GGNS LOT 2/2020 **NRC** INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

RO MASTER EXAM

ANSWER KEY

1	B		26	D		51	C
2	A		27	A		52	A
3	B		28	A		53	A
4	B		29	A		54	C
5	A		30	D		55	D
6	C		31	B		56	C
7	C		32	C		57	C
8	C		33	C		58	A
9	B		34	D		59	B
10	B		35	D		60	D
11	D		36	B		61	D
12	B		37	C		62	D
13	A		38	C		63	B
14	D		39	B		64	D
15	D		40	D		65	D
16	A		41	C		66	D
17	C		42	B		67	A
18	B		43	B		68	C
19	B		44	A		69	C
20	C		45	C		70	B
21	B		46	C		71	A
22	A		47	C		72	C
23	C		48	A		73	D
24	A		49	D		74	C
25	A		50	B		75	B

<table>
<tr><th>Examination Outline Cross Reference</th><th>Level</th><th>RO</th></tr>
<tr><td rowspan="5">295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

2.2.22 Knowledge of limiting conditions for operations and safety limits.</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>2.2.22</td></tr>
<tr><td>Rating</td><td>4.0</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, Per Facility Rep, changed answer ‘C’ to the RPV pressure SL.</td></tr>
</table>

Question: 1

Which of the following **VIOLATES** a Tech Spec Safety Limit (SL)?

A. Two Recirc Pumps are running in slow speed with one Flow Control Valve at MIN ED with reactor power at 27%.

B. Core flow has just been reduced to 70 Mlbm/hr per ONEP; the STA recognizes that MCPR is now reading 1.10.

C. MSIV closure scram with reactor pressure rising to 50 psig above the ATWS/ARI initiation setpoint and all SRVs opened.

D. Plant startup with reactor pressure at 585 psig and both Reactor recirc pumps trip. the STA recognizes that MCPR is now reading 1.15.

Answer: B
Explanation: The Safety limits for GGNS: 2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow Thermal power shall be < 21.8% RTP 2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation. With core flow at 70 Mlbm/hr it is safe to assume >685 psig reactor pressure and >10% core flow, therefore, MCPR must be > 1.15 and it is 1.10. ‘B’ is correct

Distracters:

'A' is wrong, This is a low flow condition that is normal when the plant is performing a startup and shifting the recirc pumps to fast speed. This also does not meet the criteria in Tech Specs 2.1. Plausible due to a reactor recirc low flow condition.

'C' is wrong, The Reactor pressure safety limit is 1325 psig, 50 psig above the ATWS/ARI setpoint, 1126 psig would be only 1176 psig, below the SL.

'D' is wrong, First reactor pressure must be >685 psig to make MCPR an issue and MCPR of 1.15 is allowable. If power and flow were at the correct levels.

K/A Match

Knowledge of Tech Spec LCOs and Safety Limits are required to answer this question.

Technical References:

Tech Spec 2.1.2, Adm 203
051-02-III-3, Reduction in Recirculation System Flow Rate ONEP Rev. 118

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-B1300, Rev 15 OBJ. 16
GLP-OPS-TS001, Rev 16 OBJ. 3a

Question Source:	**Bank # 133**	2012 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

Examination Outline Cross Reference	Level	RO
295003 Partial or Complete Loss of AC Power AK3 Knowledge of the reasons for the following responses or actions as they apply to partial or complete loss of AC power: AK3.05 Reactor SCRAM	Tier	1
	Group #	1
	K/A	295003 AK3.07
	Rating	3.7
	Revision	1
Revision Statement: **Rev 1, Per Facility Rep, Reworded answer 'C' for clarity.**		

Question: 2

The plant is operating at rated thermal power when the normal feeder breaker to Bus 15AA trips due to a ground fault on the bus itself.

Since Bus 15AA cannot be immediately re-energized, the Loss of AC Power ONEP directs operators to manually scram the reactor.

Which of the following describes a reason for scramming the reactor?

A. Anticipates the automatic scram or control rod drifts as a result of a loss of Instrument Air to Containment.

B. It is a conservative action based on the sustainability of plant operation due to the unavailability of Plant Air Compressor 'A'.

C. It is a conservative action to place the plant in a safe shutdown and cooled down condition before a station blackout event might occur.

D. Anticipates the automatic scram that will occur on high drywell pressure as drywell temperature rises due to the loss of two drywell chillers.

Answer: A
Explanation: P53-F001, Instrument Air Supply Header To CTMT, is an air-operated valve that fails closed on loss of power to its solenoid. This Div 1 isolation valve's solenoid is powered from 15AA. Therefore, the 15AA loss fails F001 closed, cutting off Instrument Air to that header feeding CTMT. The inboard MSIVs and scram air header are loads on that header. MSIVs will begin to drift closed as air pressure lowers in their control units. An auto-scram will result from the MSIV closure. Control Rods will begin to drift in as scram air pressure lowers to the point that scram valves begin to open. Because this question is asking

for just “a reason”, this single failure mechanism is enough to justify the correct answer. Pre-empting the auto-scram by inserting a manual scram conforms to the “Conservative Decision-Making” requirements of EN-OP-115, Conduct of Operations.

Distracters:

‘B’ is wrong for the reason already discussed. Also, PAC ‘A’ is powered from Bus 16AB, not 15AA.

‘C’ is wrong. With a loss of one ESF bus the Candidate may associate this with nearing a station black out but not the reason for this action.

‘D’ is wrong. This choice is alluding to the two (of 4 total) drywell chillers that are ESF bus powered (the other 2 being BOP bus powered). However, the 2 that are ESF powered are fed from bus 16AB, not from bus 15AA.

K/A Match

Knowledge of reason for reactor scram per ONEP procedure.

Technical References:

05-1-02-I-4 (Loss of AC ONEP) Rev. 053
EN-OP-115, Conduct of Operations, Rev 026

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP OBJ. 6

Question Source:	**Bank # 466**	NRC Exam 2015
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

ESF (R20) is listed as #4 on the System Importance to CDF.

“Failure to align alternate power to 4.16 KV or 6.9 KV buses.” Is listed as #6 on Operator Action Importance to CDF.

3.1.3 **IF** it is obvious that 15AA **OR** 16AB bus **CANNOT** be promptly energized, **THEN PERFORM** following as applicable:.

a. **IF** bus 15AA is deenergized,
THEN INSERT a manual reactor scram due to loss of instrument air to CTMT and impending control rod drifts and/or MSIV closure.

b. **IF** bus 16AB is deenergized **AND** investigation per step 3.1.1 indicates bus 16AB cannot be restored before plant conditions and/or LCO action(s) requires plant shutdown,
THEN INITIATE manual reactor scram

3.1.4 **IF** a LOCA signal is present, **THEN CLOSE** 15B42 (16B42) feeder on P864 to energize MCC 15B42 (16B42).

3.1.5 **IF** Aux building isolation dampers closed on loss of power **THEN DEPRESS** SGTS DIV 1(2) MAN INIT pushbuttons (A **AND** C for Division 1, B **AND** D for Division 2).

3.1.6 **REFER** to applicable Tech Specs 3.8.1, 3.8.2, 3.8.7, 3.8.8 for required actions.

3.1 7 **IF** a diesel generator is running, **THEN** periodically **MONITOR** the emergency diesel generator locally.

a. **MONITOR** outside air fan operation.

(1) **IF** fan failed to start, **THEN START** per 04-1-01-P75-1 / 04-1-01-P81-1SOI.

(2) **IF** fan **CANNOT** be started, **THEN OPEN** doors **AND USE** temporary fans as needed to prolong diesel operation for critical loads.

3.1.8 On 1H13-P601 panel, **RESET** respective DIV 1 (2) NSSSS isolation logic **IF** possible.

3.1.9 **IF** power was lost to Bus 15AA, **THEN OPEN/CHECK OPEN** the following valves:

a. 1P41-F239

b. 1P41-F240

NOTE
Control Room Air Conditioning Units Z51-B002A(B) Temperature Control Valves fail open on loss of instrument air, causing compressors to unload on low freon pressure, thus reducing unit cooling capacity.

5.0 INSTRUCTIONS

NOTE
Implementation of the Prevention, Detection, Correction (PDC) model IAW EN-PL-100, Nuclear Excellence Model and EN-HU-102, Human Performance Traps & Tools is everyone's responsibility.

5.1 Conditions Requiring an Immediate Manual Reactor Scram

1. **IF** ANY of the following conditions occur,
   THEN Licensed Operators immediately **INSERT** a manual scram:

   - Safety of the reactor is in jeopardy
   - Operating parameters exceed any of the reactor protection set points and an automatic shutdown does not occur
   - Prior to taking any manual action that will result in an automatic scram
   - Core thermal hydraulic instability is observed
   - As directed by plant procedures

5.2 Conservative Decision Making

1. **ENSURE** safe operation of the facility takes precedence over all other considerations including economic and competitive pressures. (SOER 94-01)

2. **ENSURE** Nuclear and industrial safety is maintained at the forefront of all decisions. (SOER 94-01)

3. **WHEN** reactor safety is uncertain,
   THEN REDUCE power or **PERFORM** an immediate reactor shutdown.

4. **MANAGE** risk by understanding and controlling plant status to ensure operating margin is maintained.

5. During electrical equipment failures or malfunctions, **TAKE** action to provide positive isolation of damaged equipment to prevent inadvertent re-energization (SER 3-10).

6. **AVOID** hasty decisions - there are few time critical actions that require immediate response.

7. **ENSURE** crew recognizes time critical decisions based on a degrading trend in operating margin.

Examination Outline Cross Reference	Level	RO
295004 Partial or Total Loss of DC Power AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : AA1.03 A.C. electrical distribution	Tier	1
	Group #	1
	K/A	295004 AA1.03
	Rating	3.4
	Revision	1
Revision Statement: **Rev 1. Per Ops Rep, Changed order of distractors in the description section.**		

Question: 3

Bus 15AA is de-energized.

Battery bus 11DA is de-energized.

Circuit breaker 152-1511, FDR FRM ESF XFMR 12, is available to energize bus 15AA.

Which of the following describes the control for the circuit breaker?

A. Will still have automatic trip capability.

B. Close and reopen once at the breaker.

C. Not be capable of remote or any local operation.

D. Open remotely but must be closed manually, at the breaker.

Answer: B
Explanation: Any 4160 breaker cannot be controlled remotely without DC, all operations must be done locally. Any breaker that is already closed when the loss of DC occurs can be opened due to the opening springs are compressed by the closing of the breaker, the opening springs are also compressed when the breaker is closed. If the breaker is already open then the breaker can be closed and reopened once.

Distracters:

‘A’ is wrong; trip coil is DC powered.

‘C’ and ‘D’ are wrong; the breaker will not operate remotely without DC power, but will operate manually at the breaker as described above.

K/A Match

Knowledge of Electrical distribution breakers to an ESF bus with a loss of DC control power.

Technical References:

05-1-02-I-4, Loss of AC ONEP Rev. 053
05-1-02-V-19, Loss of 125VDC, Rev 01
GLP-OPS-L11, Plant DC System Lesson Plan, Rev 20
04-1-01-L11-1, Plant DC System SOI, Rev 129

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-L11 OBJ. 10.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3.0	

PRA Applicability:

Plant DC is listed as #7 on the System Importance to CDF

Lesson Content	Instructor Notes

- <u>Alarms</u>
  - The undervoltage (27) devices and the ground detection (64) devices generate computer alarms only; there is no control room annunciator associated with any of these buses.

A. SYSTEM INTERRELATIONSHIPS

1. <u>Normal AC and ESF Power Distribution Systems</u> — *Objective 10.1*
   - The DC Power System supplies control power to the 480 V LCC, 4.16 KV, 6.9 KV and 13.8 KV circuit breakers.
   - Loss of DC Power will result in the inability to remotely operate circuit breakers.
2. <u>480V AC, ESF Division I, II and III</u> — *Objective 10.2*
   - Supplies power to the battery chargers for the Plant DC System Interface
     - Bus 11DA Battery Chargers 1A4 (LCC 15BA6) and 1A5 (LCC 15BA3)
     - Bus 11DB Battery Chargers 1B4 (LCC 16BB6) and 1B5 (LCC 16BB3)
     - Bus 11DC Battery Charger 1C4 (MCC 17B01)
     - Bus 11DD Battery Chargers 1D4 (LCC 15BA1) and 1D5 (LCC 15BA2)
     - Bus 11DE Battery Chargers 1E4 (LCC 16BB1) and 1E5 (LCC 16BB2)
     - Bus 11DK Battery Chargers 1K4 (LCC 15BA1) and 1K5 (LCC 15BA2)
     - Bus 11DL Battery Chargers 1L4 (LCC 16BB1) and 1L5 (LCC 16BB2)
   - Loss of ESF Division I and II power supplies results in loss of the battery chargers to buses 11DA, 11DB, 11DD, 11DE, 11DK and 11DL.
3. <u>480V AC, BOP</u> — *Objective 10.1*
   - Supplies power to battery chargers for the Plant DC System.
     - Bus 11DC Battery Chargers 1C5 (MCC 13B11)
     - Bus 11DG Battery Chargers 1G4 & 2G5 (LCC 18BG1) and 1G5 & 2G4 (LCC 28BG1)

Examination Outline Cross Reference	Level	RO
295005 Main Turbine Generator Trip AA2. Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: AA2.04 Reactor pressure	Tier	1
	Group #	1
	K/A	295005 AA2.04
	Rating	3.7
	Revision	1
Revision Statement: **Rev 1, Per Facility Rep, changed wording in answer 'D'**		

Question: 4

A main turbine / generator trip has occurred at 45%.

Which one of the following describes the effect on reactor pressure?

A. held constant at 935 psig.

B. initial rise then controlled at approximately 935 psig.

C. initial rise then controlled between 926 psig and 1033 psig.

D. held constant at 935 psig then lowers to and stabilizes at approximately 850 psig.

Answer: B
Explanation: At 45% power and a turbine trip occurs a Reactor Scram will also occur. Pressure control will be on the Main Bypass valves but initially Reactor pressure will rise then be controlled at 935psig due to the setpoint on Pressure Set is 935 psig.
Distracters: 'A' is wrong; Even though the bypass valves will open, reactor pressure will rise before they open to maintain pressure, plausible due to too much power is being produced not to have a pressure rise. If it were a lower power then a scram would not have occurred and pressure would have remained constant. . 'C' is wrong, this assumes the MSIVs have closed due to the pressure band given is the open and close setpoints for SRVs on LO-LO set. Plausible due to setpoints on the LO LO set system

‘D’ is wrong; the setpoint can only be manually lowered below 935. Plausible due to the 850 psig number is the lower band number for manual pressure control in the EOPs
K/A Match Knowledge of reactor pressure changes during a Turbine trip.
Technical References: 05-1-02-I-4, Turbine and Generator Trips Rev. 041 GLP-OPS-N3202, Main Turbine EHC Control System Lesson Plan, Rev 11 03-1-01-1, Cold S/D to Generator Carrying Min Load IOI, Rev 183
Handouts to be provided to the Applicants during exam: None
Learning Objective: GLP-OPS-N3202 OBJ. 4.0, 13.0

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

COLD SHUTDOWN TO GENERATOR CARRYING MINIMUM LOAD
REACTOR HEATUP AND PRESSURIZATION
SAFETY RELATED

STEP	ACTIVITY	INITIALS

NOTE

SRVs actuators Shall be stroked **OR** the SRVs Shall be manually opened once per 18 months per Tech Spec SR 3.4.4.3 **AND** SR 3.5.1.7. It is desired to perform surveillance each Outage per 06-ME-1B21-R-0008, however, 06-OP-1B21-R-0002 May be performed to satisfy surveillance requirements.

43. **IF** scheduled, **THEN PERFORM** 06-OP-1B21-R-0002 within 12 hours after Reactor Steam Dome Pressure is greater than **OR** equal to 800 psig. __________

44. **WHEN** RFPT discharge pressure is greater than 800 psig, **THEN PLACE** the zinc skid in operation at the flowrate recommended by Chemistry per SOI 04-1-01-N21-1. __________

~ 935 psig

NOTE

The desired TURB STM PRESS DEMAND setpoint is 935 psig for rated conditions. **ADJUST** setpoint between 930 psig **AND** 940 psig to maintain 935 psig. **MONITOR** Turbine Steam Pressure Demand periodically, **USING** Turbine Steam Pressure Demand-LTD **AND** computer point N32K601, **WHEN** setpoint varies from 935 psig.

45. **WHEN** TURB STM PRESS DEMAND setpoint has been raised to 935 psig, **THEN STOP** raising pressure setpoint, **AND** allow Bypass Valves to continue to open as power is increased. __________

46. **DISCONTINUE** taking data on Data Sheet I and **ATTACH** to this instruction. **ENSURE** the last reading is taken after Step 45. __________

47. **CLOSE** the following Main Steam line drain Valves on 1H13-P601:

    B2-F021, INBD MSL DR TO MN CNDSR __________

    B21F068, OTBD MSL DR TO MN CNDSR __________

Examination Outline Cross Reference	Level	RO
295006 SCRAM AK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM : AK1.02 Shutdown margin	Tier	1
	Group #	1
	K/A	295006 AK1.02
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 5

A reactor scram has occurred.

Per Technical Specifications, which of the following would still ensure that sufficient shutdown margin is maintained and that the reactor will remain subcritical under all conditions?

A. One center core control rod is at position 48.

B. Two peripheral control rods are at position 48.

C. 50% of the control rods are at position 02 or beyond.

D. Groups 1 through 4 of the control rods are at position 04.

Answer: A

Explanation:

Per Tech Specs definitions, "SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: (c) All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.

After a scram the crew can determine the reactor shutdown with only one control rod not full in. If more than one is withdrawn then a calculation must be performed.

Distracters:

'B' is wrong; Even if it is a peripheral rod only one can be withdrawn.

.

'C' and 'D' are wrong, see the above explanation. Plausible due to 'C' is used in EP-2A to determine if the reactor can remain shutdown and 'D' is used when determining core density.

K/A Match Knowledge of concept of shutdown margin after a reactor scram.		
Technical References: Tech Spec Section 1.1 Definitions		
Handouts to be provided to the Applicants during exam: None		
Learning Objective: GLP-OPS-TS001 OBJ. 4.13		
Question Source:	**Bank # 721**	2013 NRC (Q5)
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is $\geq$68°F; corresponding to the most reactive state, and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

Examination Outline Cross Reference	Level	RO
295016 Control Room Abandonment AK2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: AK2.02 Local control stations: Plant-Specific	Tier	1
	Group #	1
	K/A	295016 AK2.02
	Rating	4.0
	Revision	1
Revision Statement: **Rev. 1, Validation, added switch number for transfer switch.** **Rev 2, added Div 2 to answer A**		

Question: 6

The control room has been abandoned due to a fire in panel P864.

Control of the plant has been established at the Remote Shutdown Panels (RSPs).

At local panel P152 (Area 25A, El. 111'), operators have placed the following switch in the ON position:

- Transfer Switch for Lockout Transfer Relay**, C61-HSS-M150**

When placing this switch in the ON position, it isolates what equipment from the control room?

A. Isolates ALL the Div 2 ESF powered equipment.

B. Isolates BOTH Div 1 and 2 ESF powered equipment.

C. Isolates ALL the equipment controlled from RSP P150.

D. Isolates BOTH equipment controlled from RSPs P150 and P151.

Answer:	**C**
Explanation: Per the site electrical drawings, C61-HSS-M150 energizes lockout relays on H22-P152 which electrically isolate components operated at the H22-P150 (Division 1) Remote Shutdown Panel from the Main Control Room. Other selected Division 1 powered/controlled equipment have isolations provided on Alternate Shutdown Panels and the Diesel Control Panel. Not all Division 1 equipment is affected by Alternate Shutdown /Remote Shutdown Panels and NO Division 2 equipment is affected.	

Distracters:

'A' is wrong. Plausible because division 1 equipment only is bypassed. The transfer switch does not effect any Division 2 equipment.

'B' is wrong. Plausible because P150 and P151 are division one and division two equipment only. The transfer switch does not isolate ALL division one and two equipment from control room.

'D' is wrong. Plausible because P150 is isolated but the P151 is not and is manned at the same time.

K/A Match

Knowledge of Alternate Shutdown /Remote Shutdown Panels and their interrelations with the Main Control Room.

Technical References:

05-1-02-II-1, Shutdown from Remote Shutdown Panels ONEP sections 1.3; 1.10; Attachments III, IV, and XXI.
GLP-OPS-C6100

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP, objective 54; 55
GLP-OPS-C6100, objective 11

Question Source:	**Bank # 579**	2015 NRC
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

LESSON BODY	NOTES
Reactor water level indication is from wide range, calibrated for the wide range conditions.	***Figures 1 and 2***
The level indicator, spanned for a wider range than the normal wide range level indication, provides water level indication from normal operating conditions to cold shutdown conditions.	

Alternate Shutdown Panels P152, P295, P296, P298, and P299

4.2 The purpose/function of the Alternate Shutdown Panels is to provide isolation/control capabilities for Division I equipment necessary for the safe shutdown of the reactor in the event of a Control Room fire. Use of the Alternate Shutdown Panels is also used in the event of a Security Threat. **5.2** The Alternate Shutdown Panels are located in the following areas:	***Shutdown from the Remote Shutdown Panels ONEP, Section B, Subsequent Actions***
P152 – Area 25, Elevation 111' P295 – Area 7, Elevation 119' P296 – Area 9, Elevation 119' P298 – Area 7, Elevation 166' P299 – Area 25, Elevation 111'	***Review appropriate Attachments of the ONEP that delineate the lineup of the Alternate Shutdown Panels***
Panel P152 consists of one, two-position, transfer switch referred to as "Lockout Transfer Relay C61-HSS-M150" and 36 lockout-type relays. **8** Both the transfer switch and Panel P152 are powered from 125V DC Bus 11DA.	***♫ Knowledge: Before operating a component, confirm an understanding of its function and interactions with other components.***
With Lockout Transfer Relay C61-HSS-M150 in the OFF position, the relays are reset and the panel is in standby. **6, 9, 10** When the transfer switch is taken to ON, the relays trip. This isolates and disables Control Room indications and controls for the Division I components listed in ONEP 05-1-02-II-1, Shutdown from the Remote Shutdown Panel.	***Refer to ONEP 05-1-02-II-1, Shutdown from the Remote Shutdown Panel, and review appropriate Attachment.***
11 When the relays trip, an alternate set of control power fuses is placed in service for the components of Table 1. If a fire has occurred, the normal control power fuses may have blown due to electrical faults.	***Table 1***

SHUTDOWN FROM THE REMOTE SHUTDOWN PANEL

HANDSWITCH LINEUP EFFECTS

NOTE

C61-HSS-M150 at 1H22-P152 trips all lockout transfer Relays R1 through R36 with the following system effects.

LOCKOUT Transfer Relays 1C61-R1 Through R36

1) For the following transfer relays Will affect valve circuits are follows:
   a) Disables Control Room indication **AND** control
   b) Disables AUTO open/close signals
   c) Valve has light indication at RSD panel (always)
   d) Valve has control at RSD panel (always)

Lockout Transfer Relay	Component Affected
1C61-R1	1P41-F001A, SSW A DISCH MOV
1C61-R2	1P41-F014A, SSW TO RHR HX A INLET MOV
1C61-R3	1P41-F068A, SSW TO RHR HX A OUTLET MOV
1C61-R4	1P41-F007A, SSW A BASIN TRANSFER MOV
1C61-R5	1P41-F006A, SSW A RECIRC TRANSFER MOV
1C61-R6	1P41-F018A, D/G 11 HX INLET MOV
1C61-R7	1E12-F011A, RHR HX TO SUPP POOL MOV
1C61-R9	1P41-F005A, SSW CLG TWR A RETURN MOV
1C61-R13	1E12-R027A, RHR A INJECTION SHUTOFF VLV
1C61-R14	1E12-F042A, LPCI A INJECTION VALVE **AND** 1E12-F003A, RHR HX A SHELL SIDE OUTL VLV
1C61-R15	1E12-F040A, RHR A DISCH TO RWSTE **AND** 1E12-F048A, RHR HX A BYPASS VALVE
1C61-R17	1E12-F006A, RHR A SDC SUCTION VLV **AND** 1E12-F008, RHR SDC OUTBD ISLN VLV
1C61-R18	1E12-F024A, RHR TEST RETURN VLV, (1E12-F024A Will AUTO CLOSE, **IF** 1E12-F004A IS FULL CLOSED)

SHUTDOWN FROM THE REMOTE SHUTDOWN PANEL

HANDSWITCH LINEUP EFFECTS (Cont.)

LOCKOUT Transfer Relays 1C61-R1 Through R36 (Cont.)

Lockout Transfer Relay	Component Affected
1C61-R19	1E12-F053A, SDC RTN TO FDW
1C61-R20	1E12-F004A, RHR A SUPP POOL SUCT VLV **AND** 1E12-F047A, RHR A HX SHELL SIDE INLET VALVE
1C61-R22	TURBINE TRIP & THROTTLE VLV, (TRIPS Will still function)
1C61-R23	1E51-F022, TEST RTN TO CST
1C61-R24	1E51-F059, TEST RTN TO CST
1C61-R25	1E51-F031, SUCT FROM SUPP POOL
1C61-R26	1E51-F010, SUCT FROM CST
1C61-R27	1E51-F046, RCIC LUBE OIL CLR WATER VLV
1C61-R29	1E51-F013, RCIC INJECTION VLV
1C61-R30	1E51-F045, RCIC STM SUPPLY VLV
1C61-R32	1E51-F019, MIN FLOW VLV
1C61-R33	SRVs 1B21-F051A, F051B, F047D
1C61-R34	SRVs 1B21-F051D, F047G, F051F
1C61-R35	1E51-F095, BYPASS VLV FOR 1E51-F045
1C61-R36	CRDH SYS 1NC11C001A-A AUX OIL PMP

Examination Outline Cross Reference	Level	RO
295018 Partial or Total Loss of CCW AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENTCOOLING WATER : AA2.02 Cooling water temperature	Tier	1
	Group #	1
	K/A	295018 AA2.02
	Rating	3.1
	Revision	0
Revision Statement:		

Question: 7

The plant is operating at rated conditions.

CCW Temperature Control Valve (P44-F501) has failed closed due to a controller failure.

Which of the following will auto isolate due to the effects of this condition?

A. Control Rod Drive Pumps

B. Reactor Recirculation Pumps

C. Reactor Water Cleanup System

D. Fuel Pool Cooling and Cleanup System

Answer: C
Explanation: With the Temp control valve failing close the CCW temp will rise causing RWCU temp to rise and cause the G33-F004 to auto close at 140 degrees Non regen outlet temp. Therefore, B is the correct answer.
Distracters: All distracters are plausible due to being loads of the CCW system. 'A' and 'B' is wrong, but plausible will not auto isolate. 'D' is wrong but plausible because it will isolate on low flow not high temp
K/A Match

Knowledge of effects on system loads due to high temperature.		
Technical References: 04-1-01-G33-1 Step 3.1		
Handouts to be provided to the Applicants during exam: None		
Learning Objective: GLP-OPS-G3336 Objective 8.2		
Question Source:	**Bank # 723**	2013 NRC (Q7)
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

1.2 Changes required for implementation of 1994 TSIP were incorporated in Revision 100. For historical reference this statement should not be deleted.

2.0 ATTACHMENTS

2.1 Attachment I - Manual Valve Lineup Checksheet

2.2 Attachment II - Remote Operated Valve Lineup Checksheet

2.3 Attachment III - Electrical Lineup Checksheet

2.4 Attachment IV - System Alarm Index

2.5 Attachment V - Alteration Record Sheet

2.6 Attachment VI - Defeating Interlocks for Fill and Vent per Section 5.7.

2.7 Attachment VII – Actions for Loss of Plant Chilled Water with Noble Chem in service.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 During system operation, non-regenerative heat exchanger outlet temperature indicated on TI-R607 RWCU temperature, position 5, **AND** non-regenerative HX OUTL TEMP on 1H13-P680 Should be maintained less than 130°F. This is to avoid resin damage **AND** a system isolation at 140°F.

3.2 Time unfiltered water is being rejected to main condenser Should be minimized. This condition Can introduce waterborne radioactive contaminants into Condensate System.

3.3 During reactor blowdown operations, do **NOT** open F046, RWCU BLWDN VLV TO MN CNDSR **OR** F041, RWCU BLWDN TO MN CNDSR BYP **AND** F035, RWCU BLWDN VLV TO RADWST simultaneously with vacuum in main condenser. Vacuum Will be lost via radwaste.

Examination Outline Cross Reference	Level	RO
295019 Partial or Complete Loss of Instrument Air AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: AK2.09 Containment.	Tier	1
	Group #	1
	K/A	295018 AK2.09
	Rating	3.3
	Revision	0
Revision Statement: **New Question**		

Question: 8

The plant is operating at rated power when a total loss of instrument air to the Containment occurs.

Which of the following describes the effects on Containment and Drywell cooling water?

	Containment	Drywell
A.	Available	Available
B.	Available	Unavailable
C.	Unavailable	Available
D.	Unavailable	Unavailable

Answer: C

Explanation:

A loss of Instrument Air will cause the Containment isolation valves for Plant Chilled water to auto close, PCW supplies the containment coolers. A loss of PCW will cause the Containment Temperature to rise. This isolation for this system cannot be overridden to allow cooling.

A loss of Instrument Air has no effect on the Drywell Chilled Water System, because a different cooling water system supplies the Drywell coolers. The Drywell Chilled Water System uses MOVs for Ctmt and Drywell isolation valves.

‘C’ is Correct

Distracters:

‘A’ is wrong – but plausible, the candidate must recognize the loss of air only affects air operated valves not MOVs therefore Cooling to the Containment fan is isolated but Drywell cooling is not.

‘B’ is wrong – opposite effects

‘D’ is wrong - but plausible, the candidate must recognize the loss of air only affects air operated valves not MOVs therefore Cooling to the Containment fan is isolated but Drywell cooling is not

K/A Match

Knowledge of effects on the containment with a loss of instrument air.

Technical References:

04-1-01-P71-1, PCW, step 3.6
04-1-01-P72-1, Drywell Chilled Water

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-P7100, objective 13.1
GLP-OPS-M5100, Objective 6.5 & 9.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

Plant Air is listed as #15 on the System Importance to CDF.

3.6 During plant operations without Plant Chilled Water, area temperature Could exceed maximum allowable for mechanical **AND** electrical equipment, at which time plant shutdown Could be required.

3.7 Oil level in the oil feeder bowls supplying **ALL** pumps Should be **CHECKED** periodically.

3.8 During light heat load conditions (single chiller operation) **SUPPLY** Plant Service Water only to the chiller in operation to prevent refrigerant migration into the oil of the chillers **NOT** in operation.

3.9 **IF** a chiller starts to surge **CORRECT** problem **OR** Shutdown the Chiller Unit . **MINIMIZE** the amount of time that the Chilled water is lowered below design flow duing system swaps.

3.10 Plant Chilled Water supplies cooling water to the Circ Water pump thrust bearing oil coolers. **BEFORE** performing **ANY** operation that May significantly reduce cooling to a running Circ Water pump **AND** cause approaching 150 deg F Circ Water pump motor thrust bearing oil temperature indicated on 1N71-TIS-N045A(B), **SHIFT** the cooling water supply to circ water blowdown per 04-1-01-N71-1."

3.11 **ATTEMPT** to schedule Plant Chiller rotations **OR** Plant Chiller wet layups during PSW chemical additions to improve water chemistry in the chillers that are taken out of service.

3.12 EC# 0026 installed gags on P71-F300 through P71-F307. An open signal Must be applied to
the control room handswitch for these valves following a loss of power **OR** loss of instrument air because the actuator Would be attempting to close the valve **AND** **IF** a gag fell out these valves Would close. The actuator Would also be applying force to the gag also **IF** an open signal is **NOT** present.

3.13 MCC breakers feed the Aux Oil Pump Motors **AND** heaters 52-132208 (Plant Chiller "A"), 52-132152 (Plant Chiller "B") **AND** 52-141107 (Plant Chiller "C").

3.14 With a Steam Line Tunnel Cooling Fan (1T41C001A(B)) in standby, the associated PCW Steam Tunnel Cooler Inlet MOV 1P71FA14A(B) will automatically open when the Auxiliary Building Steam Tunnel temperature reaches 107°F.

