pool temperature before which boron injection must be initiated if a reactor depressurization (due to exceeding the Heat Capacity Temperature Limit) with the reactor at power is to be precluded. This temperature has been chosen to be 110°F.