CTMT
TEMP
HI

04-1-02-1H13-P870-3A-F3	
Revision: 103	Page 1 of 2

Safety Related

Alarm Device 1M71-TAH-L609A

1.0 POSSIBLE CAUSES

NOTE

Containment temperature points #7 thru #10 on 1M71-R604A for Elevation 208 and 240 temperatures do not alarm until 95°F.

1.1 Containment temperature of 90°F due to:

1.1.1 Loss of containment cooling.

1.1.2 Steam leak in the containment.

1.1.3 Loss of coolant accident.

1.1.4 Fire in the containment.

1.1.5 Possible hydrogen ignition (following core uncovery)

2.0 AUTOMATIC ACTION

2.1 None

3.0 IMMEDIATE OPERATOR ACTION

3.1 Check Containment Cooling System operating in accordance with SOI 04-1-01-M41-1.

3.2 Refer to EP-3, Containment Control.

3.3 Check the containment temperature on recorders 1M71-R604A&B on 1H13-P870 to determine if the alarm is a localized problem or over all containment problem.

NOTE

Containment individual area temperatures and average temperature may be obtained from PDS computer points identified on 06-OP-1000-D-0001 or from recorders 1M71-R604A and B.

3.4 Verify each individual containment area temperature is less than or equal to 105°F. If an individual temperature exceeds 105°F, refer to TRM 6.7.3.

3.5 Verify Containment average temperature is less than or equal to 95°F. If average temperature exceeds 95°F, refer to Tech. Spec. 3.6.1.5.

3.6 Check the containment and drywell pressure recorders on 1H13-P870 for abnormal trends.

3.7 Check the containment area rad monitors for abnormal readings.

3.8 Dispatch an operator to check locally, if the containment is accessible.

System Plant Chilled Water System (PCW) MPL No. P71

Checksheet Name Remote Operated Valve Lineup Checksheet

Instruction Step 4.1.1c

VALVE NO.	HANDSWITCH NAME	SWITCH NO.	PANEL NO.	REQ POS SW	REQ POS VLV	VLV DEV.	INITIALS 1st	INITIALS 2nd
F306	PCW SPLY HDR TO TURB BLDG	HS-M605	1H13-P870-3C	AUTO	OPEN **			
F307	PCW SPLY HDR TO TURB BLDG	HS-M606	1H13-P870-9C	AUTO	OPEN **			
F148	PCW RTN FM SMPL WTR CLRS/CTMT CLRS	HS-M608	1H13-P870-3C	AUTO	OPEN			
F149	PCW RTN FM SMPL WTR CLRS/CTMT CLRS	HS-M609	1H13-P870-9C	AUTO	OPEN			
F150	PCW SPLY TO SMPL WTR CLRS/CTMT CLRS	HS-M610	1H13-P870-3C	AUTO	OPEN			
F304	PCW RTN HDR FM TURB BLDG	HS-M614	1H13-P870-3C	AUTO	OPEN **			
F305	PCW RTN HDR FM TURB BLDG	HS-M615	1H13-P870-9C	AUTO	OPEN **			
F300	PCW RTN FM DG RM CLRS	HS-M616	1H13-P870-3C	AUTO	OPEN **			
F301	PCW RTN FM DG RM CLRS	HS-M617	1H13-P870-9C	AUTO	OPEN **			
F302	PCW SPLY TO DG RM CLRS	HS-M618	1H13-P870-3C	AUTO	OPEN **			
F303	PCW SPLY TO DG RM CLRS	HS-M619	1H13-P870-9C	AUTO	OPEN **			
F299		HSS-M003	LOCAL	COOLING	CLOSED			
F298					OPEN *			
F019					OPEN *			
FA14A	Steam Tunnel "A" Cooler Inlet Isolation	HS-M620A	1H13-P842	AUTO	OPEN			
FA14B	Steam Tunnel "B" Cooler Inlet Isolation	HS-M620B	1H13-P842	AUTO	OPEN			

* - F019 **AND** F298 are permanently gagged open.

** - These valves are mechanically gagged open by EC# 0026 But an Open signal Must be applied to the control room handswitch following a loss of power **OR** loss of instrument air because the actuator Would be attempting to Close the valve **AND** **IF** a gag fell out these valves Would Close. The actuator Would also be applying force to the gag also **IF** an Open signal is **NOT** present.

Note Exceptions: ______________________________

Performed by: ______________________________ Date __________

Reviewed by: ______________________________ Date __________

(Shift Supervision)

ATTACHMENT 2
SYSTEM RESPONSE

System	**Component/Description**		**Response**		**Restoration**
Control Rod Drive	SCRAM Valves		FAIL OPEN		04-1-01-C11-1, Control Rod Drive Hydraulic System
Condenser Air Removal	1N62-F505A(B), SJAE STEAM SUPPLY VALVEs		FAIL CLOSED		04-1-01-N62-1, Condenser Air Removal
Drywell Cooling	<u>Div 1 Fan Dampers</u> 1M51-F005 1M51-F015 1M51-F003 1M51-F012 1M51-F001 1M51-F010	<u>Div 2 Fan Dampers</u> 1M51-F008 1M51-F017 1M51-F004 1M51-F013 1M51-F002 1M51-F011	<u>Div 1 Dampers</u> FAIL CLOSED	<u>Div 2 Dampers</u> FAIL OPEN	04-1-01-M51-1, Drywell Cooling System
Containment Cooling	<u>Fan Inlet Dampers</u> 1M41-F020A 1M41-F020B 1M41-F020C	<u>Fan Outlet Dampers</u> 1M41-F021A 1M41-F021B 1M41-F021C	FAIL CLOSED		04-1-01-M41-1, Containment Cooling System
Steam Tunnel Cooling	<u>Outside Ctmt Dampers</u> 1T41-F005A 1T41-F005B	<u>Inside Ctmt Dampers</u> 1M41-F029 1M41-F030	FAIL CLOSED		04-1-01-T41-1, Auxiliary Building Ventilation System 04-1-01-M41-1, Containment Cooling System
Combustible Gas Control	<u>Normal DW Vac Relief</u> 1E61-F007 1E61-F020	<u>Ctmt Purge</u> 1E61-F009 1E61-F010 1E61-F012 1E61-F013 1E61-F015	FAIL CLOSED		04-1-01-E61-1, Combustible Gas Control System
Offgas	Instrument Air Supply For Valve Stem Seal Air 1N64-F045, Adsorber Train Bypass 1N64-F051A, Adsorber Inlet Valve 1N64-F051B, Adsorber Inlet Valve 1N64-F051C, Adsorber Inlet Valve 1N64-F051D, Adsorber Inlet Valve 1N64-F060, Offgas Discharge to Vent		Valve Stem Seal to Offgas valves is LOST F045 FAILS CLOSED F051s FAIL OPEN F060 FAILS OPEN		04-1-01-N64-1, Offgas System

LESSON BODY | **NOTES**

Flow instrumentation is provided for the chilled water supply to all of the Drywell coolers for input to the plant computer.

Temperature elements are installed in the inlet and outlet chilled water lines for drywell coolers M51-B001A/B, M51-B002A/B, M51-B005A/B and M51-B006A/B for use in conjunction with the Leak Detection System.

Temperature elements are installed in the chilled water outlet line for the steam tunnel cooler inside containment for input to the plant computer.

The chilled water supply to each cooler coil is provided through a manually controlled motor-operated isolation valve, operated from Control Room Panel P870.

Red and green position indicating lights are provided for the valves on Control Room Panel P870.

Drywell Chilled Water Containment/Drywell Isolation Valves

The Drywell Chilled Water System isolation function is provided by the following Containment and Drywell isolation valves:

Figure 2

- P72-F121, Outboard Containment Chilled Water Supply Isolation Valve
- P72-F124, Outboard Drywell Chilled Water Supply Isolation Valve
- P72-F122, Outboard Containment Chilled Water Return Isolation Valve
- P72-F123, Inboard Containment Chilled Water Return Isolation Valve
- P72-F125, Inboard Drywell Chilled Water Return Isolation Valve
- P72-F126, Outboard Drywell Chilled Water Return Isolation Valve

5.5

EO-5.5

The motor-operated isolation valves can be manually controlled with the associated three-position, keylocked, CLOSE/AUTO/OPEN, spring return to AUTO handswitch on Control Room Panel P870.

Key removable in AUTO position.

Red and green position indicating lights are provided for each valve on Control Room Panel P870.

Auxiliary Building, Containment, Drywell Isolation

11 During a Loss of Instrument Air, the air-operated isolation valves, P71-F148, F149, F150, F300, F301, F302, F303, F304, F305, F306, and F307, will close.

Objective 11

Figure 14

*Valves F300, F301, F302, F303, F304, F305, F306, and F307 are currently gagged open per GG-EC-0026 and will not close but do receive a close signal.

The closure of isolation valves F304, F305, F306, or F307 will trip the Primary and Secondary Chilled Water Pumps, thereby making all of the Plant Chilled Water System unavailable for service.

*These valves are currently gagged open per GG-EC-0026.

Containment Isolation valves P71-F148, F149 and F150 will close on a Group 6 Containment Isolation:

Low Reactor Vessel Level -41.6”

High Drywell Pressure 1.23 psig

These valves cannot be overridden and therefore, PCW to the containment cannot be reestablished until the initiating signal has cleared.

SYSTEM INTERRELATIONSHIPS

♫ & HU Knowledge – Maintain integrated plant knowledge.

Plant Service Water System (P44)

14.1 The Plant Service Water System provides cooling (the heat sink) to the plant chiller condensers.

Objective 14.1

Loss of plant service water results in the loss of chilled water, leading to high area temperatures and, subsequently, a plant shutdown.

Instrument Air System (P53)

14.2 The Instrument Air System supplies air to operate all the air-operated valves in the Chilled Water System, including the isolation valves, the three-way temperature control valves, and the Expansion Tank makeup water valve (F556A).

Objective 14.2

Loss of instrument will cause these valves to fail closed resulting in a complete loss of chilled water.

*Valves F300, F301, F302, F303, F304, F305, F306, and F307 are currently gagged open per GG-EC-0026 and will not close but do receive a close signal.

Examination Outline Cross Reference	Level	RO
295021 Loss of Shutdown Cooling AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : AK3.01 Raising reactor water level.	Tier	1
	Group #	1
	K/A	295021 AK3.01
	Rating	3.3
	Revision	0
Revision Statement:		

Question: 9

The plant is in Mode 3.

Residual Heat Removal (RHR) B is operating in Shutdown Cooling.

Both Reactor Recirc pumps trip and are unable to be restarted.

The primary reason for the contingency to raise reactor water level for this condition is to:

A. raise circulation through the idle recirc loops.

B. establish a flow path from the core to the feedwater annulus.

C. establish a flow path through the SRVs to the Suppression Pool.

D. provide additional coolant mass to the RPV to absorb decay heat.

Answer:	**B**
Explanation: With no Recirc pumps operating, raising reactor water level to above 82 inches allows natural circulation from the core up through the steam separator to the feedwater annulus for core cooling.	
Distracters: ‘A’ is wrong but is plausible because a historically common error is to imagine natural circulation to be through recirc loop piping, but it is actually from core to downcomer to core. ‘D’ is wrong but is plausible because adding water would provide more heat sink, but that is not why the procedure requires raising level.	

'C' is wrong but is plausible because the procedure does require raising level to the main steam lines to establish flow through the SRVs for Alternate Shutdown Cooling, but this is not warranted since RHR B is already operating in shutdown cooling. Alternate Shutdown Cooling would involve pumping from the suppression pool to the reactor using an ECCS pump and returning to the suppression pool through SRVs to establish a coolant loop.

K/A Match

Knowledge of reasons why reactor water level is raised on loss of shutdown cooling.

Technical References:

05-1-02-III-1, Inadequate Decay Heat Removal ONEP, CAUTION before step 3.4.2

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP, objective 17 & 20

Question Source:	**Bank # 865**	2007 NRC (Q9)
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(2)	
Level of Difficulty:	3.0	

PRA Applicability:

RHR is listed as #3 on the System Importance to CDF.

CAUTION

IF adequate core circulation is NOT maintained, THEN Reactor coolant temperature indication is NOT accurate. Natural circulation through core Must be established by raising Reactor water level to allow flow through core AND Feedwater annulus.

3.4.2 **IF** forced circulation (no Reactor Recirculation pumps running) has been lost, **THEN RAISE** Reactor water level to 615" actual (this is 82" above instrument zero).

3.4.3 **IF** RHR Shutdown Cooling **AND** ADHRS capability has been completely lost **AND** RWCU is available, **THEN PERFORM** the following:

a. **RAISE** RWCU reject flow to maximum (approximately 360 gpm)

b. **RAISE** RPV water level using Condensate **AND** CRD as necessary to maintain desired water level.

c. **REFER** to SOI 04-1-01-G33-1 for RWCU Alternate Shutdown Cooling operation.

d. **MONITOR** Reactor coolant temperature **AND** Reactor pressure very closely to determine **IF** this method Will allow **AND** maintain required cooling.

NOTE

Graphs in Attachment I are derived from decay heat analysis for use in a specific refueling outage. **WHEN** used at other times, the graphs provide guidance but May **NOT** be accurate.

e. **REFER** to Attachment I for an estimation of time to 200°F.

Examination Outline Cross Reference	Level	RO
295023 Refueling Accidents 2.4.41 Knowledge of the emergency action level thresholds and classifications.	Tier	1
	Group #	1
	K/A	2.4.41
	Rating	2.9
	Revision	0
Revision Statement:		

Question: 10

Refueling is in progress, a dropped fuel bundle causes rad levels to rise.

The CRS has entered EP-4 due to Offsite Rad Release.

For the CRS to enter EP-4, the Rad Release must be at the level that the declaration of a(n) __________ is expected?

A. Unusual Event

B. Alert

C. Site Area Emergency.

D. General Emergency.

Answer: B
Explanation: Per EP-4 entry conditions "Offsite Rad Release declaration of Alert Expected", so if the CRS has entered EP-4 due to expected or actual offsite rad release then an Alert should be declared. Answer B is correct
Distracters: All distracters are plausible due to being EAL classifications.
K/A Match Knowledge of EOP entry conditions and EAL classifications and thresholds.

Technical References: 02-S-01-40 Rev. 09, EP Technical Bases		
Handouts to be provided to the Applicants during exam: None		
Learning Objective: GLP-OPS-EP4, objective 5		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Entry Conditions

Any Operating Limit exceeded (Table SC-1)	**Differential pressure** above 0 in.	**Offsite Rad Release** declaration of **Alert** expected	**Spent Fuel Pool Level** below **207.49 ft.**	**Spent Fuel Pool Temperature** above **120°F**

Discussion

Entry Conditions

The *Auxiliary Building Control* entry conditions are symptomatic of an emergency or a condition that could become an emergency. Adverse effects on the operability of equipment located in the secondary containment and conditions directly challenging secondary containment integrity were specifically considered in the selection of these entry conditions. The specified parameters are the same as the key parameters controlled by the procedure or are closely related to them. The specified setpoints have been chosen to be operationally significant, unambiguous, readily identifiable, and familiar to operators.

High differential pressure is indicative of a potential loss of secondary containment structural integrity and could result in uncontrolled release of radioactivity to the environment. While differential pressure is used as an entry condition to EP-4, no separate "Differential Pressure" branch is included in the procedure. Since the margin between the normal differential and the structural building strength is small, the only practical control methods are those specified for controlling secondary containment temperature and radiation level:

Operating area coolers

Operating SGTS

Isolating breaks

High area temperatures are indications of fires and of breaks into the secondary containment. Either may jeopardize the habitability of the secondary containment and the operability of the equipment in it.

The high area radiation, high HVAC exhaust radiation, and high area water level entry conditions provide additional indications of breaks into the secondary containment. High radiation levels may limit access to the secondary containment and to the equipment in it. Water collecting in secondary containment spaces could flood equipment necessary for safe operation of the plant.

Spent fuel pool temperature above the high temperature alarm setpoint is indicative of a loss of spent fuel pool cooling. Continued heatup of the spent fuel pool may result in release of volatile fission products, increased secondary containment humidity, and eventual loss of spent fuel pool inventory due to boiling.

Discussion (continued)

Spent fuel pool level below the low level alarm setpoint is indicative of a loss of spent fuel pool inventory due to leakage or boiling. Continued loss of inventory may result in increased area radiation levels and eventual uncovery of spent fuel bundles.

An offsite radioactivity release approaching the emergency plan “Alert” level is an indication of a primary system discharge outside the primary and secondary containments. The Alert level is sufficiently high that it is not expected to occur during normal plant operations but low enough that the condition does not pose an immediate threat to the health and safety of the public

Table SC-1 lists the specific areas and associated action levels for parameters referenced in the entry condition.

Auxiliary Building Control must be entered at the beginning of the flowchart whenever any entry condition occurs, or clears and then occurs again, even if the procedure is already being executed.

Examination Outline Cross Reference	Level	RO
295024 High Drywell Pressure EA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: EA1.05 RPS	Tier	1
	Group #	1
	K/A	295024 EA1.05
	Rating	3.9
	Revision	0
Revision Statement:		

Question: 11

A RPV leak has occurred

A RO is monitoring automatic actions with rising drywell pressure.

Which of the following actions coincides with a RPS reactor scram signal?

A. Div 2 LSS actuation

B. HPCS system initiation

C. Div 1 Diesel Generator auto start

D. P53-F001, INST AIR CTMT ISOL, auto close

Answer: D
Explanation: With rising drywell pressure a reactor scram will automatically occur at 1.23 psig Drywell Pressure. A coinciding action will be the automatically closure of P53-F001, INST AIR CTMT ISOL valve. 'D' is correct.
Distracters: All other distractors are wrong but plausible because all auto initiate at 1.39 psig Drywell pressure
K/A Match Ability to monitor diverse indications to verify actions that have occurred. The Reactor Scram can be verified with an auto closure of Containment Isolation valves.

Technical References:

05-1-02-III-5 Rev 52, Automatic Isolation ONEP
05-1-02-I-1 Rev 132, Reactor Scram ONEP

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-C7100 Rev 16, objective 9
GLP-OPS-M7100 Rev 15, objective 8

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

AUTOMATIC ISOLATION CHECKLIST (Continued)

Containment and Drywell Isolation Group 6

Isolation Signals

- Reactor Vessel Water Level Low (Level 2): -41.6"
- Drywell Pressure High: 1.23 psig
- Drywell Pressure High: 1.39 psig (Division 3 only)

Valves Closed **Division 1**	**Panel 1H13-P870-3C**
P11-F130, REFUEL WTR XFER PMP SUCT FM SUPP POOL	(Normally Open)
P60-F009, SPCU RTN FM CNDS PC FLTRS	(Normally Closed)
P71-F150, PCW SPLY TO SMPL WTR CLRS/CTMT CLRS	(Normally Open)
P71-F148, PCW RTN FM SMPL WTR CLRS/CTMT CLRS	(Normally Open)
P72-F121, DWCW SPLY TO CTMT	(Normally Open)
P72-F122, DWCW RTN FM CTMT	(Normally Open)
P72-F125, DWCW RTN FM DRWL	(Normally Open)
P11-F075, CST WTR SPLY HDR TO CTMT	(Normally Open)
G36-F101, RWCU BKW RCV TK XFER TO RADWST	(Normally Open)
P45-F062, CTMT FLOOR DR SMP DISCH	(Normally Open)
P45-F003, DRWL FLOOR DR SMP DISCH	(Normally Open)
P21-F017, MU STR SPLY HDR TO CTMT	(Normally Open)
P45-F273, AUX BLDG XFER TANKS CTMT ISOL VLV	(Normally Closed)
P45-F068, CTMT EQUIP DR SMP DISCH	(Normally Open)
P45-F009, DRWL EQUIP DR SUMP DISCH	(Normally Open)
P53-F001, INST AIR SUPPLY HDR TO CTMT	(Normally Open)
P53-F003, INST AIR SPLY HDR	(Normally Open)
P52-F105, SVC AIR SPLY HDR TO CTMT	(Normally Open)
P45-F099, CTMT CHEMWST SMP DISCH	(Normally Open)
P45-F096, DRWL CHEM WST SMP DISCH	(Normally Closed)

IF in-service SJAE pressure controller N62-R010A(B) is being controlled in manual, **THEN MONITOR** SJAE SPLY STM PRESS on 1H13-P680 **AND DISPATCH** operator to controller to **MAINTAIN** SJAE pressure to between 120 **AND** 135 psig.

NOTIFY Chemistry to perform 06-CH-1D17-V-0061 following the Reactor Scram.

NOTIFY Load Dispatcher of plant condition.

WHEN Reactor water level is normal **AND** stable, Reactor pressure is stable, **AND** all scram signals are reset,
THEN PERFORM the following:

RESET Lo-Lo-Set.

REFER to IOI 03-1-01-4, Scram Recovery.

REFER to EN-OP-117 **AND PERFORM** Transient Snap-Shot Assessment **IF** conditions requiring an assessment have been met.

4.0 SYMPTOMS

4.1 Reactor Scram trip annunciates

4.2 Eight white SCRAM SOLENOID VALVE lights are out

4.3 Indication of rods inserted

4.4 Reactor power rapidly decreasing

4.4.1 APRM recorders indicate a one-to-two-decade instantaneous drop.

4.4.2 IRM recorders indicate a decrease **IF** recorder is selected to IRMs **AND IF** IRMs have been ranged up **AND** inserted.

4.4.3 SRM recorders indicate a one-to-two-decade instantaneous drop.

4.5 One **OR** more of the following conditions exist:

4.5.1 CRD Instrument Volume High Water Level trip (60% of full scale, **OR** 64" for Float switches)

4.5.2 Main Steam Line Isolation valves closure (6% closed)

4.5.3 Reactor Vessel high pressure (1065 psig)

4.5.4 Reactor Vessel low water level (11.4 inches)

4.5.5 Reactor Vessel high water level (53.5 inches)

4.5.6 Turbine Stop valve closure trip (40 psig)

4.5.7 Turbine Control valve fast closure trip (46 psig)

4.5.8 Intermediate Range Monitor Neutron Flux-High (120/125 divisions of full scale)

4.5.9 Average Power Range Monitor Neutron Flux-High **AND NOT** in RUN mode (18% of Rated Thermal Power)

4.5.10 Average Power Range Monitor thermal power trip in RUN mode (Setpoints in accordance with applicable TRM Figures TR 3.3.1.1-1 **OR** Figure TR 3.3.1.1-2) (Maximum is 111%)

4.5.11 Average Power Ranger Monitor Neutron Flux-High in RUN mode (118% of RTP)

4.5.12 Drywell high pressure (1.23 psig)

4.5.13 Manual scram

5.0 AUTOMATIC ACTIONS

5.1 All control rods insert to Position 00, Backup Scram valves energize, **AND** Scram Discharge Volume Vent **AND** Drain valves close.

5.2 Reactor Feedwater Level Control automatically enters single element mode of level control after reactor level is less than 11.4 inches

5.3 Reactor Feedwater Level Controller automatically raises its setpoint to approximately 54" for **WHICHEVER** of the following occurs first:

5.3.1 10 seconds has elapsed
OR
Level is greater than 12.4 inches

5.3.2 **THEN** set point is automatically reduced to approximately 4".

5.3.3 RX LVL MASTER CONT Must be raised back to 36 inches by operator action, **THEN** SETPOINT SETDOWN RESET pushbutton is depressed to clear status light.

5.4 Recirculation pumps shift to slow speed on any of the following signals:

5.4.1 Less than 3.0×10^6 lbm/hr Feedwater flow (15-second time delay)

5.4.2 +11.4 reactor water level

5.4.3 Main Steam Line to Recirc Pump Suction differential temperature less than 7.4°F (15-second time delay)

5.4.4 Turbine trip

Examination Outline Cross Reference	Level	RO
295025 High Reactor Pressure EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : EK1.03 Safety/relief valve tailpipe temperature/pressure relationships	Tier	1
	Group #	1
	K/A	295025 EK1.03
	Rating	3.6
	Revision	0
Revision Statement:		

Dec 2017 NRC Exam Q# 12

Question: 12

HANDOUT PROVIDED

A transient occurred causing Reactor pressure to rise just high enough to initiate the Lo-Lo Set subsystem of SRVs.

(1) How many SRVs initially opened?

(2) What is the peak SRV **tail pipe** temperature?

A. (1) One
(2) 411°F

B. (1) Two
(2) 411°F

C. (1) One
(2) 558°F

D. (1) Two
(2) 558°F

Answer:	**B**
Explanation:	

Lo-Lo set system initiates at 1103 psig. Convert 1103 psig to psia by adding 14.7, which is approximately 1118 psia.

Once initiated one SRV lowers its opening setpoint to 1033 psig and another lowers its opening setpoint to 1073, therefore 2 SRVs will initially be open. The other SRVs open at 1113 psig or 1123 psig.

The point of initiation is 1103 psig or 1118 psia reactor pressure; however, tailpipe pressures average about 25% of reactor steam dome pressure for a full valve lift at rated conditions.

1118 X 25% = 279.5 psia.

Saturated temperature for 279.5 psia is 411°F.

'B' is correct

Distracters:

'A' is wrong. Two SRVs will be open, not one. Plausible if the applicant doesn't remember that when LO-LO-set initiates at 1103 psig the opening setpoint of 2 SRVs lowers to 1033 and 1073, therefore, 2 SRVs will immediately open.

'C' is wrong. Temperature is for 1118 psia and two SRVs will be open, not one. Plausible if the applicant doesn't remember that when LO-LO-set initiates at 1103 psig the opening setpoint of 2 SRVs lowers to 1033 and 1073, therefore, 2 SRVs will immediately open. Also, if the applicant doesn't recall the process of steam going through a SRV at rated conditions.

'D' is wrong. Temperature is for 1118 psia. Plausible if the applicant doesn't recall the process of steam going through a SRV at rated conditions.

K/A Match

This question requires the applicant to have the knowledge of Safety/relief valve tailpipe temperature/pressure Relationships and the operational implications of the indication.

Technical References:

GLP-OPS-E2202, Rev. 10, Automatic Depressurization System (ADS) Lesson Plan pages 21 and 34 of 45

Handouts to be provided to the Applicants during exam:

Steam Tables

Learning Objective:

GLP-OPS-E2202, OBJ 18.0

Question Source:	**Bank # 1350**	**Dec 2017 NRC Exam Q# 12**
(note changes and attach parent)	**Modified Bank #**	

	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(14)	
Level of Difficulty:	4.0	
PRA Applicability:		
None.		

This effectively changes the opening setpoint of two SRVs and the closing setpoint of six SRVs in the relief mode	**11.1** Low-Low Set (LLS) is a feature of the Relief Mode that meets the containment design basis which assumes that only one SRV reopens and cycles after the initial pressure transient from a full MSIV closure isolation event. In LLS, the lowest set point SRV's opening and closing set pressures are farther apart than those for its normal Relief Mode resulting in fewer actuations to remove decay heat following an MSIV closure isolation. LLS limits cyclic transients on the SRVs, SRV tailpipes, quenchers and the Containment structure around the quenchers.
	10.4, 12.1 Low-Low Set initiates automatically when RPV pressure reaches SRV F051D's normal lift setpoint of 1103 psig.
	For each mode of operation of the SRVs, state the number of valves which open and the associated opening setpoints. (11.2) Describe the interlocks associated with the Low-Low Set Logic. (12.1) Statewhich SRVs are operated in Low-Low Set and ADS modes. (13.0)
	11.2,12.1, 13.0 Once initiated, six SRVs are capable of operating in the Low-Low Set mode with the following adjusted opening and closing setpoints:
When LLS is initiated, F051B should also open (i.e., 1103 psig initiated LLS and F051B opening is adjusted down to 1073).	**11.2, 12.1, 13.0** One SRV, F051D, lifts at 1033 psig and blows down to 926 psig. **11.2, 12.1, 13.0** A second SRV, F051B, lifts at 1073 psig and blows down to 936 psig.
F051B and these four SRVs are in the group of ten SRVs that have a normal relief setpoint of 1113 psig.	**11.1, 12.1, 13.0** The following four SRVs lift at 1113 psig and blow down to 946 psig: F047D F047G F051A F051F
Operate at normal relief setpoints.	**11.2** The following SRVs, not impacted by the initiation of Low-Low Set, open at 1113 psig and re-close at 1013 psig: F047A F047C F047H F047L F051K
These nine are the eight F041 valves plus F051C.	**11.2** The remaining nine SRVs, likewise not impacted by the initiation of Low-Low Set, open at 1123 psig and re-close at 1023 psig.

Tailpipe temperature

Figure 9

18.0 Tailpipe temperature for each of the 20 SRVs is indicated on recorder B21-R614 on panel 1H13-P614.

18.0 An annunciator is actuated on P601 when any tailpipe temperature exceeds 155°F.

Alarms must be acknowledged and reset at the recorder.

The indicated temperature for a wide-open SRV is a function of tailpipe steam pressure.

Tailpipe pressure is a function of Reactor steam dome pressure, Suppression Pool level, length of tailpipe run, etc.

Tailpipe pressure averages about 25% of Reactor steam dome pressure for a full valve lift at rated conditions (100% power, Reactor steam dome pressure = 1025 psig).

Monitoring; Do an Mollier Diagram explanation for a throttling process

For a pressure/temperature relationship at essentially saturation conditions, if an SRV lifted at rated conditions you could expect to see an initial tailpipe temperature that is T_{SAT} for 256 psig, or 408°F.

The actual temperature will be less due to ambient losses and steam condensation along the tailpipe run.

For off-rated conditions (Reactor pressure between 0 and 1025 psig), the expected tailpipe temperature remains a function of T_{SAT} for tailpipe pressures that are less than 25% of the corresponding Reactor steam dome pressure.

As Reactor pressure lowers, mass flowrate through an open SRV decreases; thus tailpipe pressure as a function of Reactor pressure decreases non-linearly.

For "weeping" SRVs (valves that are only open just enough to pass a small amount of steam), tailpipe pressure remains at essentially atmospheric.

Considering the enthalpy of the 1025 psig steam on the upstream side of the valve and the downstream pressure of about 15 psia, tailpipe temperature should ideally see slightly superheated steam at ≈ 248°F.

Actual long-term temperature will be less due to ambient heat losses along the tailpipe run.

Examination Outline Cross Reference	Level	RO
295026 Suppression Pool High Water Temperature EK3. Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: EK3.04 †SBLC injection	Tier	1
	Group #	1
	K/A	295026 EK3.04
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 13

An ATWS has occurred

Reactor power is 2.5%

(1) When will the CRS call for SBLC injection?

(2) Which of the following describes the reason for SBLC injection?

A. (1) Before 110 F
(2) Maintain Containment temperature and pressure limits

B. (1) Before 110 F
(2) Prevent Large scale power instabilities leading to core damage

C. (1) After 110 F
(2) Maintain Containment temperature and pressure limits

D. (1) After 110 F
(2) Prevent Large scale power instabilities leading to core damage

Answer: A
Explanation: Per 02-S-01-40, EP Technical Bases, Attachment V page 37 of 58 states the following: “If reactor power is at or below 5%, thermal-hydraulic instabilities are not of significant concern since the core boiling boundary will be relatively high and the core void content will be relatively low. Some oscillations may still occur, but large-scale instabilities leading to core damage are not expected. Boron injection then need not be injected unless primary containment integrity is jeopardized.” ‘A’ is correct

Distracters:

‘B’ is wrong. At <5% the bases states “large-scale instabilities leading to core damage are not expected.”

‘C’ is wrong. EP-3 states that before suppression pool 110 inject with boron not after, plausible due to the EPs uses before and after setpoints, the student must know its before setpoint.

‘D’ is wrong. At <5% the bases states “large-scale instabilities leading to core damage are not expected.” AND EP-3 states that before suppression pool 110 inject with boron not after, plausible due to the EPs uses before and after setpoints, the student must know its before setpoint.

K/A Match

This question requires the applicant to have the knowledge of the reason for injecting SBLC during an ATWS.

Technical References:

02-S-01-40, EP Technical Bases Rev 9, Attachment V, Page 37 of 58

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

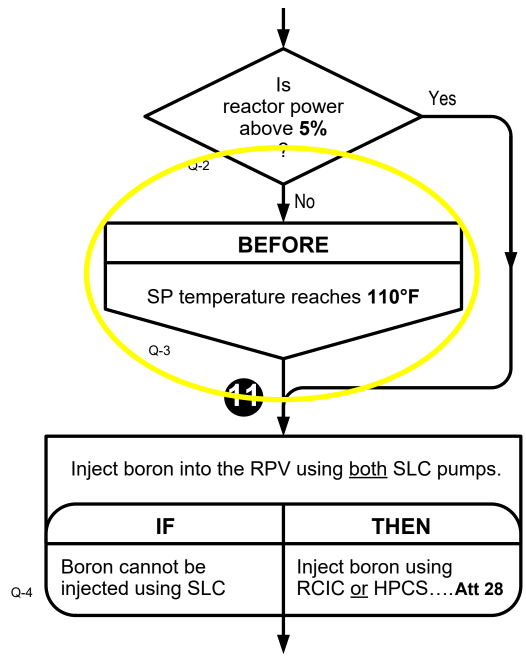
GLP-OPS-EP02A, OBJ 7.0

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Steps Q-2 through Q-4



Discussion

Steps Q-2 through Q-9 provide directions for injecting boron into the RPV. These actions are performed in parallel with Step Q-1, which identifies alternate methods of inserting control rods.

Is reactor power above 5%?

If reactor power remains above the APRM downscale setpoint (5%) following multiple attempts to scram the reactor (EP-2 Step 1, EP-2A Step 1), boron injection is initiated immediately to preclude power oscillations, avoid challenges to primary containment temperature and pressure limits, and ensure that the plant remains in a controlled state.

If reactor power is at or below 5%, thermal-hydraulic instabilities are not of significant concern since the core boiling boundary will be relatively high and the core void content will be relatively low. Some oscillations may still occur, but large-scale instabilities leading to core damage are not expected. Boron injection then need not be injected unless primary containment integrity is jeopardized.

If the MSIVs close under ATWS conditions, steam may be discharged through the SRVs, causing suppression pool temperature and primary containment pressure to increase. The requirement for boron injection when reactor power is at or below 5% is thus based upon containment temperature and pressure limits.

Examination Outline Cross Reference	Level	RO
295027 High Containment Temperature (Mark III Containment Only) EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) : EK1.03 Containment integrity: Mark-III	Tier	1
	Group #	1
	K/A	295027 EK1.03
	Rating	3.8
	Revision	0
Revision Statement:		

Question: 14

EP-3, Primary Containment Control, is being implemented.

Containment temperature is continuing to rise and is nearing 185°F.

Per the EP Technical Bases, what is the significance of nearing 185°F Containment temperature?

Containment temperature is nearing the point beyond which…

A. containment spray initiation is not allowed.

B. the Suppression Pool pressure suppression function is impaired.

C. Fuel Zone RPV water level instruments will no longer be reliable.

D. containment damage or challenge to equipment operability may occur.

Answer: D
Explanation: Per 02-S-01-40, EP Technical Bases, Attachment VI Per EP-3 Bases, page 18 of 37, this is the containment design temperature…i.e., where containment damage or challenge to equipment operability may occur. If containment temperature cannot be restored and maintained below 185°F, the containment design temperature, emergency RPV depressurization is required. This strategy limits the release of energy into the containment, thus minimizing further containment heatup, and ensures that the RPV is depressurized before temperature rises high enough to damage the containment or challenge equipment operability limits. Consistent with the definition of “cannot be restored and maintained” in

Section 3.0, a decision that containment temperature cannot be restored and maintained below 185°F can be made may be made before, when, or after temperature actually reaches this value.

'D' is correct

Distracters:

'A' is wrong. Spraying the containment is based on a CTMT temp vs. CTMT pressure and is allowed at and above 185°F.

'B' is wrong. Pressure Suppression function is based on containment pressure not temperature.

'C' is wrong. Fuel Zone level instruments are not listed in Caution 1 for reliable. Figure 1 starts at 200°F Containment temp

K/A Match

This question requires the applicant to have the knowledge of the implications of high containment temperature and the effects.

Technical References:

02-S-01-40, EP Technical Bases Rev 9, Attachment VI, Page 18 of 37

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3, OBJ 6.0

Question Source:	**Bank # 930**	NRC 2008 (Q62)
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Discussion (continued)

Core cooling takes precedence over spraying the containment in this step. Only RHR pumps not needed for core cooling may be placed in the containment spray mode. It is permissible, however, to alternate RHR pumps between modes if *continuous* operation in the LPCI mode is not required.

Terminate CTMT sprays

Containment sprays must be terminated by the time containment pressure decreases to 0 psig to ensure that pressure is not reduced below the negative design value. Terminating sprays “before” 0 psig permits use of the sprays for fission product scrubbing at low pressures or if the containment has failed, yet still avoids challenging containment integrity.

Consistent with the definition of “before,” in Section 3.0, action to terminate spray flow may be initiated at any containment pressure above 0 psig. The optimum timing of the action is event-specific:

If pressure is dropping rapidly, it may be necessary to initiate action at a significantly higher pressure to maintain containment pressure above 0 psig.

Reducing drywell pressure below the scram setpoint will clear the scram logic and maximize the margin to containment pressure limits.

If the containment has failed or if primary containment venting is anticipated, it may be desirable to continue spray operation at low pressures to scrub the containment atmosphere.

Note that while *operation* of containment sprays is permitted down to pressures approaching 0 psig, the CSIPL may prohibit spray *initiation* at low pressures.

CTMT temperature cannot be restored and maintained below 185°F

If containment temperature cannot be restored and maintained below 185°F, the containment design temperature, emergency RPV depressurization is required. This strategy limits the release of energy into the containment, thus minimizing further containment heatup, and ensures that the RPV is depressurized before temperature rises high enough to damage the containment or challenge equipment operability limits. Consistent with the definition of “cannot be restored and maintained” in Section 3.0, a decision that containment temperature cannot be restored and maintained below 185°F can be made may be made before, when, or after temperature actually reaches this value.

Note that while the suppression pool temperature design limit has been increased from 185°F to 210°F, containment air temperature design limit remains at 185°F. Step CNT-6 actions are based on containment temperature; therefore, 185°F is the proper value.

Examination Outline Cross Reference	Level	RO
295028 High Drywell Temperature EK3. Knowledge of the reasons for the following responses as they apply to (EMERGENCY): EK3.04 Increased drywell cooling	Tier	1
	Group #	1
	K/A	295028 EK3.04
	Rating	3.6
	Revision	0
Revision Statement:		

NRC May 2017 (Q14)

Question: 15

What is the reason for operating all drywell cooling when Drywell Temperature exceeds the **EP-3 entry condition**?

A. To maintain Drywell temperature less than the ADS qualification temperature where a Reactor scram is required

B. To maintain Drywell temperature less than the Drywell design temperature limit where an Emergency Depressurization is required

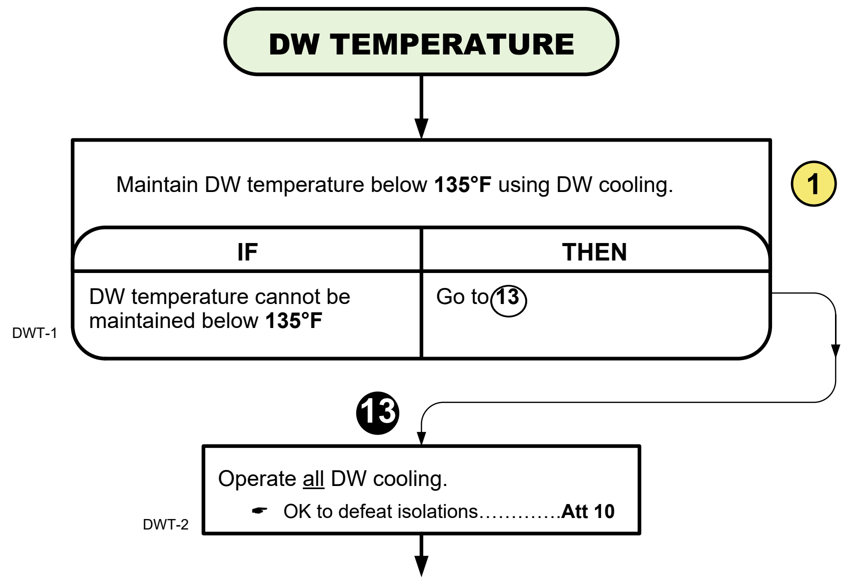
C. To prevent exceeding the Drywell temperature where the ADS/SRVs are declared INOP

D. To reduce and maintain Drywell temperature less than the Drywell temperature Tech Spec LCO.

Answer: D
Explanation: The Drywell temp. EP-3 entry condition is 135°F, 02-S-01-40, EP Technical Bases, states “As long as drywell temperature remains below 135°F, the higher of the drywell temperature LCO and the maximum normal operating drywell temp., no further operator action need be taken in the DW temp. branch. ‘D’ is correct
Distracters: ‘A’ is wrong – same as B, by maxing drywell cooling you should maintain below the ADS qualification temperature (355°F), an ED is required if this setpoint cannot be maintained. Not just a reactor scram. ‘B’ is wrong – by starting all available drywell cooling could be to prevent exceeding the design temperature limit (330°F), only a scram is required not an ED.

'C' is wrong – the ADS qualification temp is 355°F not 135°F		
K/A Match This question requires the applicant to have the knowledge of the reason for increasing cooling.		
Technical References: 02-S-01-40, EP Technical Bases Rev 9, Attachment VI, Page 10 of 37		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-EP3, OBJ 6.0		
Question Source:	**Bank # 1263**	**NRC May 2017 (Q14)**
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Steps DWT-1 and DWT-2



Discussion

Maintain DW temperature below 135°F using DW cooling

The first method used to control drywell temperature is that used during normal plant operations—drywell cooling. The first step in the DW Temperature branch thus provides a smooth transition from general plant procedures to emergency operating procedures and ensures that normal procedures are tried before more complex actions.

Operate all DW cooling

As long as drywell temperature remains below 135°F, the higher of the drywell temperature LCO and the maximum normal operating drywell temperature, no further operator action need be taken in the DW Temperature branch. If drywell temperature *cannot* be maintained below 135°F, the procedure becomes more prescriptive and requires that all drywell cooling be used. Any fan not already running should be started and cooling flow should be maximized. Interlocks may be defeated, if necessary, to establish cooling water flow. It is understood that any drywell cooling that is unavailable regardless of the reason would not be used.

Consistent with the definition of “cannot be maintained” in Section 3.0, a determination that drywell temperature cannot be maintained below 135°F can be made before temperature actually *reaches* this value.

Examination Outline Cross Reference	Level	RO
295030 Low Suppression Pool Water Level EK2. Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: EK2.08 SRV discharge submergence	Tier	1
	Group #	1
	K/A	295030 EK2.08
	Rating	3.5
	Revision	0
Revision Statement:		

Question: 16

Which of the following Suppression pool levels is required to ensure SRV discharge submergence prior to emergency depressurization?

A. 10.5 ft.

B. 14.5 ft.

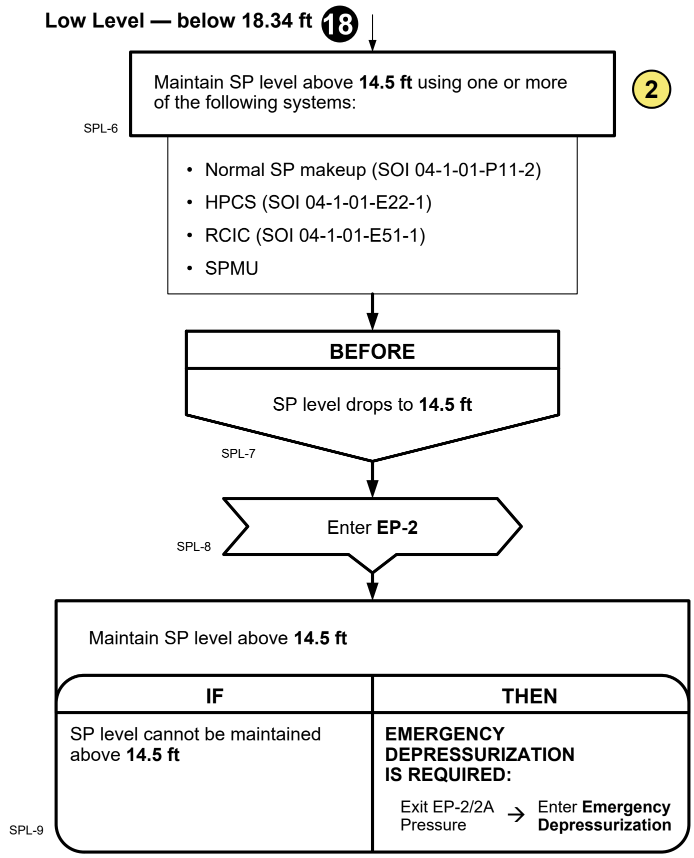
C. 14.25 ft.

D. 17.5 ft.

Answer: A
Explanation: The suppression pool level to Emergency Depress is 10.5 ft., 02-S-01-40, EP Technical Bases, SRVs may be opened only if suppression pool water level is above 10.5 ft., the bottom of the suppression pool water level indication range. If suppression pool water level were low offscale, the actual water level could be below the top of the SRV discharge devices. Opening the SRVs with the discharge devices exposed would pass steam directly into the containment airspace, bypassing the suppression pool. 'A' is correct
Distracters: 'B' is wrong – The specified suppression pool water level of 14.5 ft. corresponds to an elevation 2 ft. above the top of the horizontal vents. 'C' is wrong – At GGNS, all suppression pool temperature elements are located at or above 14.25 ft. 'D' is wrong – This is the suppression pool level level setpoint for automatic initiation of Suppression Pool Makeup system
K/A Match

This question requires the applicant to have the knowledge of the location of the SRV spargers in relation to Suppression Pool Level.		
Technical References: 02-S-01-40, EP Technical Bases Rev 9, Attachment IV, Page 48 of 53		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-EP3, OBJ 6.0		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Steps SPL-6 through SPL-9



Discussion

Restore and maintain SP level above 14.5 ft.

If suppression pool water level cannot be maintained above the low end of the control band defined in Step SPL-1, an alternate control band of "above 14.5 ft." is defined. Additional methods for adding water to maintain the alternate band are authorized, including initiation of SPMU. Dumping the upper pools adds a significant volume of water to the suppression pool and may delay the time at which more extreme actions are required if the loss of water from the suppression pool cannot be terminated. If emergency depressurization is later required, the additional water will increase the heat capacity of the pool and thus minimize containment pressure following the blowdown.

The specified suppression pool water level of 14.5 ft. corresponds to an elevation 2 ft. above the top of the horizontal vents. If a loss of coolant accident were to occur with suppression pool water level below this elevation, steam discharged through the horizontal vents may not be completely condensed and containment pressure could exceed structural limits.

Discussion (continued)

Before SP level drops to 14.5 ft. / Enter EP-2

If the loss of water from the suppression pool cannot be terminated and suppression pool water level continues to decrease, emergency RPV depressurization will ultimately be required by the Step SPL-9 contingency action. The reactor is scrammed through concurrent entry of EP-2 “before” emergency RPV depressurization is initiated to shut down the reactor and reduce the steam generation rate. The scram is prescribed indirectly, through entry of EP-2, because:

EP-2 coordinates control of RPV water level and pressure following the scram.

EP-2 branches to EP-2A, *ATWS RPV Control*, to provide appropriate guidance if all rods are not inserted by the scram.

An override in the Pressure branch EP-2 allows rapid depressurization through the main turbine bypass valves in anticipation of a possible emergency depressurization.

Refer to Section 3.0 for a discussion of the term “before.”

Caution #2

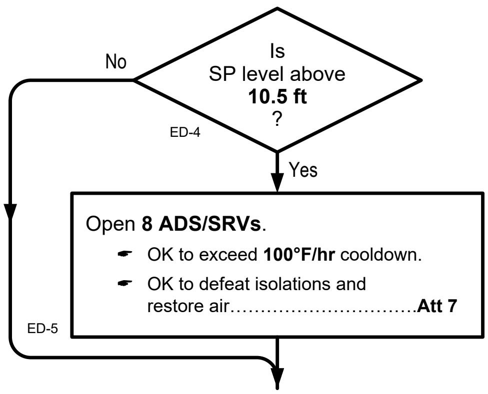
At GGNS, all suppression pool temperature elements are located at or above 14.25 ft., relatively close to the suppression pool surface. Caution #2 provides a reminder that alternative or indirect indications must be used to determine suppression pool temperature if suppression pool water level drops below the elevation of the temperature elements. (Refer to the discussion of Caution #2 in Section 4.0.)

SP level cannot be maintained above 14.5 ft.

Suppression pool water must be maintained above 14.56 ft. to ensure that steam discharged through the horizontal vents following a primary system break will be adequately condensed. If a primary system break were to occur with suppression pool water level below this elevation, pressure suppression capability would be unavailable and primary containment pressure could exceed structural limits.

If suppression pool water level *cannot* be maintained above 14.5 ft., emergency RPV depressurization is required since the RPV is not permitted to remain at pressure if pressure suppression capability is unavailable. Consistent with the definition of “cannot be maintained” in Section 3.0, a decision that suppression pool water level cannot be maintained above 14.5 ft can be made before level actually reaches this value.

Steps ED-4 and ED-5



Discussion

RPV depressurization can most easily and rapidly be performed by opening SRVs. Those associated with the ADS function are used first, consistent with automatic system functions. The ADS SRVs are generally considered the most reliable and are arranged to provide uniform distribution of the heat load around the suppression pool. If one or more ADS valves cannot be opened, other SRVs are opened until the total number of open SRVs equals the number of SRVs dedicated to ADS function.

The direction to “Open 8 ADS/SRVs” requires manual action, even if the valves are already open on high pressure; automatic valve operation in the relief mode does not accomplish the objective of this step, even if low-low set has actuated. Direct manual control must be established to ensure that the valves remain open as RPV pressure decreases and will reopen if necessary to prevent RPV repressurization.

SRVs may be opened only if suppression pool water level is above 10.5 ft., the bottom of the suppression pool water level indication range. If suppression pool water level were low offscale, the actual water level could be below the top of the SRV discharge devices. Opening the SRVs with the discharge devices exposed would pass steam directly into the containment airspace, bypassing the suppression pool. This direct discharge of steam could damage equipment needed for the safe shutdown of the plant and result in excessive containment pressures. If suppression pool water level drops below 10.5 ft. after the RPV is depressurized, however, the SRVs need not be reclosed. Once RPV depressurization has been completed, the energy addition to the primary containment through the SRVs will be within the capacity of the containment vent, even if the SRV discharges are uncovered. Maintaining the RPV depressurized then takes priority and primary containment pressure may be controlled by venting.

Defeating all isolation interlocks for the SRV pneumatic supply and restoring pneumatics is appropriate if the SRVs are unavailable due to the loss of the pneumatic supply. These actions may be performed prior to or after the system isolation dependent upon time, manpower, and the need or anticipated need for SRV use.

Examination Outline Cross Reference	Level	RO
295031 Reactor Low Water Level 2.4.3 Ability to identify post-accident instrumentation.	Tier	1
	Group #	1
	K/A	2950310 2.4.3
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 17

Which of the following is a "Post Accident" Reactor Vessel Water Level instrumentation indication in the Main Control Room?

A. H13-P680 Narrow Range

B. H13-P680 Upset Range

C. H13-P601 Fuel Zone Range

D. H13-P601 Shutdown Range

Answer: C
Explanation: Per Tech Spec 3.3.3.1 Post Accident Monitoring Instrumentation and 06-OP-1C61-M-0001, Remote Shutdown Panel and Accident Monitoring Instrumentation Channel Check, Fuel Zone level instrumentation is considered Post Accident. 'C' is correct
Distracters: 'A' is wrong – Narrow Range level instruments are listed in the Tech Specs but for RFPT trip. 'B' and 'D' are wrong – not listed in Tech Specs. Plausible due to being a range of level instrumentation and Upset is used for Feedwater level control system, Shutdown is used for high water levels.
K/A Match This question requires the applicant to have the knowledge of the location and which level instrument is considered to be post accident.
Technical References:

Tech Specs 3.3.3.1 and 06-OP-1C61-M-0001 Rev 107		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-TS001, OBJ 6.0		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Vessel Pressure	2	E
2. Reactor Vessel Water Level — Wide Range	2	E
3. Reactor Vessel Water Level — Fuel Zone	2	E
4. Suppression Pool Water Level	2	E
5. Suppression Pool Sector Water Temperature	2(c)	E
6. Drywell Pressure	2	E
7. Drywell Temperature	2	E
8. CRD Cavity Temperature	2	E
9. Primary Containment Pressure — Wide Range	2	E
10. Primary Containment Pressure — Narrow Range	2	E
11. Primary Containment Air Temperature	2	E
12. Primary Containment Area Radiation	2	F
13. Drywell Area Radiation	2	F
14. Deleted		
15. Deleted		
16. Penetration Flow Path, Automatic PCIV Position	2 per penetration flow path (a)(b)	E

(a) Not required for isolation valves whose associated penetration flow path is isolated.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) Monitoring each of six sectors.

DATA SHEET I (continued)

Instruction Section 1.0 Step 4.d.1

PARAMETER	PANEL NUMBER INSTRUMENT	READING	BAND	APPLICABLE TECH SPEC	*INITIALS
Rx Vessel Level (Remote Shutdown Panel)	1H22-P150 1C61-LI-R400A		10 Inches	SR 3.3.3.2.1 TR3.3.3.2-1.A.2	
	1H22-P151 1C61-LI-R400B			SR 3.3.3.2.1 TR3.3.3.2-1.A.2	

Instruction Section 1.0 Step 4.d.2

PARAMETER	PANEL NUMBER INSTRUMENT	READING	BAND	APPLICABLE TECH SPEC	*INITIALS
Rx Vessel Level Fuel Zone Rg (Control Room)	1H13-P601-20B 1B21-LR-R615A Green Pen		10 Inches	SR 3.3.3.1.1 3.3.3.1-1.3	
	1H13-P601-17B 1B21-LR-R615B Green Pen			SR 3.3.3.1.1 3.3.3.1-1.3	

Instruction Section 1.0 Step 4.d.2

PARAMETER	PANEL NUMBER INSTRUMENT	READING	BAND	APPLICABLE TECH SPEC	*INITIALS
Rx Vessel Level (Control Room)	1H13-P601-20B 1B21-UR-R623A		10 Inches	SR 3.3.3.1.1 3.3.3.1-1.2	
	1H13-P601-17B 1B21-UR-R623B			SR 3.3.3.1.1 3.3.3.1-1.2	

Examination Outline Cross Reference	Level	RO
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 : EA2.03 SBLC tank level	Tier	1
	Group #	1
	K/A	295037 EA2.03
	Rating	4.3
	Revision	0
Revision Statement:		

Question: 18

An ATWS has occurred.

Standby Liquid Control has been initiated.

Initial SLC storage tank level was 4800 gallons.

The first SLC pump is to be secured per EP-2A in __________ minutes?

A. 24 minutes

B. 34 minutes

C. 68 minutes

D. 116.5 minutes

Answer: B
Explanation: The SLC pumps are designed for 41.2 gpm and when SLC is initiated both pumps are started, therefore 82.4 gpm is being injected into the vessel. When SLC storage tank reaches 2000 gallons then one pump is to be shutdown per EP-2A. So 4800 gal – 2000 gal = 2800 gals. 2800 gals / 82.4 gpm = 33.98 minutes. 'B' is correct
Distracters:

‘A’ is wrong – Plausible due to if the student thinks that 2000 gal. must be pumped instead of 2000 gal indicated.

‘C’ is wrong – Plausible due to the student must remember that both pumps will be started, this is the time for one pump running.

‘D’ is wrong - Plausible due to this would be the time for the entire tank to be pumped using only one pump.

K/A Match

This question requires the applicant to have the knowledge of the SLC pump capacity, use of both pumps and when one pump is to be secured. And the ability to determine how long it will take to pump the SLC tank.

Technical References:

05-S-01-EP-2, RPV Control, Rev 47
02-S-01-40, EP Technical Bases, Rev 9
GLP-OPS-C4100, Standby Liquid Control system lesson plan, Rev 16.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C4100, OBJ 6.0 and 7.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(1)	
Level of Difficulty:	4.0	

PRA Applicability:

None.

Lesson Content	Instructor Notes

The tank is also used to verify injection flow into the reactor vessel.

- During this test, the tank is filled with demineralized water from the Makeup Water Treatment System.
- With the test tank outlet valve F031 fully open, the SLC pumps are started and the squib valves fired, injecting demineralized water into the reactor vessel.

Test Tank outlet valve F031 is a manual isolation valve with position indication provided on Control Room Panel P601 by red full open and green full closed indicating lights.

10.1, 10.2, 10.3 As stated earlier, if F031 is not fully closed, the SLC pump suction valves F001A/B will not automatically open on a pump start signal.

EO-10.1, 10.2, 10.3

The F001A/B can be opened with the local handswitch.

- Prevents contaminating the test tank with solution from the SLC storage tank.

Test Tank level indication is provided locally on the side of the tank by level gauge LG-N106.

This is a sight glass type gauge calibrated in gallons.

Ask Student Question #1

SLC Pumps

4.1 Each of the two divisions of SLC has a motor driven, positive displacement pump to inject the sodium pentaborate solution into the reactor vessel.

EO-4.1

Figures 1 & 2

Each SLC pump is capable of injecting the required volume of sodium pentaborate solution into the reactor vessel at the flow rates outlined in the Technical Specifications.

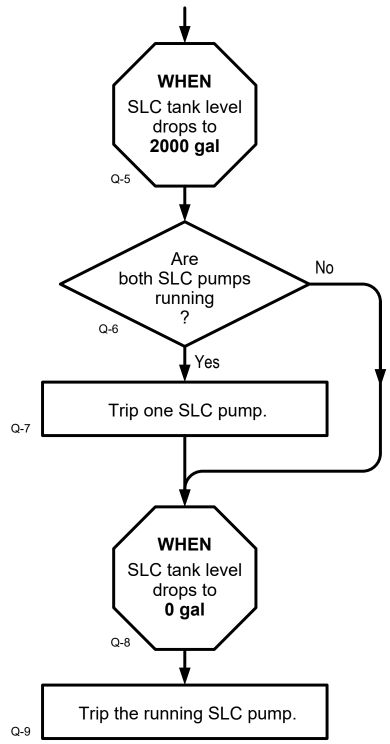
5.1 The SLC pump motors are 40 horsepower, 3-phase induction motors, powered from 480V AC MCCs 15B21 (Pump A) and 16B31 (Pump B).

EO-5.1

6, 7.1 Each pump, located inside Containment at the 185′ Elevation in the SLC System area, is a three-plunger type, positive displacement, 50 percent capacity pump, with a rated flow of 41.2 gpm at 1370 psig.

EO-6, 7.1

Steps Q-5 through Q-9



Discussion

If SLC is initiated, boron injection is continued until the entire contents of the SLC tank have been injected. If available, both pumps are initially operated to increase the injection rate and shorten the time required to inject Cold Shutdown Boron. Once SLC tank level drops to 2000 gal, injection of Cold Shutdown Boron has been completed, permitting initiation of RPV depressurization in Step P-7. One SLC pump may then be tripped to preclude vortexing at the tank suction, theorized to be possible under some conditions with two pumps in operation. The remaining contents of the SLC tank are injected, using a single SLC pump, to provide the design basis concentration margin. When the SLC tank level drops to 0 gal, the remaining SLC pump is tripped to avoid mechanical damage to the pump and preserve availability of the SLC system should operation again be needed.

Cold Shutdown Boron is the amount that results in a boron concentration equivalent to 780 ppm natural boron when injected into the RPV and mixed uniformly. Equivalent tank levels for normal boron injection using SLC and alternate boron injection using RCIC or HPCS are provided in the Emergency Depressurization branch. The derivation of the Cold Shutdown Boron action level is discussed in Section 12.0.

Examination Outline Cross Reference	Level	RO
295038 High Off-Site Release Rate EK2. Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: EK2.10 Condenser air removal system.	Tier	1
	Group #	1
	K/A	295038 EK2.10
	Rating	3.2
	Revision	0
Revision Statement:		

Question: 19

To prevent high offsite release the Mechanical Vacuum Pumps (MVPs) are secured prior to exceeding ___(1)___% power and the Steam Jet Air Ejectors (SJAEs) are placed in service at _______(2)______psig.

A. (1) 5%
(2) 300 psig

B. (1) 5%
(2) 500 – 800 psig

C. (1) 10%
(2) 300 psig

D. (1) 10%
(2) 500 – 800 psig

Answer: B
Explanation: Per 03-1-01-1, IOI, complete placing the SJAE in service at 500 to 800 psig and P&L 2.10 states, "Mechanical Vacuum Pumps will NOT be operating for Main Condenser evacuation when Reactor power is 5% or greater as indicated on IRMs using the IRM Scale reading v IRM Range chart in 17-S-02-1, Manual Core Heat Balance." 'B' is correct
Distracters: 'A' is wrong – Plausible due to this is the required power level but at 300 psig warmup is started for the SJAE not place in service. 'C' is wrong – Plausible due to this is the power level at which APRM gains are adjusted at 300 psig warmup is started for the SJAE not place in service.

‘D’ is wrong - Plausible due to this is the power level at which APRM gains are adjusted

K/A Match

This question requires the applicant to have the knowledge of the interrelationship with Condenser Air Removal system and high release rates.

Technical References:

03-1-01-1, IOI, Cold Shutdown to Generator Carrying Minimum Load, Rev. 183

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-I0I01, OBJ 3.15 and 25.8

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(12)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

COLD SHUTDOWN TO GENERATOR CARRYING MINIMUM LOAD
REACTOR HEATUP AND PRESSURIZATION
SAFETY RELATED

STEP	ACTIVITY	INITIALS

CAUTION

ADJUST RFPT speed to maintain discharge pressure 200-250 psig above Reactor. IF level control problems are experienced, it May be necessary to Control the startup LCV in manual. IF level control is still a problem, RFP discharge pressure May be adjusted to stabilize Reactor level.

34. **ENSURE** the second Reactor Feed Pump is ready for operation, **PER** SOI 04-1-01-N21-1. __________

NOTE

Bypass Control Valve before Seat Drain Valves Should only be opened upon annunciation of BSCV A/B/C UPSTRM DR LVL HI on 1H13-P680-9A-A5/B5/C5 **AND** closed upon resetting of annunciation, **OR** during unit shutdown as specified in the IOI.

35. **CLOSE** Bypass Control Valve Drain Valves **USING** BSCV UPSTRM DR VLV, F300A, B **AND** C JOG CLOSE pushbuttons on 1H13-P680-10C. __________

    N33-F300A
    N33-F300B
    N33-F300C

36. **CLOSE** the Main Steam Line Drain Bypass Valves **USING** DR LINE BYP VLVS CLOSE pushbutton on 1H13-P680. __________

500 to 800 psig

37. Per ODCM 6.11.7, any time the SJAEs are placed in service, **ENTER** an LCO until N64-F045 is placed in TREAT. __________

38. **NOTIFY** Chemistry about one hour before placing a SJAE in service so that they May obtain an isotopic grab sample on the Radwaste Building vent **AND/OR** at Offgas Post Treatment Monitor while charcoal beds are bypassed. __________

COLD SHUTDOWN TO GENERATOR CARRYING MINIMUM LOAD
REACTOR HEATUP AND PRESSURIZATION
SAFETY RELATED

STEP ACTIVITY INITIALS

39. **COMPLETE** placing SJAE in service by **OPENING** N62F003A/B **PER** SOI 04-1-01-N62-1. ____________

40. **AFTER** Mechanical Vacuum pumps have been shutdown, **THEN PERFORM** the following:

    40.1 **NOTIFY** I&C to **RESET** Main Steam Line Radiation Monitor setpoints to 3 x normal background after Mechanical Vacuum pumps are secured **IF** changed. ____________

    40.2 **NOTIFY** Chemistry that isotopic sampling of Turbine Building ventilation may be needed due to shutdown of Mechanical vacuum pumps.

NOTE

Hydrogen Injection system will automatically shutdown (EC43050) on **ANY** one of the following:

- RFPT speed less than 1500 rpm
- RPV level less than 15 inches
- RPV level greater than 50 inches

    40.3 **WHEN** a SJAE is in service **AND** the parameters to allow operations are met, **THEN PLACE** Hydrogen Water Chemistry system in service as soon as possible **PER** SOI 04-1-01-P73-1. ____________

41. **REFILL** loop Seals per SOI 04-1-01-N33-1, Seal Steam **AND** Drains. ____________

NOTE

The HPCS Hi Level Logic Must be reset prior to exceeding 600 psig **OTHERWISE**, HPCS initiation function Must be declared INOPERABLE. **REFERENCE** Tech Spec Bases 3.3.5.1, Function 3b.

42. **DEPRESS** HPCS HI LVL RESET pushbutton **WHEN** "RX LVL 8 (+55") HI" annunciator has cleared. ____________

2.3 Chemistry Limits specified in a step are to be achieved before proceeding beyond plant condition specified in that step. Chemistry department may also apply further restrictions on startup in accordance with applicable chemistry procedures.

2.4 Place in service **OR** remove from service Condensate Cleanup Demineralizers to maintain flow through an individual Vessel between 2000 **AND** 4600 gpm **AND** individual Vessel dP < 25 psid. It is very important to ensure all Condensate flow passes through the Condensate Full Flow Filters (CFFF) **AND** deep bed demineralizers **WHEN** feeding the Reactor. **IF** CFFF is **NOT** in service, the precoat filters Should be in service.

2.5 Condensate Demin differential pressure Controller is set to Control at 60 psid, unless in the process of starting the Condensate Demin System per SOI 04-1-01-N22-1.

2.6 **WITH** Condensate pumps running, the Condensate Full Flow Filtration system **OR** Precoat filters Should be in service prior to providing flow through Condensate Demineralizers **OR** to Reactor vessel. **IF** Precoat filters are removed from service for Condensate **OR** Feedwater lineup changes, they Should be returned to service as soon as possible unless Condensate Full Flow filters are placed in service.

2.7 **NOTIFY** Radwaste before any major changes in Condensate **OR** Feedwater flow **OR** flow paths, **IF** possible.

2.8 **MAINTAIN** Drywell pressure less than 1.0 psig in order to prevent inadvertent scrams, Containment isolation, **AND** ECCS initiation.

2.9 **MAINTAIN** Drywell average air temperature less than 135°F, per Technical Specifications 3.6.5.5.

2.10 Mechanical Vacuum Pumps will NOT be operating for Main Condenser evacuation when Reactor power is 5% or greater as indicated on IRMs using the IRM Scale reading v IRM Range chart in 17-S-02-1, Manual Core Heat Balance.

2.11 **TECH SPEC TRIGGER (TR 3.3.6.1-2.5)**
IF Mechanical Vacuum Pumps are taking suction from the Condenser **AND** all Control Rods Are **NOT** fully inserted, **THEN** Main Steam Line Radiation Monitor setpoints Must be set in accordance with TRM Table TR 3.3.6.1-2.5. For normal operations, this lower setpoint Does **NOT** apply **AND** the setpoint is 3 times full power background.

2.12 Line break annunciation for LPCI A, LPCI B, LPCI C, LPCS **AND** HPCS is only valid for Reactor power >80% **AND** core flow >90%.

Examination Outline Cross Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances AK1. Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AK1.01 Definition of terms: volts, watts, amps, VARs, power factor.	Tier	1
	Group #	1
	K/A	700000 AK1.01
	Rating	3.3
	Revision	0
Revision Statement: **NEW QUESTION**		

Question: 20

Per 05-1-02-I-4, Loss of AC Power ONEP:

(1) Which of the following system voltages meets the definition of a grid disturbance?

(2) If ESF bus voltage cannot be maintained, the crew will ____________.

A. (1) 495 Kv
   (2) Manually Scram the Reactor and Trip the Generator

B. (1) 495 Kv
   (2) Start Emergency Generators and separate from Offsite power

C. (1) 527 Kv
   (2) Start Emergency Generators and separate from Offsite power

D. (1) 527 Kv
   (2) Manually Scram the Reactor and Trip the Generator

Answer:	**C**
Explanation: Per ONEP 05-1-02-I-4, Loss of AC Power, Step 3.4; Any of the following are possible symptoms of Grid Instability: • Low system voltage <491Kv • Hi system voltage >525 Kv • Low system frequency <58.3 Hz-(Gen Low Frequency Alarm)	

- Oscillating system voltage
- Oscillating system frequency

And step 3.4.6;

IF any ESF bus voltage CANNOT be maintained > 3952 volts, grid frequency CANNOT be maintained > 58.3 Hz, OR 500 KV frequency/voltage is erratic, THEN START respective Emergency Diesel Generators, SYNCHRONIZE to respective bus, AND SEPARATE from Offsite power per Attachment IV.

'C' is correct

Distracters:

'A' is wrong – but plausible, the Voltage does not reach the setpoint for instability. Nowhere in this section of the ONEP does it say to scram the reactor and trip the generator.

'B' is wrong – but plausible, the Voltage does not reach the setpoint for instability.

'D' is wrong - but plausible,Nowhere in this section of the ONEP does it say to scram the reactor and trip the generator.

K/A Match

This question requires the applicant to have the knowledge of the implications if the main generator and grid voltage show indications of grid instability.

Technical References:

05-1-02-I-4, Loss of AC Power, Rev. 50, step 3.4.3 CAUTION

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-ONEP, OBJ 6
GLP-OPS-N4151, OBJ 14

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

3.3.12g (Cont.)

(9) **START** Reactor Recirculation Pumps as desired. (Temperature Stratification) 04-1-01-B33-1

(10) **START** RWCU (Temperature Stratification) 04-1-01-G33-1

NOTE

Control Room Air Conditioning Units Z51-B002A(B) Temperature Control valves fail open on loss of instrument air, causing compressors to unload on low freon pressure, thus reducing unit cooling capacity.

3.3.13 **IF** Instrument Air **CANNOT** be immediately restored, **THEN** **THROTTLE** P41F075A(B) for the running Control Room air conditioner to maintain condenser pressure between 220 **AND** 260 psig (240 psig is optimum) as indicated on local indicator SZ51-PIC-R008A(B). **IF** condenser pressure **CANNOT** be maintained > 220 psig, **THEN** **THROTTLE** P41F075A(B) in order maintain the Control Room A/C unit in operation.

a. **IF** Control Room Air Conditioner has tripped on low condenser pressure, **THEN** **RESET** the low suction/high discharge pressure switch in air compressor control cabinet **AND** **RESTART** desired compressor.

3.4 Grid Instability

NOTE

Any of the following are possible symptoms of Grid Instability:

- Low system voltage <491Kv
- Hi system voltage >525 Kv
- Low system frequency <58.3 Hz-(Gen Low Frequency Alarm)
- Oscillating system voltage
- Oscillating system frequency

Main Generator Will trip **IF** Generator frequency drops to 57 Hz.

Feeder Breakers to Bus 15AA/16AB Will trip **IF** respective bus voltage drops to 3790.5 volts for 9 seconds **OR** 2768.5 volts for 0.5 seconds.

Feeder Breakers to Bus 17AC Will trip **IF** Bus 17AC voltage drops to 3661 volts for 5 minutes (4 sec T.D. with LOCA signal) **OR** 3045 volts for 2.3 seconds.

3.4.1 **CONSULT** with System Operations Center (Woodlands) **AND** Mississippi Transmission Operations Center (Pine Bluff) to determine reason **AND** extent of grid instability.

3.4.2 **CONTACT** Duty Manager.

<u>CAUTION</u>

- GGNS Main Generator terminal MVAR limitations are +297 MVARS and -253 MVARS for normal and emergency operation.
- Maintaining Generator MVARS within these limits should prevent Main Generator stator bars from overheating and ensure Reverse Power Relay recognizes a reverse power condition.

3.4.3 **MAINTAIN** Main Generator MVARS between +297 and -253 MVARS.

3.4.4 **PERFORM** Surveillance 06-OP-1R20-W-0001-02, Plant AC **<u>AND</u>** DC Electrical Power Distribution to determine operability of AC power distribution system.

a. **REFER** to Tech Spec 3.8.1, 3.8.2, 3.8.7, **<u>AND</u>** 3.8.8, as applicable, for any inoperable AC power source **<u>OR</u>** distribution system.

<u>CAUTION</u>

- ETAP analysis (Ref. GEXI 2009-00007) states that the plant is NOT aligned with all three ESF buses on one ESF transformer.
- <u>IF</u> a LOCA incident were to occur under degraded grid voltage condition while in this configuration, the LSS under voltage bistables would trip and NOT recover above reset value within the 9 second requirement PER SDC-16 Rev. 0.
- Do NOT ALIGN all three ESF buses (15AA, 16AB, and 17AC) to one ESF transformer.

3.4.5 **<u>IF</u>** bus voltage for any ESF bus 15AA, 16AB, **<u>OR</u>** 17AC drops to ≤ 3952 volts **<u>AND</u>** the 115KV system voltage requirements are met per 06-OP-1R20-W-0001-02, **<u>THEN</u> TRANSFER** respective ESF bus feeder to ESF XFMER 12 per respective SOI 04-1-01-R21-15, 04-1-01-R21-16, **<u>OR</u>** 04-1-01-R21-17.

- Bus 15AA
- Bus 16AB
- Bus 17AC

3.4.6 **<u>IF</u>** any ESF bus voltage **<u>CANNOT</u>** be maintained > 3952 volts, grid frequency **<u>CANNOT</u>** be maintained > 58.3 Hz, **<u>OR</u>** 500 KV frequency/voltage is erratic, **<u>THEN</u> START** respective Emergency Diesel Generators, **SYNCHRONIZE** to respective bus, **<u>AND</u> SEPARATE** from Offsite power per Attachment IV.

a. **<u>WHEN</u>** time **<u>AND</u>** manpower allows, **<u>THEN</u> STATION** Operator(s) at running Diesel Generators to monitor operation.

Examination Outline Cross Reference	Level	RO
295008 High Reactor Water Level AA1. Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL: AA1.06 HPCS: Plant-Specific.	Tier	1
	Group #	2
	K/A	295008 AA1.06
	Rating	2.8
	Revision	0
Revision Statement:		

Question: 21

An event occurred causing Drywell pressure to rise to 1.6 psig and stabilize there.

Reactor water level lowered to -65 inches.

HPCS has auto initiated

RPV water level is restoring.

(1) At what reactor water level will HPCS stop injection?

And

(2) What is required to restore injection?

A. (1) Level 8 Narrow Range
(2) Level less than level 8 and depress the HPCS HI LVL RESET pushbutton

B. (1) Level 8 Wide Range
(2) Level less than level 8 and depress the HPCS HI LVL RESET pushbutton

C. (1) Level 8 Narrow Range
(2) Level less than level 8 and depress the HPCS INIT RESET pushbutton

D. (1) Level 8 Wide Range
(2) Level less than level 8 and depress the HPCS INIT RESET pushbutton

Answer: B
Explanation: HPCS will auto initiate at 1.39 psig Drywell pressure or -41.6 inches RPV level. Once initiated injection will auto stop at level 8 (+53.5 inches) on the Wide range indicators. At that point the white indicating light above the HPCS HI LVL RESET pushbutton will illuminate. With the initiation signal still present

level must be below level 8 and the operator must depress the HPCS HI LVL RESET pushbutton to resume injection.

‘B’ is correct

Distracters:

‘A’ is wrong – Plausible due to Narrow range level instrumentation also performs actions at level 8 (RCIC system shutdown) and level 9 Feedwater pump trip.

‘C’ is wrong – Plausible due to Narrow range level instrumentation also performs actions at level 8 (RCIC system shutdown) and level 9 (+56 inches) Feedwater pump trip.

‘D’ is wrong – Plausible due to the High Level Reset must be depressed first then the initiation button can be used.

K/A Match

This question requires the applicant to have the ability to operate and monitor HPCS injection, when it auto stops and how to resume injection.

Technical References:

04-1-01-E22-1, HPCS SOI, Rev. 127
GLP-OPS-E2201, HPCS Lesson Plan, Rev 13

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2201, OBJ 9.4

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:

HPCS is listed as #16 on the System Importance to CDF.

Lesson Content	Instructor Notes
▪ With the P601 handswitch in the AUTO position, the valve will automatically open upon a HPCS initiation provided: - it has NOT been manually overridden and - a Level 8 signal is NOT present.	Objective 9.4 Figure 10 2 handed operation is allowed to override HPCS
▪ In the OPEN position, the valve will open as long as a Level 8 (Wide Range) signal is NOT present.	
▪ The Reactor water level is monitored by transmitters B21-LT-N073C and L which send signals to level switches B21-LS-N674C and L respectively.	
- When both level switches indicate a level of 53.5 inches, a signal is generated to close the injection valve.	Both switches required to energize K13
- Any of the following conditions will cause the injection valve to automatically reopen provided an initiation signal is still present and manual override has not been activated: ▪ when the high water level signal is reset ▪ at Level 2	
- Additionally, when Reactor water level rises to 53.5 inches, the level switches actuate the following: ▪ RX LVL 8 (+55") HI annunciator at P601-16A-A3, ▪ HPCS HI WTR LVL SEAL-IN annunciator at P601-16A-B3, ▪ White HPCS WATER LEVEL HI status lamp above the HPCS HI LEVEL RESET Pushbutton on P601-16B.	Table 2 Figure 8
▪ The injection valve can be closed by the handswitch when an initiation signal is present. - If the injection valve must be reopened, it has to be done from the handswitch.	

Lesson Content	Instructor Notes
▪ In the NORMAL position, this switch bypasses the motor-operated valve thermal overload devices in the following HPCS valves to ensure they will operate upon HPCS initiation: - Pump Suction from CST Valve, F001 - Pump Suction from Suppression Pool Valve, F015 - Injection Shutoff Valve, F004	
- Minimum Flow to Suppression Pool Valve, F012 - Inboard Test Return to CST Valve, F010 - Outboard Test Return to CST Valve, F011 - Test Return to Suppression Pool Valve, F023	Figure 14 depicts MOV test switch logic (typical of all HPCS MOVs).
▪ Placing HS-M614 in the TEST position inserts the thermal overload devices into the motor circuits to allow testing of the valve and its protective features.	
▪ With the switch in TEST the HPCS MOV IN TEST STATUS lamp on P601-16B is illuminated.	
▪ In addition, the HPCS SYS OOSVC annunciator at P601-16A-H5 activates.	Table 2
16. HPCS High Level Reset Pushbutton	
▪ When a Reactor water Level 8 (+53.5") signal is received, the signal is sealed in the logic as indicated by activation of annunciator HPCS HI WTR LVL SEAL-IN at P601-16A-B3 and HPCS WATER LEVEL HI status lamp illuminated on P601-16B above the HPCS HI LVL RESET pushbutton. ▪ To manually reset this signal after the Level 8 signal has cleared, the HPCS HI LVL RESET pushbutton, HS-M612, is depressed. ▪ When the signal has cleared, a white indicating light above the pushbutton is extinguished.	Figure 8 Table 2

3.8 **IF** F004 is manually reopened after auto closure on a high reactor water level signal (white light above HPCS HIGH LEVEL RESET pushbutton is On **AND** the "HPCS HI WTR LVL SEAL-IN" annunciator 1H13-P601-16A-B3 in Alarm), HPCS HIGH LEVEL RESET pushbutton Must be **DEPRESSED** first. With a LOCA signal present, F004 Will automatically maintain water level between -42 inches **AND** +55 inches.

3.9 The HPCS PMP SUCT FM SUPP POOL F015 handswitch placed to the CLOSED position (with an AUTO open signal present) Will override an opening signal from Either condensate storage tank low level signal **OR** suppression pool high level signal.

3.10 **IF** the HPCS pump is stopped with an initiation signal present, the HPCS pump Will **NOT** start automatically. **IF** needed, HPCS pump Must be started manually. The reinstatement of Auto Start Feature requires resetting HPCS initiation logic by **DEPRESSING** the HPCS INITIATION SIGNAL RESET pushbutton. The reactor low level (-41.6 in) logic Can be reset only after the low level condition has cleared. The HPCS high drywell pressure (1.39 psig) initiate logic Can be reset at any time with Shift Supervision's permission.

3.11 The HPCS pump motor May be started twice successively from ambient temperature **OR** once from rated temperature. **WHEN** needed for testing conditions, HPCS Can be assumed to have returned to ambient temperature after 100 minutes at standstill **OR** rated temperature after 50 minutes running time, at which time another start is permissible.

3.12 The HPCS System Should **NOT** be placed in operation to supply reactor **WHEN** reactor is operating, **OR** to fill vessel under normal conditions.

3.13 Repetitive HPCS injection Should be minimized to reduce number of system thermal cycles.

3.14 The HPCS System Should be placed out of service anytime a system flush is performed. **CHECK** that F003 **AND** F031 are LOCKED CLOSED before returning system to service.

3.15 The HPCS pump motor full load amperage is 434 amps with a 1.0 service factor.

3.16 Do **NOT** place the HPCS MOV TEST handswitch to TEST position during ONEPS, EPS, **OR** transient situations. HPCS MOV TEST handswitch May be placed to TEST position during surveillance testing, special testing **OR** verification that thermals are **NOT** tripped following an automatic valve actuation. The handswitch Should remain in TEST position for 30 seconds following valve operation.

3.17 MOV test switches should **NOT** be left in TEST for longer than eight hours. **IF** any switch is left in TEST for longer than eight hours, **THEN** the associated valve(s) **MUST** be declared Inoperable until the switch is returned to the NORM position. (TRM LCO 6.8.2)

3.18 At no time during operation Shall ECCS be placed in MANUAL by an operator unless mis-operation in AUTOMATIC is confirmed (by at least two independent indications) **OR** adequate core cooling is assured. While in MANUAL, frequent **CHECKS** of initiating **OR** controlling parameters Shall be made.

3.18.1 At no time during operation Shall ECCS be secured by an operator, unless there are at least two independent indications that adequate core cooling is assured.

Examination Outline Cross Reference	Level	RO
295010 High Drywell Pressure AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : AK1.03 Temperature increases.	Tier	1
	Group #	2
	K/A	295010 AK1.03
	Rating	3.2
	Revision	0
Revision Statement: **Per NRC review (10 free view) SAT**		

Question: 22

A LOCA has caused drywell pressure to reach 2.8 psig.

Drywell temperature is 211°F

A complete loss of offsite power has occurred, including the Port Gibson line.

All DGs have re-powered the ESF buses.

NO Operator actions have been performed.

Which of the following identifies the method of heat removal from the drywell under the current plant conditions?

A. Heat removal from the drywell is only by ambient losses without air circulation.

B. Drywell Coolers and Drywell Chilled Water System 'B' are operating on SSW 'B'.

C. 'A' Drywell Cooler fans are circulating air but the coolers are without cooling water flow.

D. 'B' Drywell Cooler fans are circulating air but the coolers are without cooling water flow.

Answer: A
Explanation: Drywell pressure caused an LSS LOCA Shed and Sequence. The Drywell Coolers are powered from 15B42 and 16B42, which are shed on a LOCA and not automatically re-energized. Operator action is required to restore power to the Drywell Coolers. 15B42 and 16B42 are shed and re-energized automatically on a loss of power, however the LOCA signal will override the loss of power sequence. Additionally, the Drywell Chilled Water supply and return valves will be automatically isolated due to the Drywell Pressure.

‘A’ is correct

Distracters:

‘B’ is wrong – Plausible, procedures does not allow Drywell Chilled Water to be restored by reopening isolation valves without complete system startup when Drywell temperature is > 200°F, SSW ‘B’ is aligned for LOCA cooling which is not aligned to the DW chilled water system and ‘B’ Chilled water system is powered from Div 2 16AB which is locked out on a LOCA by LSS.

‘C’ and ‘D’ are wrong – Plausible, the Drywell Coolers are powered from 15B42 and 16B42, which are shed on a LOCA and not automatically re-energized

K/A Match

This question requires the applicant to recognize the implications with drywell temperature when drywell pressure is high.

Technical References:

04-1-01-R21-1 Table 1, Rev 106

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-M5100.00 Objective: 13, 14, 19, 21

Question Source:	**Bank # 418**	X Not used on NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

ESF (R20) is listed as #4 on the System Importance to CDF.
Offsite power is listed as #12 on the System Importance to CDF.

TABLE 1 (Continued)

DIVISION 1 LSS

EQUIPMENT	SHED	SEQUENCING TIME IN SECONDS			REMARKS
		BUV	LOP	LOCA	
D/G Aux Jacket Water Pump A 1P75C004A	X	(15)	(15)	(20)	Permissive with low jacket water pressure and diesel at rated speed. Normally will not sequence unless engine driven pump fails.
Fdr Bkr for MCC 15B42 52-15405	X	15	15	-	Can be manually closed after LOCA.
Battery Charger 1D4 52-15102	X	10	20	-	Can be manually closed locally after LOCA.
Battery Charger 1D5 52-15202	X	10	20	-	Can be manually closed locally after LOCA.
Battery Charger 1K4 52-15104	X	10	20	-	Can be manually closed locally after LOCA.
Battery Charger 1K5 52-15204	X	10	20	-	Can be manually closed locally after LOCA.
Hydrogen Recombiner A 1E61C003A	X	(15)	(15)	(20)	Will sequence only if running prior to load shed. Otherwise requires manual start.
Drywell Purge Compressor A 1E61C001A	X	-	-	30	Permissive with Containment/drywell low dP. Also initiates Combustible Gas Control System (no time delay).
Control Rod Drive Pump A 1C11C001A	X	-	-	-	Must be manually re-started.
Fuel Pool Cooling Pump A 1G41C001A	X	-	-	-	Must be manually re-started.

TABLE 1 (Continued)

DIVISION 2 LSS

EQUIPMENT	SHED	SEQUENCING TIME IN SECONDS			REMARKS
		BUV	LOP	LOCA	
RHR Pump C 1E21C002C	X	-	-	0	
D/G Aux Main Lube Oil Pump 1P75C007B	X	(0)	(0)	(0)	Permissive with low lube oil pressure and diesel at rated speed. Normally will not sequence unless engine driven pump fails.
RHR Pump B 1E12C002B	X	-	-	5	
Enclosure Bldg. Recirc Fan B 1T48C001B	X	(0)	(0)	5	Permissive with SBGT System manual or automatic initiation signal.
SSW Pump B P41C001B	X	5	5	10	Will also initiate SSW System B (no time delay) on a LOCA or on an actual LOP (i.e., realign various valves).
Control Room Air Hdlg Unit B SZ51B002B	X	5	5	10	Breaker closing time; not necessarily load actuation.
D/G Room Outside Air Fan A 1X77C001B	X	5	5	10	Breaker closing time; not necessarily load actuation.
Swgr Rm Air Handling Unit B Heating Coil 1Z77B001B	X	10	10	15	Permissive with low temperature and associated supply fan running.
Swgr Rm Air Handling Unit B Heating Coil 2Z77B001B	X	10	10	15	Permissive with low temperature and associated supply fan running.
SSW Cooling Tower Fan C 1P41C003C	X	10	10	15	
SSW Cooling Tower Fan D P41C003D	X	10	10	15	
Fdr Bkr 152-1603 (XFMR for Drywell Chillers)	X	20	20	-	Locked out on LOCA. Can be manually started on BUV/LOP.

Examination Outline Cross Reference	Level	RO
295020 Inadvertent Containment Isolation AK3. Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: AK3.03 Drywell/containment temperature response.	Tier	1
	Group #	2
	K/A	295020 AK3.03
	Rating	3.2
	Revision	1
Revision Statement: **Per Ops Rep, Changed the stem to state "the CRS directs to restore the Aux Building"**		

Question: 23

An inadvertent Containment isolation signal has occurred and will not reset.

The CRS directs to restore the Auxiliary Building per the Automatic Isolations ONEP.

5 minutes after the actions are complete:

Which of the following describes the reason for Containment temperature and Drywell Temperature response?

A. Containment and Drywell temperature will remain the same due to cooling water can be bypassed and restored.

B. Containment and Drywell temperature will rise due to ventilation cooling water will isolate and not able to restore.

C. Containment temperature will rise due to ventilation cooling water will isolate and not able to restore, Drywell temperature will remain the same due to cooling water can be bypassed and restored.

D. Containment temperature will remain the same due to cooling water can be bypassed and restored. Drywell temperature will rise due to ventilation cooling water will isolate and not able to restore.

Answer: C
Explanation: After an isolation signal the ONEP allows to restore certain systems that can bypass the isolation signal: Instrument Air Drywell Chilled Water

Therefore, Drywell temperature will remain the same due to no interruption with cooling water.

The Containment coolers are cooled by Plant Chilled water system that receives an isolation signal and no ability to bypass the isolation, therefore no cooling water will be supplied to the coolers and Containment temperature will rise.

'C' is correct

Distracters:

'A' is wrong – Plausible, as stated above CTMT coolers will no longer have cooling and cannot be restored, Drywell temp will maintain cooling and no rise in temperature.

'B' is wrong – Plausible, the Drywell Coolers will maintain cooling

'D' is wrong – Plausible, as stated above the Drywell Coolers will maintain cooling and CTMT coolers will no longer have cooling

K/A Match

This question requires the applicant to recognize the reasons for a change in parameters within the Drywell and Containment during Containment isolation.

Technical References:

GLP-OPS-P7100, Rev 15
05-1-02-III-5, Automatic Isolation ONEP Rev. 52

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-M5100.00 Objective: 13, 14, 19, 21

Question Source:	**Bank # 418**	X Not used on NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Auxiliary Building, Containment, Drywell Isolation

Objective 11

Figure 14

11 During a Loss of Instrument Air, the air-operated isolation valves, P71-F148, F149, F150, F300, F301, F302, F303, F304, F305, F306, and F307, will close.

*Valves F300, F301, F302, F303, F304, F305, F306, and F307 are currently gagged open per GG-EC-0026 and will not close but do receive a close signal.

The closure of isolation valves F304, F305, F306, or F307 will trip the Primary and Secondary Chilled Water Pumps, thereby making all of the Plant Chilled Water System unavailable for service.

*These valves are currently gagged open per GG-EC-0026.

Containment Isolation valves P71-F148, F149 and F150 will close on a Group 6 Containment Isolation:

Low Reactor Vessel Level -41.6"

High Drywell Pressure 1.23 psig

These valves cannot be overridden and therefore, PCW to the containment cannot be reestablished until the initiating signal has cleared.

♫ & HU Knowledge – Maintain integrated plant knowledge.

SYSTEM INTERRELATIONSHIPS

Plant Service Water System (P44)

Objective 14.1

14.1 The Plant Service Water System provides cooling (the heat sink) to the plant chiller condensers.

Loss of plant service water results in the loss of chilled water, leading to high area temperatures and, subsequently, a plant shutdown.

Instrument Air System (P53)

Objective 14.2

14.2 The Instrument Air System supplies air to operate all the air-operated valves in the Chilled Water System, including the isolation valves, the three-way temperature control valves, and the Expansion Tank makeup water valve (F556A).

Loss of instrument will cause these valves to fail closed resulting in a complete loss of chilled water.

*Valves F300, F301, F302, F303, F304, F305, F306, and F307 are currently gagged open per GG-EC-0026 and will not close but do receive a close signal.

FOR EMERGENCY USE ONLY

P53/P72 Auxiliary Building Restoration

Restore Auxiliary Building Isolations as follows:

1. **IF** Plant Air Compressors are available,
   THEN OPEN/CHECK OPEN the following Instrument Air valves on 1H13-P870:

   Section 3C

   - P53-F001(After 30 sec T.D.)

   Section 9C

   - P53-F007(After 30 sec T.D.)

CAUTION

IF Drywell Chilled Water is restored after isolation when Drywell temperature exceed 200°F, THEN a water hammer may occur and rupture Drywell cooler tubes due to voiding in piping.

IF Drywell temperatures exceeds 200°F (CRD Cavity temperatures excluded), THEN Drywell Chilled Water should NOT be unisolated until controlled startup can be performed PER SOI 04-1-01-P72-1 or Drywell temperature has returned to < 200°F.

2. **IF** all Drywell temperatures are less than 200°F (CRD cavity temperature excluded) **AND** Drywell Chillers are available,
   THEN RESTORE Drywell Chilled Water as follows:

   a. **IF** required,
      THEN RE-ENERGIZE MCC's 15B42 and 16B42 on 1H13-P864.

   b. After a 30 second time delay, **OPEN** the following valves on 1H13-P870:

      Section 3C

      - P72-F121
      - P72-F122
      - P72-F125

      Section 9C

      - P72-F123
      - P72-F126
      - P72-F124

Examination Outline Cross Reference	Level	RO
295029 High Suppression Pool Water Level EA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL : EA2.03 Drywell/containment water level	**Tier**	1
	Group #	2
	K/A	295029 EA2.03
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 24

A plant operator reports a pipe break in the containment causing water to drain into the Suppression Pool.

Suppression Pool level is currently 19.5 ft and rising.

Due to a logic failure Suppression pool makeup inadvertently initiates.

A RO is monitoring Suppression Pool Water level and reports that level is stable at 24.3ft.

Which of the following describes the reason Suppression Pool Level indication has stabilized?

A. Overflowing the Drywell weir wall.

B. Maximum indication for wide range.

C. Maximum amount of water from the upper pools.

D. Level has reached the point of transmitters being submerged.

Answer: A
Explanation: The design of the Suppression Pool Makeup system is without considering any other inputs, an inadvertent SPMU dump will raise Suppression Pool level to ≈23.8' if pool level was initially at its normal level of 18.5' or raise level approximately 5.3 ft. With suppression pool level at 19.5’ plus the 5.3’ dumped is 24.8’. the Weir wall to the Drywell stands at 24.3’ therefore the level stabilizing is from overflowing ot the weir wall. ‘A’ is correct

Distracters:

‘B’ is wrong – but plausible, the max indication on the Suppression Pool level indicator is 25.5’ wide range.

‘C’ is wrong – but plausible, the design of the Suppression Pool Makeup system is without considering any other inputs, an inadvertent SPMU dump will raise Suppression Pool level to ≈23.8' if pool level was initially at its normal level of 18.5' or raise level approximately 5.3 ft. With suppression pool level at 19.5’ plus the 5.3’ dumped is 24.8’. the Weir wall to the Drywell stands at 24.3’.

‘D’ is wrong – Plausible, the upper pools dump will fill the suppression pool to 5” below the top of the weir wall with suppression pool level at the high level alarm setpoint.

K/A Match

This question requires the applicant to interpret the indications of a high suppression pool level.

Technical References:

02-S-01-40, EP Technical Bases, Rev 009
GLP-OPS-E3000. Rev 13

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

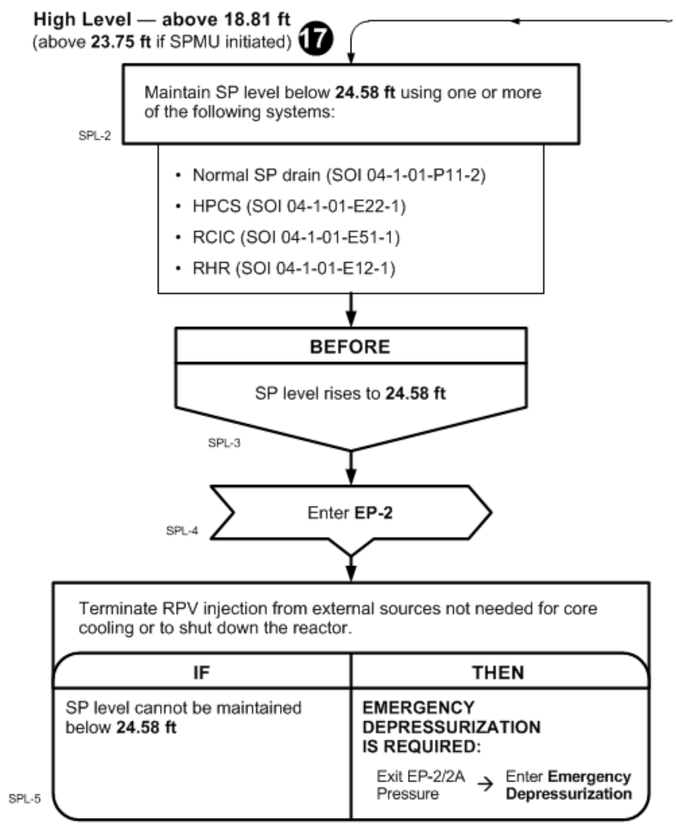
GLP-OPS-EP3 Objective: 7

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(9)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Steps SPL-2 through SPL-5



Discussion

Maintain SP level below 24.58 ft.

If suppression pool water level cannot be maintained below the high end of the control band defined in Step SPL-1, an alternate control band of “below 24.58 ft.” is authorized. Additional methods for removing water to maintain the alternate band are authorized.

The specified suppression pool water level of 24.58 ft. is the Maximum Pressure Suppression Primary Containment Water Level (MPSPCWL) and the bounding value of the SRV Tail Pipe Level Limit (STPLL).

The MPSPCWL is the highest primary containment water level at which the pressure suppression capability of the containment can be maintained. For Mark III containments, the MPSPCWL is generically defined to be 3 in. above the top of the weir wall. Irrespective of the containment airspace volume, the pressure suppression feature of the primary containment can be assumed to function as designed only when primary containment water level is below this elevation. If the

♫ Conservatism – Ensure that margins to an undesirable state are maintained

The elevations of the separator pool penetrations are designed to limit the volume of water that can be dumped to the Suppression Pool.

This volume limitation along with adequate weir wall freeboard ensures that no Drywell flooding over the weir wall occurs due to an inadvertent opening of the dump valves.

During normal operation, the fuel pool and transfer canal gates are removed to ensure that sufficient water is available for Suppression Pool Makeup.

Without considering any other inputs, an inadvertent SPMU dump will raise Suppression Pool level to ≈23.89' if pool level was initially at its normal level of 18.56'.

This is about 5" below the top of the weir wall.

If a large LOCA (-150.3" or 1.39 psig) occurs, the ECCS systems will automatically start to recirculate pool water through the core and RPV via the Drywell.

The volume and flowrate required to fill the Drywell enough to overflow the weir wall and return to the pool will draw pool level down at a maximum rate of 0.86 feet per minute.

One line of SPMU will raise pool level ≈0.82 feet per minute once both valves are full open, and that takes ≈80 seconds from the baseline automatic initiation setpoint at 16.83'.

The actual automatic initiation setpoint is 17.5' to account for off-calibration due to water temperature and density.

This means that pool level could drop to as low as 15.69' before SPMU starts bringing it back up again.

The system therefore meets the SPMU design criterion of maintaining at least 2 feet of water above the top of the first row of pool vents, or 14.5 feet.

If a smaller LOCA (-41.6" or 1.23 psig) occurs, SPMU will initiate after a 29-minute time delay to ensure that the combined Upper Pool and Suppression Pool volumes are available as a heat sink for small breaks which do not lower the Suppression Pool to 17.5' but continue to dump vessel blowdown energy into the pool.

An inadvertent dump of the Upper Containment Pool during refueling removes 7' 1″ of water from the pool.

If a fuel bundle were in the grapple of the Refueling Platform at its highest position, approximately one foot of water would be left over the bundle.

While this is adequate for bundle cooling, it represents a potential radiation hazard for personnel on the refueling floor.

Examination Outline Cross Reference	Level	RO
295032 High Secondary Containment Area Temperature 2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	Tier	1
	Group #	2
	K/A	295032 2.4.47
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 25

HANDOUT PROVIDED

A steam line within the Auxiliary Building Main Steam Tunnel has a leak.

Current MST temperature is 105°F.

MST room temp is rising at 5° F / min

(1) When will automatic actions occur and entry into EP-4?

(2) If the break cannot be isolated, which of the following will cause the CRS to direct an Emergency Depressurization?

A. (1) 16 minutes
(2) RCIC Room temperature of 212°F and MST temperature of 250°F

B. (1) 16 minutes
(2) RCIC Room temperature of 185°F and MST temperature of 250°F

C. (1) 29 minutes
(2) RCIC Room temperature of 212°F and MST temperature of 185°F

D. (1) 29 minutes
(2) RCIC Room temperature of 185°F and MST temperature of 185°F

Answer: A
Explanation: Per 02-S-01-40, EP Technical Bases, EP-4 Table SC-1, Automatic actions will occur at 185°F in the MST, a Group 1 isolation will occur. At 5°F/ min it will take 16 minutes for the steam tunnel to rise 80°F.

For EP-4 to call for a Emergency Depressurization it requires an unisolable leak and two Max safe values of the same type (i.e. temp). Therefore per Table SC-1 the Max safe for the MST is 250°F and RCIC is 212°F

'A' is correct

Distracters:

'B' is wrong – Plausible, RCIC room temp only makes the Operating Limit and the Max Safe is required for ED

'C' is wrong – Plausible, 29 minutes is the time to reach the Max Safe value for the MST of 250°F. 185°F is the setpoint for a Group 1 isolation and EP-4 entry. Also, MST room temp only makes the Operating Limit and the Max Safe is required for ED

'D' is wrong – Plausible, 29 minutes is the time to reach the Max Safe value for the MST of 250°F. 185°F is the setpoint for a Group 1 isolation and EP-4 entry. Also, RCIC and MST room temp only makes the Operating Limit and the Max Safe is required for ED

K/A Match

This question requires the applicant to diagnose and recognize trends of a steam leak using reference material.

Technical References:

02-S-01-40, EP Technical Bases, Rev 009
05-S-01-EP-4, Auxiliary Building Control, Rev. 31

Handouts to be provided to the Applicants during exam:

Operating Hard Card, OPG-37, Aux Building Area Parameters.

Learning Objective:

GLP-OPS-EP4 Objective: 5 and 9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		

RCIC is listed as #17 on the System Importance to CDF.

Operating Hardcard	Aux Building Area Parameters	Date: 7/7/11

AREA	OPERATING LIMIT	MAX SAFE VALUE				
	TEMPERATURE					
MSL PIPE TUNNEL TEMP	185°F(P601-18A-A3 & A4) (P601-19A-A3 & A4)	250°F E31-N604A,B,C,D,E,F				
RHR EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F E31-N608A/B,N610A/B				
RCIC EQUIP AREA TEMP	185°F (P601-21A-G3)	212°F (E31-N602A/B)				
	RADIATION LEVEL					
RHR ROOM A D21K601	10^2 MR/HR (P844-1A-D4)	8 X 10^4 MR/HR				
RHR ROOM B D21K602	10^2 MR/HR (P844-1A-D4)	8 X 10^4 MR/HR				
RHR HX A HATCH D21K611	10^2 MR/HR (P844-1A-C4)	8 X 10^4 MR/HR				
RHR HX B HATCH D21K612	10^2 MR/HR (P844-1A-C4)	8 X 10^4 MR/HR				
RCIC ROOM D21K603	10^2 MR/HR (P844-1A-D4)	8 X 10^4 MR/HR				
MSL RAD MONITOR	Setpoint Log (P601-19A-D4)	8 X 10^4 MR/HR				
SGTS FLTR TRN D21K613	2.5 MR/HR (P844-1A-C5)	8 X 10^2 MR/HR				
	WATER LEVEL					
RHR RM A	89´ 8˝ (P680-8A1-A2)	93´ 6˝ (P870-2A-E1)				
RHR RM B	89´ 8˝ (P680-8A1-B2)	93´ 6˝ (P870-10A-G1)				
RHR RM C	90´ 6˝ (P680-8A1-C2)	93´ 6˝ (P870-10A-G2)				
RCIC RM	90´ 6˝ (P680-8A1-C4)	93´ 6˝ (P870-2A-A1)				
LPCS RM	90´ 6˝ (P680-8A1-A4)	93´ 6˝ (P870-2A-F1)				
HPCS RM	90´ 6˝ (P680-8A1-B4)	93´ 6˝ (P870-5A-H1)				

"TWO MAX SAFE'S" = 2 TEMPS **or** 2 RADS **or** 2 WATER LEVEL

Table SC-1: Aux Building Area Parameters

Table SC-1
Aux Building Area Parameters

Area	Operating Limit	Max Safe Value
TEMPERATURE		
MSL PIPE TUNNEL TEMP	185°F (P601-19A/18A-A3/A4)	250°F (E31-N604A,B,C,D,E,F)
RHR-A EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608A, N610A)
RHR-B EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608B, N610B)
RCIC EQUIP AREA TEMP	185°F (P601-21A-G3)	212°F (E31-N602A/B)
RWCU-PUMP ROOM 1 TEMP	170°F (P680-11A-A1)	NA
RWCU-PUMP ROOM 2 TEMP	170°F (P680-11A-A2)	NA
HVAC EXHAUST RADIATION LEVEL		
AUX BLDG FHA VENT EXHAUST	3.6 mr/hr (P601-19A-B9/C9)	NA
AUX BLDG FHA POOL EXHAUST	30 mr/hr (P601-19A-B10/C10)	NA
RADIATION LEVEL		
RHR ROOM A	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR ROOM B	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR HX A HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RHR HX B HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RCIC ROOM	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
MAIN STEAM LINE RAD MONITOR	Set Point Log (P601-19A-D4)	8×10^4 mr/hr
SGTS FLTR TRN	2.5 mr/hr (P844-1A-C5)	8×10^2 mr/hr
WATER LEVEL		
RHR RM A	89 ft 8 in. (P680-8A1-A2)	93 ft 6 in. (P870-2A-E1)
RHR RM B	89 ft 8 in. (P680-8A1-B2)	93 ft 6 in. (P870-10A-G1)
RHR RM C	90 ft 6 in. (P680-8A1-C2)	93 ft 6 in. (P870-10A-G2)
RCIC RM	90 ft 6 in. (P680-8A1-C4)	93 ft 6 in. (P870-2A-A1)
LPCS RM	90 ft 6 in. (P680-8A1-A4)	93 ft 6 in. (P870-2A-F1)
HPCS RM	90 ft 6 in. (P680-8A1-B4)	93 ft 6 in. (P870-5A-H1)

Examination Outline Cross Reference	Level	RO
295035 Secondary Containment High Differential Pressure EK2. Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: EK2.01 Secondary containment ventilation	Tier	1
	Group #	2
	K/A	295035 EK2.01
	Rating	3.6
	Revision	0
Revision Statement:		

Question: 26

Auxiliary Building Ventilation (T41) and Fuel Handling Area Ventilation (T42) are in service when a controller malfunction causes the Pressure Control Damper T42-F021 to partially stroke open (from its previous position).

Which of the following describes the response of Auxiliary Building d/p, and the reason for that response?

Aux Building d/p becomes…

A. more negative because air is being exhausted from the building at a higher flowrate.

B. more negative because outside air is being supplied to the building at a higher flowrate.

C. less negative because air is being exhausted from the building at a higher flowrate.

D. less negative because outside air is being supplied to the building at a higher flowrate.

Answer: D
Explanation: F021 is at the inlet to the supply fans. With all else remaining constant, if F021 opens more, a greater supply air flowrate occurs, which lessens the building d/p. 'D' is correct

Distracters:

‘A’ is wrong – Plausible, if F021 opens more, a greater supply air flowrate occurs, which lessens the building d/p. F021 is at the inlet to the supply fans

‘B’ is wrong – Plausible, if F021 opens more, a greater supply air flowrate occurs, which lessens the building d/p.

‘C’ is wrong – Plausible, F021 is at the inlet to the supply fans

K/A Match

This question requires the applicant to have the knowledge of secondary containment ventilation and the interrelations with building D/P.

Technical References:

02-S-01-40, EP Technical Bases, Rev 009
05-S-01-EP-4, Auxiliary Building Control, Rev. 31

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-T4200 Objective: 3.1

Question Source:	**Bank # 487**	X Not used on NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Lesson Content	Instructor Notes

2 The Fuel Handling Area Ventilation System is designed to maintain areas served by the system between 65ºF and 80ºF at 50% relative humidity.

EO-2

These Fuel Handling Area design air temperatures are based on an assumed outside air temperature between 15º and 95ºF.

2 Redundant supply and exhaust fans maintain the fuel handling area at a slightly negative pressure with respect to surrounding areas, and thus maintains a controlled air flow from areas of lower to higher radioactivity.

SYSTEM OVERVIEW

The Fuel Handling Area Ventilation System serves Zone 5 of the Auxiliary Building, elevations 185′ and 208′.

The Fuel Handling Area Subsystem consists of two 100 percent capacity fuel handling area supply and exhaust fans, a supply air heating coil, and three 100 percent capacity fuel handling area fan coil units.

The Fuel Pool Sweep Subsystem includes two 100 percent capacity fuel pool sweep supply and exhaust fans, and three 100 percent capacity heating coils: an outside air heating coil, a preheat coil, and a reheat coil.

Fuel Handling Area Subsystem

The two Fuel Handling Area Supply Fans supply makeup air to the fuel handling area.

Figure 1

Outside air is drawn through a prefilter and discharged to the fuel handling area through a supply air heating coil.

3.1 A pressure control system modulates a damper in the supply air ductwork to regulate the air flow and maintain the fuel handling area at a slightly negative pressure.

EO-3.1

3.2 In conjunction with the inlet air control for the supply fans, the two Fuel Handling Area Exhaust Fans ventilate and maintain the fuel handling area at a slightly negative pressure with respect to the outside and surrounding areas.

EO-3.2

3.2 This slightly negative pressure helps control contamination by ensuring that airborne activity, if present, does not leak to surrounding areas or the environment.

Lesson Content	Instructor Notes
6.2.2 Three-position STOP/STDBY/START handswitches and red/green indicating lights for Fuel Handling Area Exhaust Fans C002A and B are located on Control Room Panel P842.	EO-6.2.2
8.1 An exhaust fan with its handswitch in STDBY automatically starts on loss of the operating exhaust fan as sensed by fan low differential pressure.	EO-8.1
8.1, 8.2 The fuel handling area exhaust fan inlet and outlet dampers, F006A/B and F005A/B, open and close when the associated fan is started and stopped. Red/green indicating lights are also provided for the dampers on Control Room Panel P842.	EO-8.1,8.2
8.1 Exhaust fans C002A and B will automatically trip if either of the secondary containment isolation dampers, F003 or F004, is not full open (may be caused by SBGT System initiation).	EO-8.1
	Ask Student Question #2
Fuel Handling Area Pressure Control	
10.1 The Fuel Handling Area Pressure Control System has two differential pressure transmitters that measure the difference in air pressure between the outside and the fuel handling area.	EO-10.1
6.2.3 The signal from these transmitters is sent to a two-pen recorder R602 and to a three-position CHAN A/AUTO/CHAN B handswitch on Control Room Panel P842.	EO-6.2.3
The differential pressure transmitters also provide signals to a high value selector. The higher signal is the highest absolute pressure, e.g., if the two differential pressure transmitters provided outputs equivalent to –0.20″ wc and –0.25″ wc, the –0.20″ wc signal would be the high value. The high value selector passes the higher of the two signals on to the three-position handswitch. The higher signal is selected with the handswitch in AUTO.	Figure 4
The two transmitter inputs to the handswitch are selected in the CHAN A and CHAN B positions.	Alarm window P842-1A-E4 and logic removed IAW EC-11966. Windy conditions caused erroneous alarm.
6.2.3 The differential pressure signal, as determined by the handswitch position and high value selector, is sent to a differential pressure controller R600 on Control Room Panel P842.	EO-6.2.3

Lesson Content	Instructor Notes

The differential pressure signal is compared to the controller tape setpoint, normally set at 25, to generate the controller output signal which varies the position of the air-operated control damper F021 on the inlet to the fuel handling area supply fans.

If the sensed differential pressure is higher than the setpoint on the differential pressure controller, the damper opens to admit more air.

If the sensed differential pressure is lower, the differential pressure controller closes down on the damper to admit less air.

6.2.3 The set point can be varied from the controller located on Control Room Panel P842.

EO-6.2.3

If the Fuel Handling Area Ventilation System is unable to maintain a negative pressure of >0.1″ wc, annunciator FUEL HANDLING AREA PRESSURE LOW alarms.

Ask Student Question #4

Fuel Handling Area Subsystem Fan Coil Units

Annunciator P842-1A-E3

5.2.5, 9.1.3 Fuel Handling Area Fan Coil Units B002, B005 and B007, located in the Auxiliary Building on elevations 245′, Area 9; 185′, Area 10; and 139′, Area 9; respectively, are all powered from MCC 12B52.

EO-5.2.5, 9.1.3

3.3, 4.2.5 The fan coil units recirculate air and control the temperature in the areas they serve.

EO-3.3, 4.2.5

Each unit contains a cooling coil which is cooled by plant chilled water.

Units B002 and B005 also contain an electric heating coil.

3.3, 10.2 For each of the fan coil units, a local temperature controller senses the temperature of the return air to the fan coil unit.

EO-3.3, 10.2

If cooling is required, the controller modulates the position of a plant chilled water supply valve to adjust the air temperature.

♫ Monitoring – Identify degrading parameter and equipment trends.

3.3, 10.2 When heating is required, a local temperature switch for fan coil units B002 and B005 energizes the associated heating coil to warm the air.

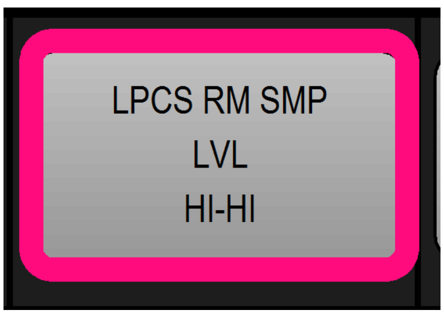
EO-3.3, 10.2

The heating coil trips on low air flow in the fan coil unit.

Examination Outline Cross Reference	Level	RO
295036 Secondary Containment High Sump/Area Water Level 2.1.30 Ability to locate and operate components, including local controls.	Tier	1
	Group #	2
	K/A	295036 2.1.30
	Rating	4.4
	Revision	0
Revision Statement:		

Question: 27

The plant is at rated conditions when the following alarm is received.



(1) Which of the following describes the crew's response from this alarm?

(2) What location will the actions be performed?

A. (1) Enter EP-4, Operate all sump pumps
(2) LPCS room

B. (1) Enter EP-4, Operate all sump pumps
(2) Radwaste Control Room

C. (1) Enter ARI P680-8A1-A4, Manually start sump pumps
(2) LPCS room

D. (1) Enter ARI P680-8A1-A4, Manually start sump pumps
(2) Radwaste Control Room

Answer:	A
Explanation: The alarm given is outlined in "glowing cerise" color designating that this alarm corresponds to an EP-4 entry condition. EPs have precedence over ARIs, therefore EP-4 should be entered first and actions performed.	

The action per EP-4 is to "Operate sump pumps as necessary to restore and maintain area water levels below their operating limits" LPCS sump pump handswitches are located inside the LPCS pump room. 'A' is correct
Distracters: 'B' is wrong – Plausible, Radwaste control room contains controls for multiple sump systems and the Auxiliary transfer pump but not room sump pump controls, those are local 'C' is wrong – Plausible, EPs have precedence over ARIs 'D' is wrong – Plausible, EPs have precedence over ARIs. Radwaste control room contains controls for multiple sump systems and the Auxiliary transfer pump but not room sump pump controls, those are local
K/A Match This question requires the applicant to have the knowledge of secondary containment high sump water levels and the actions required. Also must have the knowledge of local controls.
Technical References: 02-S-01-40, EP Technical Bases, Rev 009 05-S-01-EP-4, Auxiliary Building Control, Rev. 31 04-1-01-P45-2, Floor Drain Sump System SOI, Rev. 24
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-EP4 Objective: 5 & 9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Table SC-1: Aux Building Area Parameters

Table SC-1
Aux Building Area Parameters

Area	Operating Limit	Max Safe Value
TEMPERATURE		
MSL PIPE TUNNEL TEMP	185°F (P601-19A/18A-A3/A4)	250°F (E31-N604A,B,C,D,E,F)
RHR-A EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608A, N610A)
RHR-B EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608B, N610B)
RCIC EQUIP AREA TEMP	185°F (P601-21A-G3)	212°F (E31-N602A/B)
RWCU-PUMP ROOM 1 TEMP	170°F (P680-11A-A1)	NA
RWCU-PUMP ROOM 2 TEMP	170°F (P680-11A-A2)	NA
HVAC EXHAUST RADIATION LEVEL		
AUX BLDG FHA VENT EXHAUST	3.6 mr/hr (P601-19A-B9/C9)	NA
AUX BLDG FHA POOL EXHAUST	30 mr/hr (P601-19A-B10/C10)	NA
RADIATION LEVEL		
RHR ROOM A	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR ROOM B	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR HX A HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RHR HX B HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RCIC ROOM	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
MAIN STEAM LINE RAD MONITOR	Set Point Log (P601-19A-D4)	8×10^4 mr/hr
SGTS FLTR TRN	2.5 mr/hr (P844-1A-C5)	8×10^2 mr/hr
WATER LEVEL		
RHR RM A	89 ft 8 in. (P680-8A1-A2)	93 ft 6 in. (P870-2A-E1)
RHR RM B	89 ft 8 in. (P680-8A1-B2)	93 ft 6 in. (P870-10A-G1)
RHR RM C	90 ft 6 in. (P680-8A1-C2)	93 ft 6 in. (P870-10A-G2)
RCIC RM	90 ft 6 in. (P680-8A1-C4)	93 ft 6 in. (P870-2A-A1)
LPCS RM	90 ft 6 in. (P680-8A1-A4)	93 ft 6 in. (P870-2A-F1)
HPCS RM	90 ft 6 in. (P680-8A1-B4)	93 ft 6 in. (P870-5A-H1)

Step 4

WATER LEVELS

Operate sump pumps as necessary to restore and maintain area water levels below their operating limits (Table SC-1).

IF	THEN
4 Any area water level cannot be restored and maintained below its operating limit (Table SC-1)	Go to 21

Discussion

Operate sump pumps

The first method used to control secondary containment water level is that used during normal plant operations—operation of the sump pumps. This step thus provides a smooth transition from general plant procedures to emergency operating procedures and ensures that normal procedures are tried before more complex actions.

Any area water level cannot be restored and maintained below its operating limit

If an area water level cannot be maintained below its maximum normal operating value, a primary system break should be suspected. Any break should then be isolated in accordance with Step 5 and the need for additional action evaluated in accordance with Steps 6 and 7.

System Floor Drain Sump System MPL No. P45-2

Checksheet Name Remote Operated Pump Lineup Checksheet

Instruction Step 4.1.1.d

PUMP NO.	PUMP DESCRIPTION	SWITCH NO.	PANEL NO.	REQ POS	PUMP DEV	INITIALS 1st ck
	AUXILIARY BLDG 93’					
C017A	HPCS Rm Flr Dr Smp Pmp "A"	HS-M025A	HSSM024	AUTO		
C017B	HPCS Rm Flr Dr Smp Pmp "B"	HS-M025B	HSSM024	AUTO		
C017A/ C017B	HPCS Rm Flr Dr Smp Pmp's A/B Mode Switch	HSS-M024	HSSM024	ALTERNATE		
C005A	Aux Bldg North Flr Dr Smp Pmp "A"	HS-M008A	HSSM009	AUTO		
C005B	Aux Bldg North Flr Dr Smp Pmp "B"	HS-M008B	HSSM009	AUTO		
C005A/ C005B	Aux Bldg-North Flr Dr Smp Pmp's A/B Mode Switch	HSS-M009	HSSM009	ALTERNATE		
C015A	RHR Rm C Flr Dr Smp Pmp "A"	HS-M020C	HSSM019C	AUTO		
C015B	RHR Rm C Flr Dr Smp Pmp "B"	HS-M021C	HSSM019C	AUTO		
C015A/ C015B	RHR Rm C Flr Dr Smp Pmps A/B Mode Switch	HSS-M019C	HSSM019C	ALTERNATE		
C016A	LPCS Rm Flr Dr Smp Pmp "A"	HS-M023A	HSSM022	AUTO		
C016B	LPCS Rm Flr Dr Smp Pmp "B"	HS-M023B	HSSM022	AUTO		
C016A/ C016B	LPCS Rm Flr Dr Smp Pmp's A/B Mode Switch	HSS-M022	HSSM022	ALTERNATE		
C007A	Aux Bldg South Flr Dr Smp Pmp "A"	HS-M015A	HSSM016	AUTO		
C007B	Aux Bldg South Flr Dr Smp Pmp "B"	HS-M015B	HSSM016	AUTO		
C007A/ C007B	Aux Bldg-South Flr Dr Smp Pmp's A/B Mode Switch	HSS-M016	HSSM016	ALTERNATE		
	TURBINE BLDG EAST 93’					
C011A	Turb Bldg East Flr Dr Smp PMP"A"	HS-M040A	HSSM039	AUTO		
C011B	Turb Bldg East Flr Dr Smp Pmp "B"	HS-M040B	HSSM039	AUTO		
C011A/ C011B	Turb Bldg East Flr Dr Smp Pmps A/B Mode Switch	HSS-M039	HSSM039	ALTERNATE		
C034	Turb Bldg East Flr Dr Stby Smp Pmp	HS-M041	HSSM041	AUTO		

Date ______________

Examination Outline Cross Reference	**Level**	**RO**
203000 RHR/LPCI: Injection Mode (Plant Specific)	**Tier**	2
	Group #	1
A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**K/A**	203000 A2.03
	Rating	3.2
A2.03 Valve closures	**Revision**	0
Revision Statement:		

Question: 28

The crew is currently performing actions to mitigate a LOCA.

Current conditions

- Drywell Pressure 1.56 psig
- RPV water level -20 inches and rising by using HPCS and RCIC
- All Low Pressure ECCS Injection valves are OVERRIDDEN CLOSED per EP-2

The LOCA gets worse and level begins to lower.

CRS orders an Emergency Depressurization at -160” wide range

(1) Which of the following describes the impact on the RHR systems ability to automatically inject into the RPV?

(2) What will be the crew’s action?

A. (1) NOT available to AUTO inject
(2) Manually OPEN injection valves using Handswitch.

B. (1) Will automatically inject at < 476 psig reactor pressure
(2) Verify automatic injection valve opening for all low pressure ECCS systems

C. (1) NOT available to AUTO inject
(2) Depress the RHR A/LPCS and RHR B/RHR C Manual Initiation Pushbuttons

D. (1) Will automatically inject when -150.3 inches is reached
(2) Verify automatic injection valve opening for all low pressure ECCS systems

Answer: A

Explanation:

When Low Pressure ECCS systems are overridden all automatic actions of those components are blocked. By taking the injection valves to overridden closed no auto actions will occur on those valves. Manual opening is required for injection.

'A' is correct

Distracters:

'B' is wrong – Plausible, normal action without any overriding, however, the student sees another auto initiation signal present.

'C' is wrong – Plausible, depressing the auto initiation pushbuttons has no effect on the overridden valves.

'D' is wrong – Plausible, if the student is confused about another initiation signal being received then this is plausible.

K/A Match

This question requires the applicant to predict the impact on the LPCI system with a valve closure and using procedures take actions.

Technical References:

04-1-01-E12-1, RHR SOI , Rev. 154

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200 Objective: 8.9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:
RHR is listed as #3 on the System Importance to CDF.

3.2.8 **CLEAN** RHR suction strainers when after-start suction pressures reach values shown below:

SYSTEM	SUCTION PRESSURE AT DESIGN FLOW	DESIGN FLOW
RHR A	4.0 psig	7450 gpm
RHR B	4.5 psig	7450 gpm
RHR C	5.0 psig	7450 gpm

3.2.9 The RHR pump room coolers (T51) are required for operability of the respective RHR pump.

3.2.10 The RHR pump seal cooler is only required for operability of Shutdown Cooling Mode of RHR with reactor coolant temperature above 200°F. RHR pump seal cooler is inoperable **WHEN** SSW flow through the cooler is less than 4.5 gpm (0.13”H20) **AND** reactor coolant temperature is above 200°F.

3.3 **WHEN** MANUAL OVERRIDE is activated on a pump, **OR** loop injection, **OR** test return valve, AUTO features are disabled until initiation signals are RESET **OR** if power is lost to the ESF Bus. The Operator May manipulate the pump **AND/OR** valve **USING** the component handswitch. Manual Override is defeated if a Containment Spray signal is received.

3.4 On automatic initiation of the RHR System in the LPCI mode of operation, the RHR Injection valves (E12-F042A, B, C) **DO NOT** receive an Open permissive until RPV pressure decreases to approximately 476 psig. However, the RHR Injection valves May be manually overridden CLOSED before reaching the open permissive setpoint.

Also, on a LPCI initiation, the E12-F048A/B receive an interlock-to-open signal for 3.41 minutes. Attempting to Close E12-F048A/B prior to 3.41 minutes after LPCI initiation May cause respective valve breaker to trip.

3.5 Unless misoperation in Automatic is confirmed by at least two independent indications **OR** Adequate Core Cooling is ensured, ECCS Shall **NOT** be placed in MANUAL during operation.

IF in manual, frequent **CHECKS** of initiating or controlling parameters Shall be made.

Unless there are at least two independent indications that Adequate Core Cooling is ensured, ECCS Shall **NOT** be secured during operation.

3.6 For F003A(B) **AND** F048A(B):

3.6.1 Do **NOT** throttle F003A(B) below 3000 gpm unless the E12-F048A(B) is full open. For throttling restrictions for E12-F003A(B) and E12-F048A(B) in Shutdown Cooling Mode, **REFER** to 04-1-01-E12-2.

3.6.2 Do **NOT** reduce RHR A(B) flow below 7300 gpm when throttling with F048A(B). There are no restrictions on throttling F048A(B) when F003A(B) is full open.

Examination Outline Cross Reference	Level	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) K3. Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: K3.01 Reactor pressure	Tier	2
	Group #	1
	K/A	205000 K3.01
	Rating	3.3
	Revision	1
Revision Statement: **Per Ops Rep, Changed stem to state when effect will this have on pressure control.**		

Question: 29

The plant is in Mode 3 with MSIVs closed

RHR B was started a few moments ago in Shutdown Cooling

Reactor pressure is currently 30 psig

A spurious Group 3 isolation has occurred.

Reactor pressure will be controlled and maintained by…

A. SRVs

B. initiating RCIC.

C. the Turbine building Main Steam line Drains

D. the Main Turbine Bypass Valves

Answer: **A**
Explanation: When in Mode 3 decay heat is still high and at 30 psig the Reactor is still very hot. Any loss of cooling will cause reactor pressure to rise until acted upon When shutdown with MSIVs closed the only pressure control is SRVs. 'A' is correct
Distracters:

‘B’ is wrong – Plausible, even though RCIC can be used as level and pressure control, at <60 psig reactor pressure RCIC is isolated and unavailable.

‘C’ and ‘D’ are wrong – Plausible, Reactor pressure will rise, steam line drains and bypass valves are unavailable due to MSIVs are closed.

K/A Match

This question requires the applicant to have the knowledge of the effects on reactor pressure during a loss of shut down cooling.

Technical References:

GLP-OPS-E1200 Rev 10
03-1-01-4, Rev 117

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200 Objective: 8.9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

RHR is listed as #3 on the System Importance to CDF.

SENIOR REACTOR OPERATOR INITIALS/DATE

6.0 SCRAM RECOVERY WITH MSIVs CLOSED

6.1 **VERIFY** on-scale Neutron Monitoring is established on the SRMs **AND/OR** IRMs. _________/_______

CAUTION

The first reading taken Must be Pre-Scram data. (Steady state conditions immediately prior to SCRAM)

6.1.1 Per Tech Spec SR 3.4.11.1 **BEGIN TAKING** temperature Data on Data Sheet I. **USE** Data Sheet II as an aid in **MONITORING** temperature trend. **RECORD** pre-scram Data as first reading. **VERIFY** at least once per 30 minutes that the cooldown rate is less than 90°F/hr **AND** that the reactor vessel metal temperature **AND** reactor coolant pressure are **determined** to be to the right of the limit lines of curve B on Figure 1 of the PTLR per Technical Specification 3.4.11. _________/_______

6.2 **MAINTAIN** RPV pressure **AND** level as follows:

6.2.1 To maintain RPV pressure, **USE** the following means (listed in order of preference):

a. RCIC turbine in-service per SOI 04-1-01-E51-1. _________/_______

6.2.1 (Cont)

SENIOR REACTOR OPERATOR INITIALS/DATE

NOTE

Suppression Pool Cooling Should be placed in service before opening Relief Valves **OR** as soon as possible after opening Relief Valves

Relief Valves which are known to be leaking/weeping **OR** Will be changed out the next Refueling Outage are designated by a colored key **AND** are preferred to be used for maintaining RPV pressure **WHEN** Suppression Pool Cooling is in service.

b. **OPEN AND CLOSE** SRVs to remain in pressure band:

NOTE

INITIATE a Condition Report(CR) if SRVs are opened.

(1) **IF** Suppression Pool Cooling is in service, **THEN USE** any known leaking/ weeping SRV to reduce **OR** maintain RPV pressure **IF** there are none leaking **OR** weeping **THEN USE** any Non-ADS SRVs relief valve to control RPV pressure. ______/______

(2) **IF** Suppression Pool Cooling is **NOT** in service **THEN** operation of the Relief Valves Should be rotated to allow for even heating of the Suppression Pool while reducing **OR** maintaining RPV pressure. **USE** Figure A as a guide. ______/______

3.0 SUBSEQUENT OPERATOR ACTIONS

CAUTION

IF unable to restore adequate decay heat removal before reaching Mode 3, ADHRS Must be isolated by closing E12-F066A, E12-F066B, E12-F410, AND E12-F424.

3.1 **IF** forced circulation **OR** cooling to Reactor coolant is lost **OR** degraded, **THEN MONITOR** Reactor coolant temperature **AND** Reactor vessel pressure.

NOTE

Graphs in Attachment I are derived from decay heat analysis for use in a specific refueling outage. **WHEN** used at other times, the graphs provide guidance but May **NOT** be accurate.

3.1.1 **IF** secondary containment is **NOT** established, **THEN ESTABLISH** secondary containment within Attachment I applicable time to 200°F curve.

a. **IF** any Secondary Containment penetrations are identified open, **THEN NOTIFY** appropriate person/department to immediately isolate penetration **OR** implement contingency measures to close penetration opening.

b. **IF** penetration **CANNOT** be closed, **THEN IMPLEMENT** measures to minimize penetration opening.

c. For any identified hoses/cables blocking penetration opening, **EVALUATE** impact to plant prior to removing **OR** disconnecting. **NOTIFY** affected department for assistance.

d. **PERFORM** 06-OP-1T10-M-0001 to close Secondary Containment doors **AND ENSURE** identified penetrations are closed.

Examination Outline Cross Reference	Level	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) K6. Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : K6.08 RHR service water: Plant-Specific	Tier	2
	Group #	1
	K/A	205000 K6.08
	Rating	3.5
	Revision	0
Revision Statement:		

Question: 30

The plant is in Mode 4

RHR A is operating in Shutdown Cooling

Standby Service Water Pump 'A' trips.

What actions will be performed to maintain control of the plant?

A. Raise reactor water level to +82".

B. Raise reactor water level to between +101" and +129".

C. Start LPCS and inject to maintain 55 psig in the reactor.

D. Place RHR Pump 'B' and SSW pump 'B' in Shutdown Cooling.

Answer: D
Explanation: Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, "**IF** inservice loop of Shutdown Cooling has been lost, **THEN PLACE** redundant loop in Shutdown Cooling **OR** ADHRS in service per SOI 04-1-01-E12-2." Just a loss of the SSW system causes a loss of cooling water flow to the RHR heat exchangers, therefore a loss of cooling. 'D' is correct
Distracters:

‘A’ is wrong – Plausible, Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, **<u>IF</u>** forced circulation (no Reactor Recirculation pumps running) has been lost, **<u>THEN</u> RAISE** Reactor water level to 615" actual (this is 82" above instrument zero). This would be the action of both Recirc pumps were lost. ‘B’ is wrong – Plausible, Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, **<u>IF</u>** adequate cooling **<u>CANNOT</u>** be maintained **<u>OR</u>** restarted within the estimated time to 200°F, **<u>THEN</u> PERFORM** the following: **SLOWLY RAISE** RPV water level using any available injection systems to between +101 inches **<u>AND</u>** +129 inches to establish flow through open SRVs back to Suppression Pool. ‘C’ is wrong - Plausible, Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, this step is performed after the above step in ‘B’ is performed: **<u>IF</u>** RPV pressure does **<u>NOT</u>** stabilize above 55 psig, **<u>THEN</u> START** additional RHR pump(s) **<u>AND</u> OPEN** its associated Injection Valve (E12-F042A,B,C). **<u>OR</u>** **START** LPCS pump **<u>AND</u> OPEN** Injection Valve E21-F005 (1H13-P601) until RPV pressure is above 55 psig.
K/A Match This question requires the applicant to have the knowledge of the effects during a loss of Service Water for shut down cooling.
Technical References: 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, Rev 46
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-E1200 Objective: 13.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		

RHR is listed as #3 on the System Importance to CDF.

NOTE

Potential leak sources Could be from temporary plugs (FW, Jet pump, MSL, , etc.), RHR pump min flow valves (associated alarm), reactor cavity to drywell seals, freeze seals, LLRTs, tagging **OR** similar evolutions in progress **OR** recently completed, etc. Potential makeup sources include P11, CRD, ECCS, N19, N21, P64, etc.

3.3 **IF** water level in Containment **OR** Auxiliary Building pools has abnormally lowered, **THEN DETERMINE** cause **AND REFILL**.

3.4 Decay Heat Removal for Fuel within Reactor Vessel with RPV Head Installed

3.4.1 **IF** inservice loop of Shutdown Cooling has been lost, **THEN PLACE** redundant loop in Shutdown Cooling **OR** ADHRS in service per SOI 04-1-01-E12-2 [SSD].

a. **IF** a MOV has failed closed, **THEN REOPEN** by **PERFORMING** the following:

(1) **IF** the valve failed closed due to spurious isolation signal, **THEN RESTORE OR DEFEAT** the isolation signal.

(2) **IF** the isolation signal **CANNOT** be restored **OR** defeated, **THEN OPEN** the associated breaker **AND MANUALLY OPEN** the valve.

Examination Outline Cross Reference	Level	RO
209001 Low Pressure Core Spray System K5. Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : K5.05 System venting	Tier	2
	Group #	1
	K/A	209001 K5.05
	Rating	2.5
	Revision	0
Revision Statement:		

Question: 31

LPCS system is operating in test return to the suppression pool.

Power feeding the 15AA Bus is lost and restored by the emergency DG.

(1) What is required to restore the system?

(2) What implications would occur if not performed?

A. (1) Jockey pump running only
(2) System damage could occur

B. (1) System filled and vented
(2) System damage could occur

C. (1) Jockey pump running only
(2) Injection time is shortened

D. (1) System filled and vented
(2) Injection time is shortened

Answer: B
Explanation: When the 15AA bus is lost, all pump and MOV power is lost. The LPCS pump will trip and all MOVs will fail as is. The Test return valve being open to the suppression pool and the pump not running causes the system to drain to the suppression pool. Per 04-1-01-E21-1, LPCS System Operating Instruction Precaution 2. System damage could occur if LPCS pump is started without discharge line filled and pressurized. 'D' is correct

Distracters:

‘A’ is wrong – Plausible, the jockey pump is required to maintain the system pressurized but after system draining a system fill and vent is required.

‘C’ is wrong – Plausible, the jockey pump is required to maintain the system pressurized but after system draining a system fill and vent is required and the purpose of the jockey pump is to shorten the injection time by keeping the system pressurized.

‘D’ is wrong - The purpose of the jockey pump is to shorten the injection time by keeping the system pressurized and not the most concern.

K/A Match

This question requires the applicant to have the knowledge of the effects during a loss of Service Water for shut down cooling.

Technical References:

04-1-01-E21-1, LPCS SOI, Rev 44

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2100 Objective: 4.5, 7.0, 11.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

1.0 PRECAUTIONS AND LIMITATIONS

1.5 Precautions

1. When MANUAL OVERRIDE is actuated, automatic pump restart or valve opening is disabled until initiation signals are reset, or if power is lost to ESF bus and then restored.

2. System damage could occur if LPCS pump is started without discharge line filled and pressurized.

3. Unnecessary injection of suppression pool water to Reactor is to be avoided

4. LPCS pump damage could occur pump run out flow of 9100 gpm is exceeded.

5. Division 1 suppression pool level instrumentation reference legs are maintained full by LPCS jockey pump.

6. During non-emergency conditions MOV damage could occur if operated with LPCS MOV TEST switch on 1H13-P601-21B NOT in TEST.

7. LPCS MOV TEST switch on 1H13-P601-21B should remain in TEST position for 30 seconds following valve operation.

8. At no time during operation is ECCS to be placed in MANUAL by an Operator unless misoperation in AUTOMATIC is confirmed by at least two independent indications or adequate core cooling is ensured.

9. While in MANUAL, frequent checks of the initiating or controlling parameters are to be made. At no time during operation is ECCS is be secured by an Operator, unless there are at least two independent indications that adequate core cooling is ensured.

10. On an automatic initiation (-150.3" Vessel Level or 1.39 psi Drywell Pressure) of LPCS System, 1E21F005, LPCS INJ SHUTOFF VLV does NOT receive an open permissive until RPV pressure decreases to 476 psig.

11. The LPCS Injection Valve 1E21-F005 has open permissive from Control Room handswitch whenever pressure is less than 530 psig between 1E21-F005 and Testable Check Valve 1E21-F006.

12. Suppression Pool Cleanup may be required following operation of LPCS System on Suppression Pool.

Examination Outline Cross Reference	Level	RO
209001 Low Pressure Core Spray System 2.2.40 Ability to apply Technical Specifications for a system.	Tier	2
	Group #	1
	K/A	209001 2.2.40
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 32

Which of the following would require Low Pressure Core Spray (LPCS) system to be declared inoperable?

A. LPCS MOV Test Switch in TEST for 4 hours

B. LPCS flow indication on H13-P601 stuck downscale

C. LPCS Pump Discharge Flow trip unit E21-FIS-N651 failed upscale causing E21-F011, LPCS MIN FLOW valve to close.

D. Red bulb socket above the handswitch for LPCS INJ MOV E21-F005 on H13-P601 non-functional

Answer: C

Explanation:

All answers are plausible since they deal with LPCS components. The designated answer is correct since per SOI 04-1-01-E21-1 Section 5.1, step 2, the minimum flow valve must be capable of performing its intended function, otherwise, LPCS must be declared inoperable. With the flow switch failed upscale, min flow valve E21F011 would automatically close, which is not its standby line up, and would not automatically open when required.

Distracters:

'A' is incorrect, because TRM 6.8.2 allows the MOV Test switches to be in test for up to 8 hours before valve operability is affected.

'B' and 'D' are incorrect, because these components are not required for LPCS to perform its safety function.

K/A Match

This question requires the applicant to recognize the into Tech Specs.

Technical References:

04-1-01-E21-1, LPCS SOI, Rev 44 Tech Specs 6.8.2		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-E2100 Objective: 7.0, 11.1, 13.0		
Question Source:	**Bank # 905**	2007 NRC Exam Q 31
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

5.0 INSTRUCTIONS

5.1 Conditional Actions

1. **IF AT ANY TIME** problems develop at any time during performance of procedure,
   THEN NOTIFY Shift Supervision.

2. **IF AT ANY TIME** Pump min flow valve for LPCS will NOT perform its intended function,
   THEN DECLARE LPCS system INOPERABLE.

3. **IF AT ANY TIME** Suppression Pool Level is greater than 14.5',
   THEN Adequate NPSH for the LPCS pump exists.

4. **IF AT ANY TIME** LPCS System is OPERABLE,
   THEN ENSURE LPCS pump room cooler (T51) is required.

5. **IF AT ANY TIME** MOV thermal overloads in force (Test switch in TEST) for longer than 8 hours,
   THEN:

   PLACE Test switches to NORM

   OR

   DECLARE associated valves Inoperable PER TRM 6.8.2

6. **IF AT ANY TIME** LPCS pump motor breaker is racked-in,
   THEN PERFORM LPCS Pump Breaker Operability Check.

6.8 ELECTRICAL POWER SYSTEMS

6.8.2 MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LCO 6.8.2 The thermal overload protection of each valve whose motor operator performs a safety function, shown in Table 6.8.2-1, shall be FUNCTIONAL or shall be bypassed either continuously or only under accident conditions, as indicated, by an FUNCTIONAL bypass device.

APPLICABILITY: When the motor operated valve is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The thermal overload protection for one or more of the above required valves not FUNCTIONAL or not bypassed either continuously or only under accident conditions, as applicable.	A.1 Bypass the thermal overload. OR A.2 Declare the affected valve(s) nonfunctional.	8 hours 8 hours

Examination Outline Cross Reference	Level	RO
209002 High Pressure Core Spray System K5. Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): K5.04 Adequate core cooling: BWR-5,6 .	Tier	2
	Group #	1
	K/A	209002 K5.04
	Rating	3.8
	Revision	0
Revision Statement:		

May 2017 NRC Exam Q 32

Question: 33

Which of the following assures heat removal from the reactor is sufficient to prevent failure of the fuel clad when using High Pressure Core Spray system?

A. Flow rate of 550 gpm and Reactor Water level is -205”

B. Flow rate of 550 gpm and Reactor Water level is -220”

C. Flow rate of 7115 gpm and Reactor Water level is -215”

D. Flow rate of 7400 gpm and Reactor Water level is -225”

Answer: C

Explanation:

Adequate core cooling is defined as HPCS OR LPCS flow above 7000 gpm and Reactor water level above -217” (spray cooling)

Distracters:

‘A’ is wrong – Flow rate is too low, level is not high enough for Minimum Zero Injection RPV Water Level of -204”

‘B’ is wrong – flow rate and level is below min required. Plausible due to 550 gpm is the design flow for HPCS at high pressure.

‘D’ is wrong – flow rate is good however, water level is too low. Plausible due to 7400 is rated flow for RHR pumps.

K/A Match

This question requires the applicant to recognize the into Tech Specs.

Technical References: 04-1-01-E22-1 Rev 121 GLP-OPS-E2200 Rev 10 02-S-01-43 Rev 3		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-EP01 Objective: 4a		
Question Source:	**Bank # 1278**	**May 2017 NRC Exam Q 32**
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

4.0 ATTACHMENTS

None

5.0 DEFINITIONS

5.1 Adequate Core Cooling - Is assured whenever any of the following conditions exist:

5.1.1 Reactor water level is at **OR** above -167 inches (Top of Active fuel). This is the preferred condition.

5.1.2 Reactor water level is at **OR** above -191 inches (Minimum Steam Cooling RPV Water Level).

5.1.3 Reactor water level is at **OR** above -204 inches (Minimum Zero Injection RPV Water Level) without RPV injection.

5.1.4 RPV pressure is at **OR** above the Minimum Steam Cooling Pressure (MSCP).

5.1.5 HPCS **OR** LPCS flow above 7000 gpm **AND** Reactor water level above -217 inches (Spray Cooling).

Examination Outline Cross Reference	Level	RO
211000 Standby Liquid Control System A1. Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: A1.06 Flow indication: Plant-Specific	Tier	2
	Group #	1
	K/A	211000 A1.06
	Rating	3.8
	Revision	3
Revision Statement: **Rev 1 - Feedback – added "Per Att. 6."** **Rev 2 - Removed pump running reference**		

Question: 34

An ATWS is in progress at >5% power.

SLC has been initiated.

04-1-01-C41-1, Standby Liquid Control SOI, Attachment 6, Verification of Standby Liquid Control Injection is being performed.

Per Attachment 6, **CHECK** SLC INJECTING INTO the RPV by observing which of the following?

1. Discharge pressure greater than reactor pressure
2. Tank level lowering
3. Squib valves fired
4. Nuclear Instrumentation lowering

A. 2, 3, 4

B. 1, 2, 3

C. 1, 3, 4

D. 1, 2, 4,

Answer: **D**
Explanation: Per 04-1-01-C41-1, Standby Liquid Control SOI, Attachment 6, Verification of Standby Liquid Control Injection The positive indications that SLC is injecting into the reactor is SLC pump discharge pressure above reactor pressure and tank level lowering and Nuclear Instrumentation lowering.

'D' is correct.

Distracters:

'B' is wrong – Squib valve firing is not required due to indication may not be reliable. Plausible due to it is listed for checking system initiation but not for injection verification.

'C' is wrong – Pump running indication is not required if discharge pressure is above reactor pressure, a pipe break could have occurred and even though the pump is running it is not injecting into the reactor. Plausible due to it is listed for checking system initiation but not for injection verification.

'D' is wrong – Pump running indication is not required if discharge pressure is above reactor pressure, a pipe break could have occurred and even though the pump is running it is not injecting into the reactor. Plausible due to it is listed for checking system initiation but not for injection verification and Squib valve firing is not required due to indication may not be reliable. Plausible due to it is listed for checking system initiation but not for injection verification.

K/A Match

This question requires the applicant to have the ability to verify SLC flow is injecting into the reactor from multiple indications.

Technical References:

04-1-01-C41-1 Rev 125

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C4100 Objective: 12

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Attachment 6 **Page 1 of 2**

Verification of Standby Liquid Control Injection

EMERGENCY USE ONLY

1. **CHECK** system initiation by **OBSERVING** following:

   a. CHECK Squib Valves fired on following:

      1) 1C41-F004A, SLC PUMP C001A DISCHARGE SQUIB VALVE

         White SQUIB VLV READY light OFF

      2) 1C41-F004B, SLC PUMP C001B DISCHARGE SQUIB VALVE

         White SQUIB VLV READY light OFF.

      3) Annunciator SLC System A OOSVC ON

      4) Annunciator SLC SYSTEM B OOSVC ON.

      5) SLC A SYS STATUS SQUIB A LOSCONT/PWRLOSS light ON

      6) SLC B SYS STATUS SQUIB B LOSCONT/PWRLOSS light ON.

   b. **CHECK** OPEN Tank Outlet Valves:

      - 1C41-F001A, STORAGE TANK OUTLET VALVETO SLC PUMP A SUCTION
      - 1C41-F001B, STORAGE TANK OUTLET VALVETO SLC PUMP B SUCTION

   c. SLC Pump A RUNNING

   d. SLC Pump B RUNNING.

   e. RWCU ISOLATES:

      - G33-F004, RWCU PMP SUCT CTMT OTBD ISOL, CLOSED (SLC A Initiated).
      - G33-F001, RWCU PMP SUCT DRWL INBD ISOL, CLOSED. (SLC B) Initiated.
      - G33-F251, RWCU SPLY TO RWCU HXS, CLOSED (SLC B) Initiated.

2. **CHECK** SLC INJECTING INTO the RPV by observing following:

   - SLC Pump discharge pressure greater than reactor pressure.
   - SLC Tank Level lowering.
   - Nuclear instrumentation lowering.

3. **MONITOR** Reactor Power

Examination Outline Cross Reference	Level	RO
212000 Reactor Protection System K3. Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: K3.02 Primary containment isolation system/nuclear steam supply shut-off: Plant-Specific	Tier	2
	Group #	1
	K/A	212000 K3.02
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 35

The plant is operating at rated power.

The incoming feeder breaker to the 16AB bus trips and the Div 2 DG restores 16AB power.

The MSIV/ DR VLV TRIP INIT red alarm is sealed-in.

Which of the following will cause all MSIVs to close?

A. ½ Scram on 'A' side

B. Main Steam Tunnel Temp B fail high

C. Main Steam Tunnel delta temp A fail high

D. C71-S003A, MG SET A OUTPUT BREAKER trip

Answer: D
Explanation: A loss of 16AB will cause a loss of normal power to the RPS M/G set 'B'. This in turn causes a ½ scram on Div 2 (B) side and loss of power to the Div 2 solenoids for the MSIVs. To close the MSIVs power must be removed from the Div 1 solenoids. 'D' is correct. C71-S003A is a EPA breaker but with this breaker tripping will cause a reactor scram and a full MSIV closure.
Distracters: 'A' is wrong – This action will only cause a full scram not a MSIV closure.

‘B’ is wrong – With a ‘B’ temperature in the steam tunnel failing high would cause a Group 1 valve logic to initiate, however this would initiate a Div 2 isolation signal which the Div 2 solenoid is already de-energized from the loss of power.

‘C’ is wrong – Main Steam tunnel delta Temp no longer provides a Group 1 isolation signal.

K/A Match

This question requires the applicant to have the knowledge of the effects on the Containment isolation system on loss of RPS power.

Technical References:

04-1-01-C71-1 Rev 37
04-1-02-1H13-P601-19A-E4, MSIV/ DR VLV TRIP INIT

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C7100 Objective: 11.6

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(9)	
Level of Difficulty:	3.0	

PRA Applicability:

RPS is listed as #5 on the System Importance to CDF.

MSIV/DR VLV TRIP INIT

04-1-02-1H13-P601-19A-E4	
Revision: 162	Page 1 of 1

Safety Related

Alarm Device 1B21-FAH-L710

1.0 POSSIBLE CAUSES

1.1	Reactor Vessel Water Level 1	-150.3"
1.2	Main Condenser Vacuum Low	9"H
1.3	Main Steam Tunnel Temperature High	185°F
1.4	Main Steam Line Pressure Low	<849 psig (RUN MODE ONLY)
1.5	Main Steam Line Flow High	252.5 psid
1.6	Loss of ESF Bus 15AA **OR** 16AB	(De-energizes NSSSS Isolation Logic)

2.0 AUTOMATIC ACTION

2.1 Main Steam Line Isolation (Group 1), **IF** Both NSSS trip systems are tripped.

2.2 **IF** full isolation occurred, **THEN** the following valves Will close:

B21-F022A, B, C, D B21-F067A, B, C, D
B21-F028A, B, C, D B21-F016, B21-F019

3.0 IMMEDIATE OPERATIOR ACTION

3.1 **CHECK** that appropriate automatic actions occur.

3.1.1 **CHECK** MAIN STEAM LINE PILOT SOLENOID A/B indicating lights on 1H13-P622 **AND CHECK** that half of Group 1 isolation is present.

3.1.2 **DETERMINE** cause of alarm **AND RESET** half isolation as soon as possible.

3.2 **IF** reactor scram has occurred, **THEN ENTER** scram ONEP 05-1-02-III-5.

4.0 SUBSEQUENT OPERATOR ACTION

4.1 **WHEN** plant status has returned to normal, **RESET** isolation signal by **DEPRESSING** NSSS OTBD **AND** NSSS INBD ISOLATION RESET pushbuttons on 1H13-P601.

System Reactor Protection System MPL No. C71

Checksheet Name Electrical Lineup Checksheet

Instruction Step 4.1.1b

COMPONENT NO.	COMPONENT DESCRIPTION	MCC, LCC PANEL NO.	BREAKER NO.	REQUIRED POSITION	BRKR DEV	INITIALS	
						1st Ck	2nd Ck
	RPS MG A BACKUP XFMR FEED	15B42	52-154204	CLOSED			
S001A	MOTOR GENERATOR SET	13B22	52-132215	CLOSED			
	RPS MG B BACKUP XFMR FEED	16B42	52-164227	CLOSED			
S001B	MOTOR GENERATOR SET	14B22	52-142229	CLOSED			
1H13-P691	PGCC PANEL 1H13-P691 C71	1DA1	72-11A30	CLOSED			
	RPS CHAN A SCRAM SOL	1C71-P001	(CB2A) 52-1C71102	CLOSED			
	RPS CHAN C SCRAM SOL	1C71-P001	(CB8A) 52-1C71108	CLOSED			
	RPS A ALTERNATE FEED	1C71-P001	(CB1A) 52-1C71101	CLOSED			
	RPS CHAN B SCRAM SOL	1C71-P002	(CB2B) 52-1C71202	CLOSED			
	RPS CHAN D SCRAM SOL	1C71-P002	(CB8B) 52-1C71208	CLOSED			
	RPS B ALTERNATE FEED	1C71-P002	(CB1B) 52-1C71201	CLOSED			
	SCRAM DISCH VOLUME POSITION INDICATOR LIGHTS FOR REACTOR PROTECTION SYSTEM	12P11	52-1P21127	CLOSED			
1H13-P692	PGCC PANEL 1H13-P692 C71	1DB1	72-11B30	CLOSED			
*	MG SET A OUTPUT BREAKER	(LOCAL)	C71-S003A	CLOSED			
*	MG SET A OUTPUT BREAKER	(LOCAL)	C71-S003C	CLOSED			
*	RPS A ALTERNATE FEED	(LOCAL)	C71-S003E	CLOSED			
*	RPS A ALTERNATE FEED	(LOCAL)	C71-S003G	CLOSED			

*Breakers close per Sections 4.1 and 4.3 of this instruction, if breakers are open.

Examination Outline Cross Reference	Level	RO
212000 Reactor Protection System A4. Ability to manually operate and/or monitor in the control room: A4.08 Individual system relay status: Plant-Specific	Tier	2
	Group #	1
	K/A	212000 A4.08
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 36

HANDOUT PROVIDED

The plant is currently in a refueling outage and operators are pulling fuses on a clearance for a Reactor Protection System (RPS) logic modification.

- Currently fuse 1C71F27B is removed.
- The Non-licensed operator has just removed fuse 1C71F14C.

Using E1173-019 determine the current status of the RPS System.

A. No half scram signal

B. Division I half scram signal

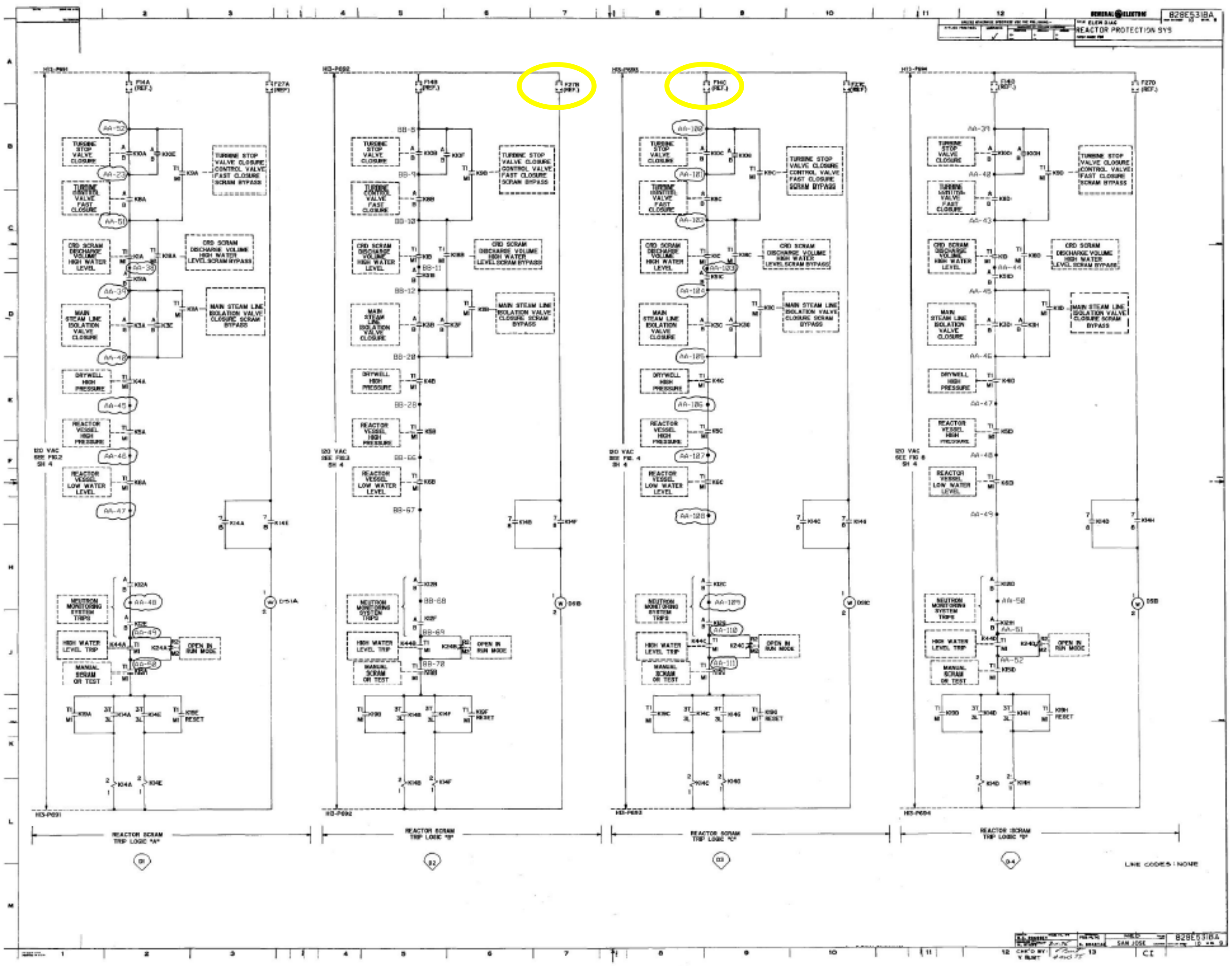
C. Division II half scram signal

D. Full scram signal

Answer: B
Explanation: Ref E1173-019; pulling fuse F27B only de-energizes white indicating light, not the K14B or K14F relays. Pulling fuse F14C, de-energizes K14C and K14G which gives a half scram in channel 'C' which is a Division I channel. 'B' is correct.
Distracters: 'A' is wrong – Pulling the K14C will cause a half scram 'C' is wrong – only affects the Division 1 side not Division 2

'D' is wrong – pulling fuse F27B only de-energizes white indicating light, not the K14B or K14F relays. Pulling fuse F14C, de-energizes K14C and K14G which gives a half scram in channel 'C' which is a Division I channel.		
K/A Match This question requires the applicant to have the ability to determine individual system relay status with given conditions with the RPS system.		
Technical References: E1173-019		
Handouts to be provided to the Applicants during exam: E1173-019		
Learning Objective: GLP-OPS-C7100 21, 5.3, 6.3		
Question Source:	**Bank #**	Site bank not on previous NRC
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3.0	
PRA Applicability:		
RPS is listed as #5 on the System Importance to CDF.		

E-1173-019	MPL No. 1C71019E



ASSOCIATED GE DRAWING NO. 828E531BA | REV. NO. 26

NOTE

1.

2. FOR GE PREFIX, SEE GE NOTE 10, E-1173-11
ALL DEVICES, COMPUTER ADDRESSES AND ANNUNCIATORS ARE PREFIXED BY 1C71 UNLESS OTHERWISE NOTED.

UPPER-TIER DRAWING
OPERATIONS CRITICAL

NO.	REVISIONS						
007	AS-BUILT PER EC 47929, AND EC 47927	WE	N/A	N/A	LJ	FH	3-25-16
006	AS-BUILT PER EC 47928, AND EC 47930	WE	N/A	N/A	DMJ	LJ	3-16-16
005	AS-BUILT PER DCP 88/0280-01 SWD: REMOVED GE REV. BLOCK AND DELETED NOTE 1.	[illegible]		ON FILE			
004	AS-BUILT PER DRN * 218	JM	TAG	ALL	EEC	JWK FOR SPH	[illegible] 87

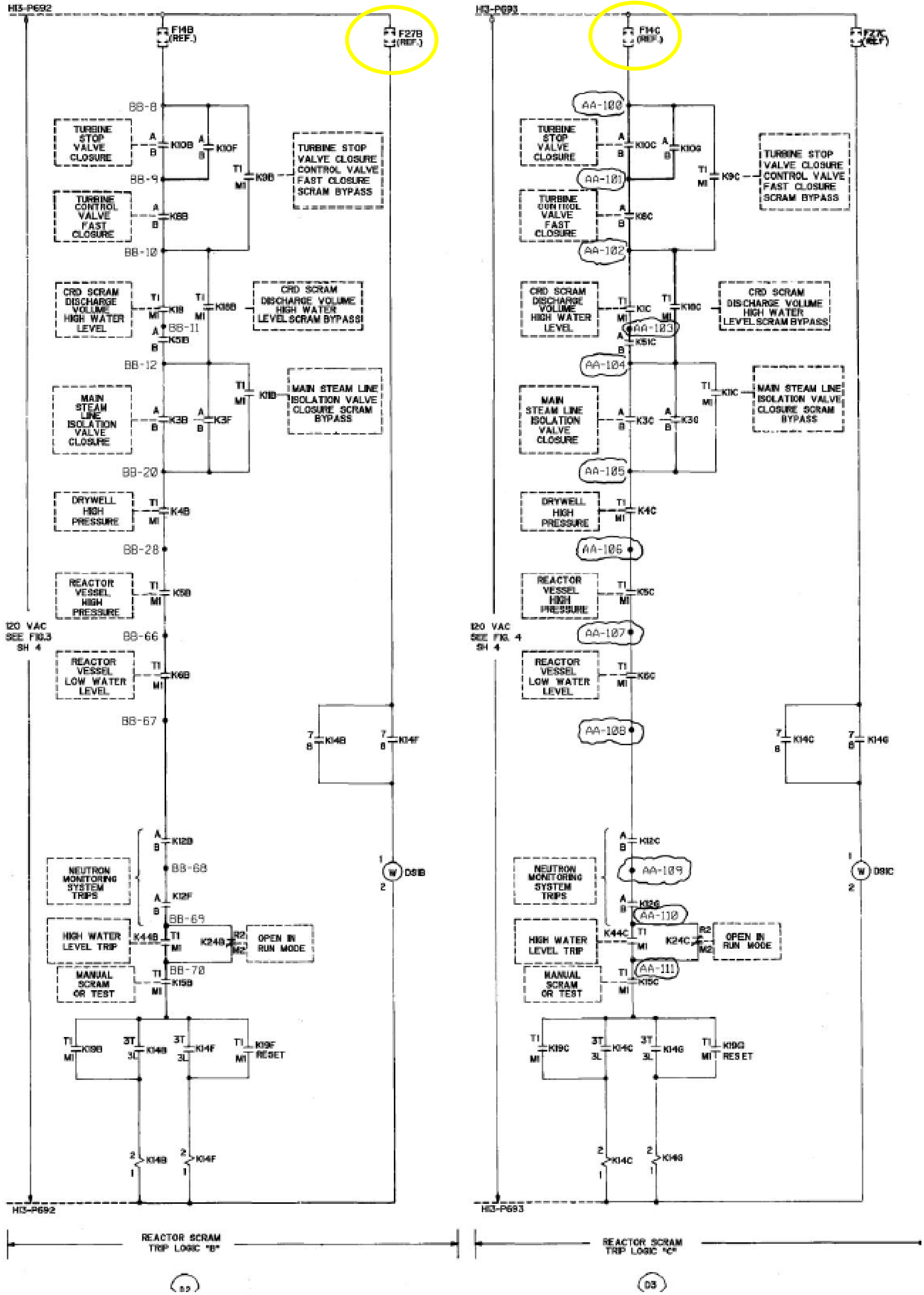
Q

GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING

SCHEMATIC DIAGRAM
C71 REACTOR PROTECTION SYSTEM
CHANNEL A, B,C, D, SCRAM TRIP LOGIC

MPL No. 1C71019E	SCALE: NONE	DRAWING NO. E-1173-019	REV. 007

DFN: E1173019.DGN


H13-P692
F14B (REF.)
F27B (REF.)
BB-8
TURBINE STOP VALVE CLOSURE
K10B
K10F
K9B
TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE SCRAM BYPASS
BB-9
TURBINE CONTROL VALVE FAST CLOSURE
K8B
BB-10
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
K1B
K18B
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL SCRAM BYPASS
BB-11
K51B
BB-12
MAIN STEAM LINE ISOLATION VALVE CLOSURE
K3B
K3F
K1B
MAIN STEAM LINE ISOLATION VALVE CLOSURE SCRAM BYPASS
BB-20
DRYWELL HIGH PRESSURE
K4B
BB-28
REACTOR VESSEL HIGH PRESSURE
K5B
120 VAC SEE FIG.3 SH 4
BB-66
REACTOR VESSEL LOW WATER LEVEL
K6B
BB-67
K14B
K14F
K12B
NEUTRON MONITORING SYSTEM TRIPS
BB-68
K12F
BB-69
DS1B
HIGH WATER LEVEL TRIP
K44B
K24B
OPEN IN RUN MODE
MANUAL SCRAM OR TEST
BB-70
K15B
K19B
K14B
K14F
K19F RESET
K14B
K14F
H13-P692
REACTOR SCRAM TRIP LOGIC "B"
H13-P693
F14C (REF.)
F27C (REF.)
AA-100
TURBINE STOP VALVE CLOSURE
K10C
K10G
K9C
TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE SCRAM BYPASS
AA-101
TURBINE CONTROL VALVE FAST CLOSURE
K8C
AA-102
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
K1C
K18C
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL SCRAM BYPASS
AA-103
K51C
AA-104
MAIN STEAM LINE ISOLATION VALVE CLOSURE
K3C
K3G
K1C
MAIN STEAM LINE ISOLATION VALVE CLOSURE SCRAM BYPASS
AA-105
DRYWELL HIGH PRESSURE
K4C
AA-106
REACTOR VESSEL HIGH PRESSURE
K5C
120 VAC SEE FIG. 4 SH 4
AA-107
REACTOR VESSEL LOW WATER LEVEL
K6C
AA-108
K14C
K14G
K12C
NEUTRON MONITORING SYSTEM TRIPS
AA-109
K12G
AA-110
DS1C
HIGH WATER LEVEL TRIP
K44C
K24C
OPEN IN RUN MODE
MANUAL SCRAM OR TEST
AA-111
K15C
K19C
K14C
K14G
K19G RESET
K14C
K14G
H13-P693
REACTOR SCRAM TRIP LOGIC "C"

Examination Outline Cross Reference	Level	RO
215003 Intermediate Range Monitor (IRM) System K2. Knowledge of electrical power supplies to the following: K2.01 IRM channels/detectors	Tier	2
	Group #	1
	K/A	215003 K2.01
	Rating	2.5
	Revision	0
Revision Statement:		

Question: 37

Which of the following is the power supply to the IRM detectors?

A. RPS

B. BOP Inverters

C. ESF Inverters

D. ESF MCC AC

Answer: C

Explanation:

IRMs are powered from ESF Inverters 1Y87, 1Y88, 1Y95, and 1Y96

'C' is correct.

Distracters:

'A' is wrong – RPS is a 125Vac system but not the power supply for the IRMs

'B' is wrong – BOP inverters deliver the same power but to non ESF systems

'D' is wrong – ESF power supply is correct but a MCC is 480Vac not 120Vac

K/A Match

This question requires the applicant to have the knowledge of the IRM power supplies.

Technical References:

GLP-OPS-C5102, Rev 6, IRM Lesson Plan
04-1-01-C51-1, Neutron Monitoring SOI, Rev 32

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-C5102, Objective 6		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	
PRA Applicability:		
 None.		

Lesson Content	Instructor Notes
▪ In addition to the above functions, the voltage preamplifier also provides proper impedance matching between the amplifier and the signal cable and provides a path for power from the high voltage power supply to the detector.	
▪ There are no controls, indications, or interlocks directly associated with the voltage preamplifiers with the exception of the range switching circuitry of the amplifier/attenuator unit in the IRM drawer.	Full details of range switching to be discussed shortly.
− This range switch circuitry controls the actuation of the two relays in the voltage preamplifier that determine which of the two frequency sensitive amplifiers in the voltage preamplifier the detector output signal will pass through.	
2. IRM Drawer	Figure 11
▪ The IRM drawers house the electronic circuits and devices that permit the IRM System to function, and the controls and indications necessary to functionally check the electronic circuitry.	Objective 3.4
▪ The IRM drawers contain the following major components, each of which will be discussed separately: − Amplifier Attenuator − Inverter Module − Mean Square Analog Unit − Operational Amplifier − Trip Units	
▪ The IRM drawers, with one for each IRM channel, are located in the associated divisional Neutron and Radiation Monitoring Cabinets. − Division 1 cabinets (1H13 P669 and P671) are located in the Upper Control Room. − Division 2 cabinets (1H13 P670 and P672) are located in the Main Control Room.	
▪ The IRM drawers and associated power supplies and circuitry are powered from the following 120V AC ESF Uninterruptible Power Supplies (UPS):	Objective 6.2

Lesson Content	Instructor Notes

Channel	Cabinet	Inverter
A/E	1H13 P669 (Div 1)	1Y87
B/F	1H13 P670 (Div 2)	1Y88
D/H	1H13 P672 (Div 2)	1Y95
C/G	1H13 P671 (Div 1)	1Y96

Remember:
A
B
B
A

- This arrangement satisfies the electrical and physical separation criteria required for the Reactor Protection System.

3. High Voltage Power Supply

Figure 9

- The high voltage power supply provides the polarizing voltage necessary for the operation of the fission chamber detector.

Objective 3.5

- The high voltage power supplies are driven by 20 volt DC power supplies (total of four, one located in each of the panels 1H13-P669, P670, P671 and P672) which are in turn powered from the Uninterruptible Power Supplies discussed above.

Objective 6.3

- This DC voltage from the Uninterruptible Power Supplies is applied through the voltage preamplifiers to the IRM detectors and is normally adjusted to ~+100V DC.
  - Adjustable from +100V DC to +350V DC.
  - Increasing the applied voltage to the detector will increase its sensitivity.

4. Amplifier Attenuator

Figures 9 and 10

Objective 3.6

- The purpose of the amplifier/attenuator unit is to attenuate the output signal from the IRM detector voltage preamplifier according to the particular range selected on the associated IRM Range Switch.
  - Accomplished by switching different amounts of resistance into the signal flowpath using a bank of relays and resistors.
  - Relays are operated by control signals derived from the range switch circuitry.

Examination Outline Cross Reference	Level	RO
215004 Source Range Monitor (SRM) System K4. Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.04 Changing detector position	Tier	2
	Group #	1
	K/A	215004 K4.04
	Rating	2.8
	Revision	0
Revision Statement:		

Question: 38

A reactor startup is in progress with the reactor having just gone critical.

- Operators have verified SRM/IRM overlap
- Operators are continuing to withdraw control rods
- SRMs are being withdrawn as necessary to maintain SRM count rate between 1×10^2 and 1×10^5 cps.

The SRM ‘E’ detector fails and indicates 0 cps output.

Which of the following describes the result?

A. Div 1 half scram

B. Control Rod Insert Block

C. Control Rod Withdrawal Block

D. Detector motor auto stops until reset

Answer: C
Explanation: With Overlap confirmed then you must assume that IRMs are between range 1 and 3. If the SRM detector is not full in and indication is <100 cps a rod block is generated. ‘C’ is correct.
Distracters:

'A' is wrong – Even though this is considered an INOP, SRMs will not provide scram signals.

'B' is wrong – SRMs will not provide an INSERT block, only Withdraw blocks

'D' is wrong – there is no auto functions on the detector drive motors.

K/A Match

This question requires the applicant to have the knowledge of the SRM detector wrong position rod block.

Technical References:

GLP-OPS-C5101, Rev 11, SRM Lesson Plan
04-1-01-C51-1, Neutron Monitoring SOI, Rev 32

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5101, Objective 8.2

Question Source:	**Bank # 652**	2013 Audit Exam
(note changes and attach parent)	**Modified Bank #**	
	New	

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X

10CFR Part 55 Content:	55.41(b)(7)	

Level of Difficulty:	3.0	

PRA Applicability:

None.

Lesson Content	Instructor Notes
▪ **SRM Low Flux and Detector Position**: A rod block signal is generated if the SRM count rate is < 106 cps and the detector is not fully inserted. – This trip indicates that the SRM detector has been excessively withdrawn.	Objective 8.2 SOI P&L: A rod block will be generated for SRM channels A and E or B and F if a channel is INOP or upscale even if bypassed and the opposite channel is not fully inserted and reads less than 100 cps.
▪ The SRM Low Flux and Detector Position rod block is bypassed under any of the following conditions: – IRM Range Switches are > range 3 – Reactor Mode Switch is in RUN – SRM joystick is in BYPASS – Detector fully inserted	
b. Scram Signals	Figures 13 and 14
▪ If the RPS "shorting links" are removed (such as during refueling operations), a trip condition in a single channel will send a signal to RPS to initiate a **full** scram.	"shorting links" are unique to scram circuit
▪ **SRM Upscale Trip:** SRM upscale trip occurs at 2×10^5 cps.	Objective 8.3
▪ SRM Upscale Trip is bypassed under any of the following conditions: – "Shorting Links" are installed – SRM joystick is in BYPASS	
▪ **SRM Inoperative:** SRM inoperative signal generates a scram signal in RPS under circumstances when the output of the SRM channel may be unreliable.	Objective 8.3

Examination Outline Cross Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System K2. Knowledge of electrical power supplies to the following: K2 02 APRM channels	Tier	2
	Group #	1
	K/A	215005 K2.02
	Rating	2.6
	Revision	0
Revision Statement:		

Question: 39

Which of the following is the power supply to APRM Channel 1 cabinet?

What is the result if this power is lost?

A. 1Y99
Division 1 half scram

B. 1Y87
Division 1 half scram

C. 1Y87
Control Rod Block only

D. 1Y99
Control Rod Block only

Answer: B
Explanation: The UPS system supplies regulated 120VAC power to the particular APRM channel and Voter. Each channel and voter has individual UPS power supply. 1Y87 feeds Channel 1. A loss of 120VAC UPS will result in a half scram condition for the particular RPS due to the Voter de-energizing. 'B' is correct.
Distracters: 'A' is wrong – 1Y99 has the same features as an ESF inverter and provides power to important control features however it is considered a BOP inverter and does not supply power to the APRMs 'C' is wrong – A loss of cabinet power will de-energize the voter and cause a half scram

‘D’ is wrong – 1Y99 has the same features as an ESF inverter and provides power to important control features however it is considered a BOP inverter and does not supply power to the APRMs and A loss of cabinet power will de-energize the voter and cause a half scram.

K/A Match

This question requires the applicant to have the knowledge of the APRM power supplies and the resulting actions if lost.

Technical References:

GLP-OPS-C5104, Rev 11, APRM Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5101, Objective 7.2, 10, 11.2, 14

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Lesson Content	Instructor Notes
- Each of 4 QLVPS is assigned to an associated APRM/LPRM chassis. 120 VAC to each QVLPS chassis is converted to +5 to +15 VDC for chassis functions.	*"Quad"*
- The Internal power supply arrangement is such that a failure of one of the power supplies allows the PRNMS chassis to continue to be powered.	
- Each QLVPS chassis contains 4 separate power supplies.	
○ APRM channels 1 and 3 are powered from Division 1 UPS power	
- [10] APRM Channel 1 – 1Y87 (Division 1 UPS)	*Objective 10*
- APRM Channel 3 – 1Y96 (Division 1 UPS)	
○ APRM channels 2 and 4 are powered from Division 2 UPS power.	
- [10] APRM Channel 2 – 1Y88 (Division 2 UPS)	*Objective 10*
- APRM Channel 4 – 1Y95 (Division 2 UPS)	
• APRM Channel Indication	
○ [9] APRM chassis panels provide indications of average core thermal power, recirc loop flow inputs to the APRM circuitry, and controls which enable APRM function selection and testing.	*Objective 9* *Figure 3*
○ The APRM channels are housed within the following cabinets, as follows:	

[9] APRM Channel	Cabinet	Location	ESF Inverter
1	H13-P669	UCR	1Y87
2	H13-P670	MCR	1Y88
3	H13-P671	UCR	1Y96
4	H13-P672	MCR	1Y95

Objective 9

UCR – Upper Control Room

MCR – Main Control Room

• APRM Scram Signals

○ The APRM upscale trip has a fixed setpoint (Neutron Flux), not variable with recirc flow

Lesson Content	Instructor Notes
○ 7.2 When the mode switch is not in RUN, the UPSCALE NEUTRON flux trip circuitry changes the fixed scram setpoint to 18%.	*Objective 7.2*
- Indication of an UPSCALE NEUTRON flux trip is provided by indication on the associated APRM chassis and the red UPSC TRIP OR INOP light on P680.	
○ 7.2 The UPSCALE THERMAL power trip is the previously discussed flow biased scram signal.	*Objective 7.2*
- This signal is clamped at a maximum of 111%, regardless of flow.	
- The signal from the flux amplifier to the upscale thermal trip unit is delayed by 6 seconds to simulate a fuel thermal time constant.	
- Indication of an UPSCALE THERMAL power trip is provided by indication on the associated APRM chassis and the red UPSC TRIP OR INOP light on P680.	
○ 7.2 The INOP portion of the APRM UPSC TRIP OR INOP alarm on Control Room Panel P680 is generated by any of the following:	*Objective 7.2* *Figure 14*
- APRM channel mode switch NOT in OPERATE	
- Critical self-test fault detected	
- Firmware / Software watchdog timer timed out	
- Loss of input power	
○ The APRM Chassis will display INOP.	
- 7.1, 7.2 Each of these conditions will generate a rod block and a scram signal.	*Objectives 7.1, 7.2*
○ In addition, the INOP portion of the APRM UPSC TRIP OR INOP alarm Panel P680 is generated by:	
- 7.1 LPRM Count and Level <21 LPRM Inputs	*Objective 7.1* *Annunciator P680-5A-C10*
○ The APRM Chassis will display INOP.	
- 7.1 This condition will generate a control rod withdrawal block and APRM DNSC / TRBL annunciator.	*Objective 7.1*

Lesson Content	Instructor Notes
○ 3.8, 5 The Control Room Panel P680 IRM / APRM recorders are programmable, with a full digital display screen.	*Objectives 3.8, 5*
○ Indication on the recorders corresponds to the individual IRM or APRM channel feeding the recorder.	
- Eight 3-position APRM / OFF / IRM control switches located on P680 control whether the recorder pen is displaying the output of the associated APRM or the associated IRM.	*4 – APRM/OFF/IRM* *4 – OFF/OFF/IRM*
- During a plant startup, the switches are initially placed in the IRM position and then placed in the APRM position after the Reactor Mode Switch is transferred to RUN.	
B. SYSTEM INTERRELATIONSHIPS	
• LPRM System	
○ 11.1 Forty-four LPRMs supply the local core flux inputs to the APRM channel averaging circuit amplifiers and count circuits.	*Objective 11.1*
- < 21 LPRMs total or < 3 LPRMs per level input to an APRM channel will cause an INOP alarm in that channel.	
- The APRM INOP alarm will then cause a rod block only.	
- The associated Reactor Protection System (RPS) channel is not affected.	
• 120V AC Uninterruptible Power Supply (UPS) System	
○ 11.2 The UPS System supplies regulated 120V AC power to the particular APRM channel and Voter.	*Objective 11.2*
○ Loss of 120 VAC UPS will result in a half-scram condition for the particular RPS due to the Voter de-energizing.	
• Reactor Recirc System	

Examination Outline Cross Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System A4. Ability to manually operate and/or monitor in the control room: A4.04 LPRM back panel switches, meters and indicating lights	Tier	2
	Group #	1
	K/A	215005 A4.04
	Rating	3.2
	Revision	3
Revision Statement: **Rev 1 – Added "to initiate a control rod block" to the stem for clarification** **Rev 2 – Added (not bypassed) to stem for clarification**		

Question: 40

APRM Channel 2 has the following LPRM indications on the backpanels:

- 8 LPRMs are bypassed
- 2 additional LPRMs are indicating 0 (not bypassed)

(1) How many more LPRMs must be bypassed to initiate a control rod withdrawal block?

(2) How many LPRMs are required per level for APRM operability?

A. (1) 14
(2) 2

B. (1) 14
(2) 3

C. (1) 16
(2) 2

D. (1) 16
(2) 3

Answer:	**D**
Explanation: The APRM count and level feature monitors and indicates the number of valid LPRMs assigned to an APRM channel. Each LPRM which is not in BYPASS is counted by the APRM. LPRM outputs which are failed low or high are considered valid inputs until they are bypassed at the respective APRM cabinet. If there are less than 21 valid LPRM inputs or less than 3 LPRMs per detector level to the APRM channel, the count circuit will initiate a control rod withdrawal block and an APRM Trouble Annunciator.	

‘D’ is correct.

Distracters:

‘A’ is wrong – The student must know if the LPRM is reading 0 then it is still counted in the counting circuit. Two LPRMs are indicating 0 but are still counted. Less than 3 LPRMs per detector level to the APRM channel will cause an inop.

‘B’ is wrong – The student must know if the LPRM is reading 0 then it is still counted in the counting circuit. Two LPRMs are indicating 0 but are still counted

‘C’ is wrong – Less than 3 LPRMs per detector level to the APRM channel will cause an inop.

K/A Match

This question requires the applicant to have the ability to monitor LPRM indications in the control room to determine status.

Technical References:

GLP-OPS-C5104, Rev 11, APRM Lesson Plan
04-1-01-C51-4, Neutron Monitoring SOI, Rev 32

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5104, Objective 7.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Lesson Content	Instructor Notes
○ A full scram will **NOT** occur if 1 APRM channel is UPSC/INOP, 1 OPRM channel is DIDA/INOP and 1 OPRM channel is CDA/INOP.	
○ This would only be 1 vote for each function and would NOT result in the 2-out-of-4 Logic Modules tripping RPS.	
○ Bypassing an APRM channel also bypasses the respective OPRM channel and removes any votes, if any are present, to the 2-out-of-4 Logic Modules.	
○ The APRM chassis, while in the OPERATE mode, performs a constant **self-test**.	
○ The system determines if any detected faults are critical or non-critical. A critical fault is one that will compromise the channel's ability to perform its safety function.	
○ **APRM Trip, Control Rod Block and Alarm** signals generated by the APRM chassis are transmitted to and processed by the 2-out-of-4 Logic Module.	
• LPRM Count and Level Feature	
○ The APRM count and level feature monitors and indicates the number of valid LPRMs assigned to an APRM channel.	*Figure 2*
○ Each LPRM which is not in BYPASS is counted by the APRM.	*LPRM outputs which are failed low or high are considered valid inputs until they are bypassed at the respective APRM cabinet*
○ 7.1 If there are less than 21 valid LPRM inputs or less than 3 LPRMs per detector level to the APRM channel, the count circuit will initiate a control rod withdrawal block and an APRM Trouble Annunciator.	*Objective 7.1*
○ The LPRM counting and level feature ensures that there are at least 21 LPRM inputs or at least 3 LPRMs per detector level to an APRM to provide an accurate representation of the average radial core thermal power by initiating a control rod block if too few LPRM inputs are sensed.	*Additional details of rod blocks, including bypasses will be discussed later*
• Recirc Loop Flow Input	

Lesson Content	Instructor Notes
○ 7.2 When the mode switch is not in RUN, the UPSCALE NEUTRON flux trip circuitry changes the fixed scram setpoint to 18%.	*Objective 7.2*
- Indication of an UPSCALE NEUTRON flux trip is provided by indication on the associated APRM chassis and the red UPSC TRIP OR INOP light on P680.	
○ 7.2 The UPSCALE THERMAL power trip is the previously discussed flow biased scram signal.	*Objective 7.2*
- This signal is clamped at a maximum of 111%, regardless of flow.	
- The signal from the flux amplifier to the upscale thermal trip unit is delayed by 6 seconds to simulate a fuel thermal time constant.	
- Indication of an UPSCALE THERMAL power trip is provided by indication on the associated APRM chassis and the red UPSC TRIP OR INOP light on P680.	
○ 7.2 The INOP portion of the APRM UPSC TRIP OR INOP alarm on Control Room Panel P680 is generated by any of the following:	*Objective 7.2* *Figure 14*
- APRM channel mode switch NOT in OPERATE	
- Critical self-test fault detected	
- Firmware / Software watchdog timer timed out	
- Loss of input power	
○ The APRM Chassis will display INOP.	
- 7.1, 7.2 Each of these conditions will generate a rod block and a scram signal.	*Objectives 7.1, 7.2*
○ In addition, the INOP portion of the APRM UPSC TRIP OR INOP alarm Panel P680 is generated by:	
- 7.1 LPRM Count and Level <21 LPRM Inputs	*Objective 7.1* *Annunciator P680-5A-C10*
○ The APRM Chassis will display INOP.	
- 7.1 This condition will generate a control rod withdrawal block and APRM DNSC / TRBL annunciator.	*Objective 7.1*

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Do **NOT** operate the SRM/IRM detectors in a high neutron flux field for extended periods of time in order to prolong detector life.

3.2 During core alterations a minimum of two SRM channels Must be Operable: One in the core quadrant being altered **AND** one in an adjacent quadrant. The 90° quadrants cannot be rotated.

3.3 APRM/OPRM Subsystem

NOTE

The OPRM channels are **NOT** required to be operable during the initial part of Cycle 19 per GGNS OL Condition 2.C.(2). GGNS Will conduct monitoring of the OPRMs. The OPRM Upscale function (Function 2.f of Technical Specification Table 3.3.1.1-1) Will be disabled **AND** operated in an "indicate only" mode until monitoring is completed.

3.3.1 The APRM subsystem is divided into four APRM/OPRM channels **AND** four 2-Out-Of-4 Voter channels. Each APRM/OPRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. A trip from any one un-bypassed APRM/OPRM channel Will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. (i. e. no half-scram)

3.3.2 A trip from any two un-bypassed APRM/OPRM channels Will result in a full trip in each of the four 2-Out-Of-4 Voter channels, which in turn results in two trip inputs to each RPS trip system logic channel **AND** a full scram. Three of the four APRM/OPRM channels **AND** all four of the voter channels are required to be OPERABLE to ensure that no single failure Will preclude a scram on a valid signal. Two APRM trips **OR** two OPRM trips (of the same type) are required. One APRM **AND** one OPRM trip Will **NOT** result in a full scram.

3.3.3 Each APRM **AND** LPRM chassis is assigned 22 LPRM inputs, making a total of 44 assigned to the associated APRM channel. To provide adequate coverage of the entire core, at least 20 LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels, Must be operable for each APRM/OPRM channel .

3.3.4 For the OPRM Upscale function, LPRMs are assigned to "cells" of four detectors. A minimum of 30 cells, each with a minimum of 2 LPRMs, Must be OPERABLE for the OPRM Upscale function to be OPERABLE. **IF** more than 1 OPRM channel is inoperable, **THEN** Backup Stability Protections must be implemented per TS 3.3.1.1.

3.3.5 Each APRM receives input from two recirc loop transmitters (Loops A **AND** Loop B). The APRM Will generate a FLOW UPSCALE alarm **AND** rod block if setpoints are exceeded. It also generates a FLOW COMPARE alarm **IF** any two APRM channels' total recirc flow values (A plus B) loop differ by more than a pre-set value.

APRM
DNSC/TRBL

Safety Related

Alarm Device 1C51-XA-L602

1.0 POSSIBLE CAUSES

1.1 One **OR** more of the four APRM channels indicates $\leq$ 5% power.

1.2 One **OR** more APRM channels with any of the following conditions:

a. **ANY** self-test fault detected.

b. Less than 3 non-bypassed LPRM detectors at any given level in an APRM channel.

NOTE

Minimum LPRMs for an APRM channel required by TS 3.3.1.1, BASES is 20, **HOWEVER** setpoint is 21 for conservatism.

c. Less than 21 non-bypassed LPRM detectors for an APRM channel.

2.0 AUTOMATIC ACTION

2.1 Control rod withdrawal block from APRM Downscale (1.1) **IF** mode switch in RUN.

2.2 Control rod withdrawal block from APRM Trouble **IF** due to less than minimum LPRMs operable. (1.2.b or 1.2.c)

3.0 IMMEDIATE OPERATOR ACTION

NOTE

APRMs are expected to be downscale until reactor power reaches at least 5% of rated.

CAUTION

Do NOT DEPRESS the CLEAR LOG status key, even IF fault clears.

3.1 **DETERMINE** the cause of the alarm as follows:

3.1.1 **OBSERVE** APRM recorders **AND** APRM status light indications on 1H13-P680 to **DETERMINE** affected APRM channel for APRM Downscale.

Examination Outline Cross Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) 2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	Tier	2
	Group #	1
	K/A	217000 2.4.34
	Rating	4.2
	Revision	1
Revision Statement: **Rev 1 – Validation, validators recognized that 'A' was also correct, so added "plant operator has been dispatched to the RCIC room" for clarification**		

Question: 41

Events have occurred and the Control Room has been evacuated.

RCIC is initiated and controls are taken from the Remote Shutdown Panel.

Plant operator has been dispatched to the RCIC room.

Which of the following would cause RCIC to be unavailable for operation?

A. Overspeed trip

B. RPV level 8 is reached

C. RCIC Isolation Group 4 signal

D. Auto suction swap to Suppression Pool

Answer: **C**
Explanation: If RCIC is operating from the remote Shutdown Panel and a Group 4 isolation is received then RCIC will no longer be available for level control due to the Group 4 can't be reset from the remote shutdown panel. 'C' is correct.
Distracters: 'A' is wrong – The overspeed trip can be reset locally and RCIC restarted. 'B' is wrong – The F045 will auto close but RCIC can be restarted when level 8 auto resets. 'D' is wrong – A suction swap has not effect on the operation of the RCIC system.

K/A Match This question requires the applicant to have the Knowledge of RO tasks for RCIC performed outside the main control room during an emergency and the resultant operational effects.
Technical References: GLP-OPS-E5100, Rev 11, RCIC Lesson Plan 04-1-01-E51-1, RCIC SOI, Rev 142 05-1-02-II-1, Shutdown From the Remote Shutdown Panel ONEP, Rev 51
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-E5100, Objective 11.5

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:
RCIC is listed as #17 on the System Importance to CDF.

2.0 ATTACHMENTS

2.1 Attachment I - Manual Valve Lineup Checksheet

2.2 Attachment II - Remote Operated Valve Lineup Checksheet

2.3 Attachment III - Electrical Lineup Checksheet

2.4 Attachment IV - System Alarm Index

2.5 Attachment V - Handswitch Alignment Checksheet

2.6 Attachment VI - Emergency RCIC Manual Start/Shutdown

2.7 Attachment VII - RCIC Turbine Trip Analysis

3.0 PRECAUTIONS AND LIMITATIONS

3.1 **MAINTAIN** temperature leaving lube oil cooler between 60°F and 155°F.

3.2 Do **NOT** operate the RCIC turbine at less than 2000 rpm. The shaft-driven oil pump Will **NOT** be able to deliver 3-5 psig oil pressure to the governor actuator and bearings. Also, chattering of the Turbine Exhaust Check Valve E51-F040 Will occur.

3.3 Except in an emergency, do **NOT** allow suction temperature to exceed 140°F to prevent overheating of the bearings and turbine seals.

3.4 Except during an emergency, RCIC Should **NOT** be operated with suppression pool level less than 14'-6" to ensure proper RCIC PUMP NPSH.

3.5 **IF** a RCIC Steam Isolation occurs, the handswitches for F064 **AND** F063 Must be in the STOP **OR** CLOSE position before the isolation signal Can be reset.

3.6 In the event of a RCIC Turbine overspeed, the Turbine Trip **AND** Throttle valve Must be reset locally. All other trips Can be reset remotely from the Control Room. Attachment VII may be used as an aid.

3.7 **AVOID** pumping suppression pool water to the Reactor, **IF** possible, by maintaining the Suppression Pool below 18'9 3/4" **AND** the CST level > 5' to prevent automatic transfer of the RCIC suction.

3.8 **AVOID** unnecessary **OR** prolonged use of RCIC to prevent heating of the suppression pool.

3.9 Lubrication **AND** cooling for the gland seal compressor is provided by the air flow through the compressor. Do **NOT** allow the compressor to operate at discharge temperatures of 370°F **OR** more.

SHUTDOWN FROM THE REMOTE SHUTDOWN PANEL

RCIC OPERATION (Cont.)

Body of procedure step 3.2.11a (Cont.)

> NOTE
>
> Attachment I May be used to monitor RSD Panel level indication with respect to RCIC Turbine shutdown at Level 8. Attachment II May be used to monitor RSD Panel level indication with respect to actual Reactor Vessel level; specifically, a rising indicated level verifies actual Reactor level greater than -167 inches (lower instrument tap) **AND** less than +66 inches (upper instrument tap).

(9) **CONTROL** Reactor level by controlling RCIC speed with RCIC TURB FLO CONT (FK-R100).

(10) **IF** at any time RCIC TURBINE trips, **THEN RESET** trip as follows, **IF** needed:

> NOTE
>
> **IF** RCIC isolates, **THEN** control of RCIC from Remote Shutdown panel is lost because valve E51-F063 **AND** E51-F064 close **AND CANNOT** be reopened from RSP. In addition, isolation logic **CANNOT** be reset from RSP. No indication of a RCIC isolation is available at RSP.

(a) **CLOSE** RCIC TURB TRIP/THROT VLV 1E51-C002 (HS-M112).

(b) **SHIFT** the RCIC TURB FLO CONT to MANUAL **AND REDUCE** controller to minimum (FK-R100).

(c) **REOPEN** RCIC TURB TRIP/THROT VLV 1E51-C002 (HS-M112). **CHECK** RCIC TURBINE status lights to ensure RCIC Turbine is reset.

(d) **IF** RCIC Turbine did **NOT** reset, **THEN RESET** RCIC overspeed trip locally **AND REPEAT** Steps 10 through10(c) above.

(11) **IF** RCIC steam line isolation is required **AND** it is determined that normal RCIC steam line isolation valves (E51-F063, E51-F064, **AND** E51-F076) are **NOT** closed, **THEN PERFORM** Step 3.1.12b **OR**. 3.2.11b.

Examination Outline Cross Reference	Level	RO
218000 Automatic Depressurization System A1. Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: A1.01 ADS valve tail pipe temperatures	Tier	2
	Group #	1
	K/A	218000 A1.01
	Rating	3.4
	Revision	1
Revision Statement: **Rev 1 – Validation, made a 1 / 2 question**		

Question: 42

A loss of all RPV level indication has occurred during a LOCA.

The CRS has entered EP-5 and directed injection into the RPV.

The CRS has directed you to monitor for RPV flooding.

(1) Which of the following describes the tail pipe temperature indication for RPV flooding?

And

(2) Which valves should be monitored?

A. (1) Rising
(2) ADS valves only

B. (1) Lowering
(2) ADS valves only

C. (1) Rising
(2) LO-LO Set SRVs only

D. (1) Lowering
(2) LO-LO Set SRVs only

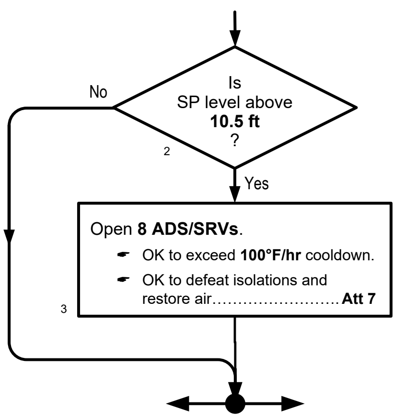
Answer: B
Explanation: Per EP-5 Flooded RPV Indications for SRV tail pipe is “Lowering SRV tailpipe temperatures”

Flooded RPV Indications

NOTE: Some combination of the following conditions should be observed as water fills the steam lines and flows through SRVs. No single indication can be relied on in all events.

- Rising RPV pressure
- Lowering SRV tailpipe temperatures
- If injecting from the suppression pool, SP level lowers as the RPV and steam lines are flooded, then stabilizes when the steam lines are full.
- If injecting from outside the CTMT, SP level rising.
- MSL or RCIC high flow trips.
- If MSIVs are open, two phase flow audible near the main steam tunnel, main steam chest, or main turbine valves.

When EP-5 was entered one of the first directions is to Emergency Depressurize the RPV. That direction is to open 8 ADS valves.



Open 8 ADS/SRVs

RPV depressurization can most easily and rapidly be performed by opening SRVs. Those associated with the ADS function are used first, consistent with automatic system functions. The ADS SRVs are generally considered the most reliable and are arranged to provide uniform distribution of the heat load around the suppression pool. If one or more ADS valves cannot be opened, other SRVs are opened until the total number of open SRVs equals the number of SRVs dedicated to ADS function.

‘B’ is correct

Distracters:

‘A’ is wrong – Even though water will be exiting the SRV the Water temperature is lower that the steam temp therefore, tailpipe temp will lower not rise, the EP-5 says to open 8 ADS valves not LO-LO set valves.

‘C’ is wrong – Even though water will be exiting the SRV the Water temperature is lower that the steam temp therefore, tailpipe temp will lower not rise.

'D' is wrong – EP-5 says to open 8 ADS valves not LO-LO set valves.

K/A Match

This question requires the applicant to have the ability to monitor tail pipe temperatures during ADS valve operation.

Technical References:

02-S-01-40, EP TECHNICAL Bases, Rev 9

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

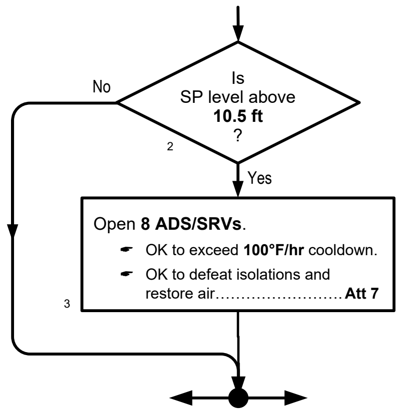
GLP-OPS-EP5, OBJ 7

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(14)	
Level of Difficulty:	3.0	

PRA Applicability:

ADS is listed as #6 on the System Importance to CDF.

Steps 2 and 3



Discussion

The initial steps of EP-5 depressurize the RPV. The depressurization is performed for the following reasons:

The SRVs are used to remove decay heat.

RPV injection flow is maximized, thus shortening the time required to flood the RPV.

Flow through any break which may exist is reduced.

Dynamic loads imposed upon the SRVs and downstream piping when RPV water level reaches the main steam lines are minimized.

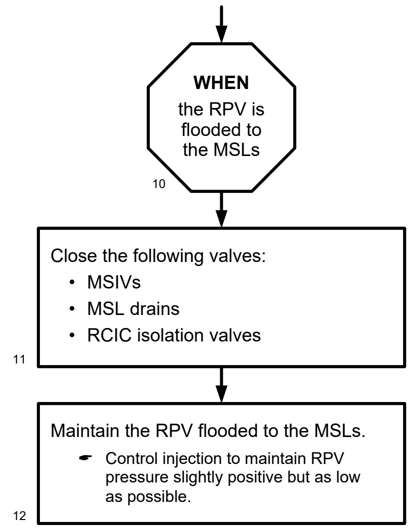
The depressurization steps are similar to those performed in the Emergency Depressurization contingency of EP-2.

Open 8 ADS/SRVs

RPV depressurization can most easily and rapidly be performed by opening SRVs. Those associated with the ADS function are used first, consistent with automatic system functions. The ADS SRVs are generally considered the most reliable and are arranged to provide uniform distribution of the heat load around the suppression pool. If one or more ADS valves cannot be opened, other SRVs are opened until the total number of open SRVs equals the number of SRVs dedicated to ADS function.

The direction to "Open 8 ADS/SRVs" requires manual action, even if the valves are already open on high pressure; automatic valve operation in the relief mode does not accomplish the objective of this step, even if low-low set has actuated. Direct manual control must be established to ensure that the valves remain open as RPV pressure decreases and will reopen if necessary to prevent RPV repressurization.

Steps 10 through 12



Discussion

If it can be determined that RPV water level has reached the elevation of the main steam lines, the core must be cooled by submergence and the objective of RPV flooding has been achieved. Injection into the RPV may then be reduced and controlled to maintain the flooded condition. Any steam lines still open are isolated to minimize the thermal-hydraulic loads on downstream equipment.

Indications that the RPV is flooded are listed adjacent to the step. While no single indication can be conclusively relied upon in all events, it is expected that some combination of the listed conditions will be observed as water fills the lines:

Flooded RPV Indications

NOTE: Some combination of the following conditions should be observed as water fills the steam lines and flows through SRVs. No single indication can be relied on in all events.

- Rising RPV pressure
- Lowering SRV tailpipe temperatures
- If injecting from the suppression pool, SP level lowers as the RPV and steam lines are flooded, then stabilizes when the steam lines are full.
- If injecting from outside the CTMT, SP level rising.
- MSL or RCIC high flow trips.
- If MSIVs are open, two phase flow audible near the main steam tunnel, main steam chest, or main turbine valves.

Examination Outline Cross Reference	Level	RO
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off K1. Knowledge of the physical connections and/or cause effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: K1.14 Containment drainage system	**Tier**	2
	Group #	1
	K/A	223002 K1.14
	Rating	2.8
	Revision	0
Revision Statement:		

Question: 43

The Plant has experienced a RPV leak inside the Drywell.

Drywell pressure is currently 1.5 psig.

What is the status of the **Containment** floor and equipment drain systems?

	Floor Drain	Equipment Drain
A.	In recirc	In recirc
B.	OFF	OFF
C.	Operating	OFF
D.	OFF	Operating

Answer: B

Explanation:

When Drywell pressure exceeds 1.23 psig a Containment isolation occurs. This action causes Containment and Floor drain system to isolate their discharge

'B' is correct

Distracters:

'A' is wrong – Plausible, the Drywell Floor and equipment drain system can go into a recirc mode if temperature of the water is elevated not the Containment sump systems.

'C' & 'D' are wrong – Both systems will isolate not just one.

K/A Match

This question requires the applicant to have the knowledge of the interrelationship between Containment Floor and Equipment drain system and the Containment isolation system.

Technical References:

04-1-01-P45-1, Rev 20

GLP-OPS-P4500, Rev 20

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P4500, OBJ 7

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(9)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Containment Building Sumps

Containment Building Equipment Drain Sump

Containment Floor/Equipment/Chemical Drain Sumps and Pumps (3.8)

Figure 1

3.8 The Containment Equipment Drain Sump, 135' Elevation, CTMT, receives drainage through gravity drains from equipment in CTMT. This sump has no overflow line and is vented to the Containment.

The sump pumps discharge to the Auxiliary Building Equipment Drain Transfer Tank, through a manual isolation valve (F244), and two air-operated CTMT isolation valves (F067 and F068) to the Auxiliary Building Equipment Drain Transfer Tank.

Describe the interlocks and trips associated with the Floor & Equipment Drains System. (7.4)

7.4 The sump pumps are interlocked such that, if either CTMT isolation valve (F067 or F068) is closed or a high level condition exists in the Auxiliary Building Equipment Drain Transfer Tank, the pumps will not auto start and will trip if running.

The Containment Building Equipment Drain sump has two temperature elements (TE-N06301/2) that monitor sump temperatures. If temperature reaches 120°F, temperature switch TS-N665 provides a CTMT EQUIP DR SMP TEMP HI alarm on P680.

Containment Building Floor Drain Sump

State the Purpose of the Containment Floor/Equipment/Chemical Drain Sumps and Pumps (3.8)

Figure 4

3.8 The Containment Building has a floor drain sump located on 135' Elevation that receives drainage through gravity drains from floor drains in the CTMT.

The sump pumps route the waste through a manual isolation valve (F245) and two air-operated isolation valves (F061 and F062) to the Auxiliary Building Floor Drain Transfer Tank.

Describe the interlocks and trips associated with the Floor & Equipment Drains System. (7.4)

♫ *Conservatism – question conditions that are out of the ordinary, unexpected, or that could erode margins to operating the plant conservatively.*

7.4 The sump pumps are interlocked such that, if either CTMT isolation valve (F061 or F062) is closed or a high level condition exists in the Auxiliary Building Floor Drain Transfer Tank, the pumps will not auto start and will trip if running.

The Containment Building Floor Drain Sump has two temperature elements that monitor sump temperature and provide a temperature signal to BOP process instrument cabinet. At 120°F, CTMT FLOOR DR SMP TEMP HI alarm annunciates on P680.

Containment Building Chemical Waste Sump

Containment Floor/Equipment/Chemical Drain Sumps and Pumps (3.8)

Describe the interlocks and trips associated with the Floor & Equipment Drains System. (7.4)

Figure 7

3.8 The Containment Building has a 950 gallon chemical waste sump, located on 161' Elevation that receives chemical wastes from components inside the Containment.

The sump pumps route the wastes through check valve F093, manual isolation valve F243, and two air-operated isolation valves (F098 and F099) to the Auxiliary Building Chemical Waste Sump. Between check valve F093 and isolation valve F243, the discharge from the Drywell Chemical Waste Sump connects to the common header.

7.4 The sump pumps are interlocked such that, if either Containment Isolation valve (F098 or F099) or the Auxiliary Building Isolation valve (F163) is closed, the pumps will not auto start and will trip if running.

Drywell Sumps

Drywell Equipment Drain Sump

Drywell Floor/Equipment/Chemical Drain Sumps and Pumps (3.9)

Figure 1

3.9 The Drywell Equipment Drain Sump is located on 93' Elevation of the Drywell and receives drainage from various equipment in the drywell. The sump is vented to the drywell and has no overflow line.

The sump pumps discharge wastes through air-operated valves F006A/B to the Containment Equipment Drain Sump discharge line, through two air-operated Drywell isolation valves (F009 and F010).

State the location of the Leak Detection Timers. (8.3)

8.3 The timers can be manually reset from the Leak Detection System Division 1 panel, 1H13-P632, 189' Elevation, Control Building.

Isolation Valves

Containment Isolation Valves

Describe the Containment/Drywell Isolation Valve closure signals associated with the Floor & Equipment Drains System. (7.2)

♫ *Monitoring – verify and report automatic system actuations or response.*

7.2 Various P45 valves isolate on a containment/drywell isolation (Group 6). The setpoints are:

1.23 psig Drywell pressure

-41.6" Reactor level

NSSSS Manual Isolation pushbuttons

The valves affected are:

F003 DRWL FLOOR DR SMP DISCH

F004 DRWL FLOOR DR SMP DISCH

F009 DRWL EQUIP DR SMP DISCH

F010 DRWL EQUIP DR SMP DISCH

F061 CTMT FLOOR DR SMP DISCH

F062 CTMT FLOOR DR SMP DISCH

F067 CTMT EQUIP DR SMP DISCH

F068 CTMT EQUIP DR SMP DISCH

F096 DRWL CHEM WST SMP DISCH

F097 DRWL CHEM WST SMP DISCH

F098 CTMT CHEM WST SMP DISCH

F099 CTMT CHEM WST SMP DISCH

F273 AUX BLDS XFER TANKS CTMT ISOL VLV

F274 AUX BLDG XFER TANKS CTMT ISOL VLV

State the location(s) from which the Containment/Drywell Isolation Valves of the Floor & Equipment Drains System can be operated. (6.2)

Containment/Drywell/Auxiliary Building Isolation Valve Closure signal Automatic Bypass (7.2)

7.2 F273 and F274 isolation signal is automatically bypassed 30 seconds following the isolation.

Examination Outline Cross Reference	Level	RO
239002 Relief/Safety Valves K4. Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: K4.02 Minimizes containment fatigue duty cycles resulting from relief valve cycling during decay-heat-dominant period late in an isolation transient (LLS logic): Plant-Specific	**Tier**	2
	Group #	1
	K/A	239002 K4.02
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 44

The plant is at rated power early in core life, when the MSIVs close causing a reactor scram.

Reactor pressure reaches 1120 psig resulting in multiple SRVs cycling.

Following the initial transient, the SRV(s) would be expected to automatically maintain reactor pressure between:

A. 926 and 1033 psig

B. 936 and 1073 psig

C. 946 and 1113 psig

D. 1013 and 1113 psig

Answer: A
Explanation: Low-Low Set initiates automatically when RPV pressure reaches SRV F051D's normal lift setpoint of 1103 psig. Once initiated, six SRVs are capable of operating in the Low-Low Set mode with the following adjusted opening and closing setpoints: One SRV, F051D, lifts at 1033 psig and blows down to 926 psig. A second SRV, F051B, lifts at 1073 psig and blows down to 936 psig. Only one SRV is needed to maintain pressure after the initial transient. 'A' is correct
Distracters:

‘B’ is wrong – Plausible, the F051B will open and close at these setpoints

‘C’ & ‘D’ are wrong – Both indicate the opening and closing setpoints of other lo-lo set valves and other relief setpoints for SRVs

K/A Match

This question requires the applicant to have the knowledge of the lo-lo set system and the reason for it to provide pressure control to minimize the cycles of SRVs.

Technical References:

GLP-OPS-E2202, Rev 9

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2202, OBJ 11.2

Question Source:	**Bank # 1086**	2014 NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

The SRVs can operate in two modes; as safety valves and as relief valves.

As a safety valve, the SRV mechanically lifts at a preset spring-set pressure.

As a relief valve, each SRV is power actuated by a pressure sensing device which in turn operates solenoid air valves to operate a pneumatic piston/cylinder and linkage assembly to open the valve.

With the SRVs open, steam from the Reactor is routed to the Suppression Pool, resulting in the depressurization of the Reactor.

Low-Low Set is sometimes written as Lo-Lo Set such as in component names.

A Low-Low Set (LLS) feature is used in conjunction with the relief function to provide sustained depressurization of the RPV through a limited number of SRVs.

Manual initiation of the SRVs is accomplished from the Control Room or the Remote Shutdown Panel.

When an initiation signal is received, air from the Instrument Air System opens the SRVs (except for the SRVs that are operated by the ADS Air System).

Figure 3

With the SRVs open, steam from the Reactor is routed to the Suppression Pool.

MAJOR COMPONENT DESCRIPTION

NOTE: Unless otherwise noted, all component numbering is prefixed with B21.

ADS/Safety Relief Valves

State the purpose of the ADS/Safety Relief Valves. (5.1)
State the locations of the Safety Relief Valves. (6.1)

Figure 2

5.1, 6.1 Twenty Safety Relief Valves, located on the main steam lines between the flow elbow and the flow restrictor within the drywell, operate to perform the following protection functions:

Relief operation actuates pneumatic operator. These setpoints do not consider Low-Low Set.

Overpressure **Relief** Operation

All twenty valves, initiating at predefined setpoints, are capable of automatically opening to limit an RPV pressure rise.

One valve opens at 1103 psig and closes at 1003 psig.
Ten valves open at 1113 psig and close at 1013 psig.
Nine valves open at 1123 psig and close at 1023 psig.

This effectively changes the opening setpoint of two SRVs and the closing setpoint of six SRVs in the relief mode	11.1 Low-Low Set (LLS) is a feature of the Relief Mode that meets the containment design basis which assumes that only one SRV reopens and cycles after the initial pressure transient from a full MSIV closure isolation event. In LLS, the lowest set point SRV's opening and closing set pressures are farther apart than those for its normal Relief Mode resulting in fewer actuations to remove decay heat following an MSIV closure isolation. LLS limits cyclic transients on the SRVs, SRV tailpipes, quenchers and the Containment structure around the quenchers.
	10.4, 12.1 Low-Low Set initiates automatically when RPV pressure reaches SRV F051D's normal lift setpoint of 1103 psig.
	For each mode of operation of the SRVs, state the number of valves which open and the associated opening setpoints. (11.2) Describe the interlocks associated with the Low-Low Set Logic. (12.1) Statewhich SRVs are operated in Low-Low Set and ADS modes. (13.0)
	11.2,12.1, 13.0 Once initiated, six SRVs are capable of operating in the Low-Low Set mode with the following adjusted opening and closing setpoints:
When LLS is initiated, F051B should also open (i.e., 1103 psig initiated LLS and F051B opening is adjusted down to 1073).	11.2, 12.1, 13.0 One SRV, F051D, lifts at 1033 psig and blows down to 926 psig. 11.2, 12.1, 13.0 A second SRV, F051B, lifts at 1073 psig and blows down to 936 psig.
F051B and these four SRVs are in the group of ten SRVs that have a normal relief setpoint of 1113 psig.	11.1, 12.1, 13.0 The following four SRVs lift at 1113 psig and blow down to 946 psig: F047D F047G F051A F051F
Operate at normal relief setpoints.	11.2 The following SRVs, not impacted by the initiation of Low-Low Set, open at 1113 psig and re-close at 1013 psig: F047A F047C F047H F047L F051K
These nine are the eight F041 valves plus F051C.	11.2 The remaining nine SRVs, likewise not impacted by the initiation of Low-Low Set, open at 1123 psig and re-close at 1023 psig.

Examination Outline Cross Reference	Level	RO
259002 Reactor Water Level Control System A3. Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: A3.04 Changes in reactor feedwater flow	Tier	2
	Group #	1
	K/A	259002 A3.04
	Rating	3.2
	Revision	0
Revision Statement:		

Question: 45

The reactor is operating at rated power.

The C34 Feedwater Flow transmitter fails full upscale to 12.75 mlbm/hr.

What effects will directly result from this condition?

A. The feedwater Master level controller will switch to MANUAL.

B. The feedwater system will automatically lower feedwater flow.

C. The feedwater system will automatically transfer to single-element control.

D. An "estimated" feedwater flow signal replaces the failure transmitter with no change in control.

Answer: C
Explanation: "Hard-failures" of feedwater flow signals are responded to by immediately de-selecting and disabling three-element control.
Distracters: 'A' is wrong – Plausible because if all four level channels hard fail the level control stations will automatically swap to MANUAL. 'B' is wrong - Plausible if the upscale feed flow signal is sensed and used as a good input to the level control system. High feed flow the system would attempt to reduce feed flow to anticipate a rise in level due to feed flow steam flow mismatch. 'D' is wrong - Plausible because an estimated feedwater flow can replace the normal total feedwater flow signal and three-element control can be re-selected and returned to service.
K/A Match

This question requires the applicant to have the knowledge of hard failure on the RPV level control system.
Technical References: GLP-OPS-C3400 Obj 6.3
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-C3400 Obj 6.3

Question Source:	**Bank # 1198**	2015 NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

State the consequences of a feedwater flow channel failure.(6.3)

- ***Figure 4B***
- ***Feedwater flow channel output is 4 – 20 mA, equivalent to 0 – 12 mlb$_m$/hr range. Results in a 0.75 mlb$_m$/hr /mA ratio. Failure signal is a low of 3 mA or a high of 21 mA, equal to –0.75 mlb$_m$/hr and 12.75 mlb$_m$/hr, respectively.***
- ***"Greater than normal" and "estimated" feedwater flow signal to be discussed in detail later.***

Either of the feedwater flow channels is considered to have "hard-failed" when the associated output signal is less than –0.75 mlb$_m$/hr or greater than 12.75 mlb$_m$/hr.

6.3 "Hard-failures" of feedwater flow signals are responded to by immediately de-selecting and disabling three-element control.

As with the steam flow signal, a feedwater flow "soft-failure" may simply be an instrument drift; however, if the "soft-failure" is significant enough, the feedwater flow signal may be unreliable.

Such is the case when the difference between the two feedwater flow signals exceeds 0.8 mlb$_m$/hr greater than normal for ≥2 seconds with plant power greater than 4.95 mlb$_m$/hr.

6.3 This significant "soft-failure" results in the immediate de-selecting and disabling of three-element control.

However, if the signal failure progresses to the "hard-failure" criteria and power is greater than 8.25 mlb$_m$/hr, an "estimated" feedwater flow signal replaces the normal total feedwater flow signal and three-element control can be re-selected and returned to service.

Examination Outline Cross Reference	Level	RO
261000 Standby Gas Treatment System K1. Knowledge of the physical connections and/or cause effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: K1.08 Process radiation monitoring system	Tier	2
	Group #	1
	K/A	261000 K1.08
	Rating	2.8
	Revision	0
Revision Statement:		

Question: 46

Which of the following will cause a Standby Gas Treatment system ‘B’ initiation?

A. Fuel Pool Sweep exhaust rad monitor ‘B’ indicating 31 mrem/hr
Fuel Pool Sweep exhaust rad monitor ‘D’ INOP trip

B. Fuel Pool Sweep exhaust rad monitor ‘B’ indicating 7.9 mrem/hr
Fuel Pool Sweep exhaust rad monitor ‘C’ indicating 8.1 mrem/hr

C. Fuel Handling Area exhaust rad monitor ‘B’ indicating 5.3 mrem/hr
Fuel Handling Area exhaust rad monitor ‘C’ INOP trip

D. Fuel Handling Area exhaust rad monitor ‘A’ indicating 4.2 mrem/hr
Fuel Handling Area exhaust rad monitor ‘C’ mode switch in STANDBY

Answer: C
Explanation: For SBGT to initiate on process rad monitor system the ‘A’ systems must see A and D monitors and ‘B’ must see B and C monitors, for Fuel handling area > 3.6 mrem/hr or INOP OR Fuel pool sweep >30 mrem/hr or INOP.
Distracters: ‘A’ is wrong, due to B and D monitors will not initiate either system ‘B’ is wrong, due to B and C monitors did not reach their setpoint of 30 mrem/hr. ‘D’ is wrong, due to A and C monitors will not initiate either system
K/A Match This question requires the applicant to have the knowledge of the interrelationship with process radiation monitoring system and the SBGT system.

Technical References: GLP-OPS-T4801, Rev. 10		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-T4801 Obj 8.7		
Question Source:	**Bank # 747**	2014 NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(11)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Lesson Content	Instructor Notes
7.4, 8.6 Each logic system may be manually initiated from Control Room Panel P870 using the associated SGTS System Initiate pushbuttons.	*EO 7.4 and 8.6* *Figure 5*
Two pushbuttons for each logic train, both of which must be simultaneously depressed to manually initiate the associated filter train. SGTS DIV 1 MAN INIT, LOGIC A and LOGIC C SGTS DIV 2 MAN INIT, LOGIC B and LOGIC D	*Div. 1 pushbuttons on P870-2B; Div. 2 pushbuttons on P870-8B*
8.6 The Div. 1 Logic System automatically initiates SGTS Subsystem A if any of the following conditions exist:	*EO 8.6* *Figure 8*
Fuel Handling Area Ventilation Exhaust channels A and D, HI-HI Rad/Inop (D17-K617A and D), 3.6 mR/hr. Fuel Pool Sweep Exhaust channels A and D HI-HI-Rad/Inop (D17-K618A and D), 30 mR/hr. LOCA signal from the Nuclear Steam Supply Shutoff System (NSSSS)	*Two pushbuttons for each NSSSS divisional logic, both of which must be depressed to manually initiate the associated SGTS filter train*
Low-low reactor water level (B21-K148A and D), -41.6″. High drywell pressure (C71-K4A and D), +1.23 psig. Manual NSSSS isolation from Control Room Panel P680 NSSSS manual isolate pushbuttons “A” and “D” (B21-K35A and D).	
8.6 The Div. 2 Logic System automatically initiates SGTS Subsystem B if any of the following conditions exist: Fuel Handling Area Ventilation Exhaust channels B and C, HI-HI Rad/Inop (RITS-K617B and C), 3.6 mR/hr. Fuel Pool Sweep Exhaust channels B and C HI-HI-Rad/Inop (RITS-K618B and C), 30 mR/hr. LOCA signal from the Nuclear Steam Supply Shutoff System (NSSSS) Low-low reactor water level (B21-K148B and C), -41.6″. High drywell pressure (C71-K4B and C), +1.23 psig. Manual NSSSS isolation from Control Room Panel P680 NSSSS manual isolate pushbuttons “B” and “C” (B21-K35B and C).	*EO 8.6* *An additional auto start feature related to the AUTO/STBY Mode Select Switch will be discussed shortly.*

Examination Outline Cross Reference	Level	RO
262001 A.C. Electrical Distribution 2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	**Tier**	2
	Group #	1
	K/A	262001 2.4.21
	Rating	4.0
	Revision	0
Revision Statement:		

Question: 47

Division 1 Diesel Generator is currently in MAINT MODE.

The voltage to ALL ESF busses drops to 3290 volts.

The voltage transient duration is 10 seconds and then voltage returns to normal.

All other plant parameters are normal.

(1) Which one of the following states the condition of the ESF busses after this voltage transient?

And

(2) What is the status of the ECCS systems?

A. (1) 15AA and 16AB are energized by their Diesel Generators
(2) All ECCS systems are available

B. (1) 16AB and 17AC are energized by their Diesel Generators
(2) All ECCS systems are available

C. (1) 16AB is energized by its Diesel Generator only
(2) Only Division 2 and 3 ECCS systems are available

D. (1) 17AC is energized by its Diesel Generator only
(2) Only Division 2 and 3 ECCS systems are available

Answer:	**C**
Explanation:	

This is a 79% BUV which after 9 seconds will cause a Div 1 & 2 Shed causing those buses to be transferred to the EDG; however, the 88% Div 3 BUV will not shed its bus unless the condition persists for 5 minutes. With Div 1 D/G in MAINT the D/G for 15AA will not start and 15AA will remain deenergized. Therefore Division 2 and 3 ECCS are the only ones available.

'C' is correct

Distracters:

'A' is wrong, the 15 bus will not energize from its D/G and Div 1 ECCS is not available

'B' is wrong, Div. 3 D/G did not get a start signal so 17AC is still energized through normal power feed and Div 1 ECCS is not available.

'D' is wrong, Div. 3 D/G did not get a start signal so 17AC is still energized through normal power feed

K/A Match

This question requires the applicant to have the knowledge to use the given information and determine the status of the ECCS safety systems.

Technical References:

GLP-OPS-R2100 Objective: 12 & 15

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-R2100 Obj 12 & 15

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

R21 ESF is listed as # 8 on the System Importance to CDF.
Div 1& 2 EDGs are listed as #11 on the System Importance to CDF.

Lesson Content	Instructor Notes
- At GGNS LSS utilizes 2 signals:	
i. [12] 90% BUV (≈ 3790 volts) with 9 sec TD	*Objective 12*
• This signal trips the incoming Offsite Bus Feeder Breakers.	
ii. [12] 70% BUV (≈ 2768 volts) with 0.5 sec TD	*Objective 12*
• Starts the associated DG	
• Sends a BUV Load Shed Signal	
• If 90% BUV is not present, LSS trips the Incoming Offsite Bus Feeder Breakers.	
• Provides an input to the DG Output Breaker closure permissive.	
- [14] The LSS Panels will initiate a BUV Load shed, removing the loads listed in E1120 "BUV SHED" column.	*Objective 14* *E1120 shows which loads shed on a BUV signal*
▪ [13]LSS LOP Shed	
- The LSS Panels receive a LOP signal in addition to the BUV signal.	*Figure 7 shows the devices that cause a LOP shed*
- [13, 18] A **LOP SHED** is when the 3 Offsite Feeders to the individual ESF Bus senses NO VOLTAGE between the 152-1900 series and the Individual ESF Bus Feeder Breakers.	*Objectives 13, 18* *Anticipation of loss of cooling water flow from P42 and P44 to specific components*
- This sends additional signals to specific valves to realign.	
i. [32] The LOP portion of the shed realigns interface valves to allow Standby Service Water to provide cooling functions normally performed by BOP Systems.	*Objective 32*
- The LOP signal originates from the 127F relays on the primary side of the 15AA and 16AB ESF bus feeder breakers.	*Point out all feeders to the bus must be de-energized to initiate a LOP shed*
i. 152-1501, 1511, 1514 for Division I	
ii. 152-1601, 1611, 1614 for Division II	
- They provide indication that there is NO Offsite power available to the ESF Busses.	*SOER 99-1 Recommendation 5*
- [14, 18] The actions of LSS on a LOP Shed are similar to those for a BUV shed.	*Objectives 14, 18* *Use E1120 to show loads shed on a LOP signal*

3.22 Bus Undervoltage Start signals

3.22.1 (73% undervoltage for 2.3 secs) Trips all three offsite supply breakers to Bus 17AC immediately, then after 2.3 secs, diesel generator auto-starts **AND** ties onto bus.

3.22.2 (88% undervoltage for four secs) Starts five minute timer. **IF** voltage remains below 88% for the duration of the five minute timer, **THEN** all 3 offsite feeder breakers to 17AC trip Open **AND** D/G auto-starts **AND** ties onto bus.

3.22.3 (LOCA with 88% BUV for > four secs) Bypasses five minute timer, trips all offsite feeder breakers to Bus 17AC **AND** D/G auto-starts **AND** ties onto bus.

3.23 Governor oil level Should be **CHECKED** before **AND** periodically during engine runs. ("B" Engine only) Since governor oil level varies with engine load, **USE** the following criteria for determining acceptable levels: Diesel Generator governor oil level Should be maintained within red band on sightglass housing when engine is shutdown. It is expected for governor oil level to drop off **AND** stabilize with level visible in sightglass when engine is started. With engine running, governor oil level Should be maintained visible in sightglass. **IF** governor oil addition is required during non-emergency conditions, **THEN** oil addition May be made with engine running before synchronization **OR** shutdown diesel generator **AND** add oil. During emergency conditions, oil May be added to governor with engine running; However, oscillations in generator load May occur. Oil addition with engine running Should be made to bring level back to black line on sightglass.

3.24 **WHENEVER** HPCS D/G output breaker is racked back in after being racked out, D/G Should be run **AND** tied to bus to ensure breaker operability.

3.25 **IF** Diesel Generator control cabinet doors are left open **OR** unlatched, they Must **NOT** be left unattended. Otherwise, the respective Diesel Generator Should be **DECLARED** inoperable.

3.25.1 Diesel generator panel doors May be open for test or work but Should **NOT** be left unattended.

3.25.2 Diesel generator panel doors May **NOT** be left open for periods of time greater than 8 hours without assessing seismic effects on the panel.

3.25.3 **IF** a seismic event should occur with the panel doors open during testing or work in progress, **THEN** **PERFORM** the following Immediately after the seismic event:

a. **MAINTAIN** the door in a stable configuration.

b. **INSPECT** panel doors for damage due to the seismic event.

c. **REPORT** any unusual observations to the control room.

3.25.4 ATTENDED is defined as "present at the panel close enough such that during a seismic event the required actions can be performed immediately." Reference CR 2017-12264.

Examination Outline Cross Reference	Level	RO
262002 Uninterruptable Power Supply (A.C./D.C.) K6. Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : K6.03 Static inverter	Tier	2
	Group #	1
	K/A	262002 K6.03
	Rating	2.7
	Revision	1
Revision Statement: **Per Ops Rep, Added first part of the question.**		

Question: 48

DC bus 11DD has been lost.

(1) Which of the following describes the actions on the UPS inverter system?

(2) Upon restoration of 11DD to the normal power supply, using 04-1-01-L62-1, Static Inverters SOI, what action should the operator perform to ensure return to normal operation?

A. (1) Inverter 1Y79 swaps to alternate source
(2) Verify 1Y79 automatically swaps to Normal Supply

B. (1) Inverter 1Y79 swaps to alternate source
(2) Depress INVERTER TO LOAD pushbutton on 1Y79 control panel

C. (1) Inverter 1Y88 swaps to alternate source
(2) Verify 1Y88 automatically swaps to Normal Supply

D. (1) Inverter 1Y88 swaps to alternate source
(2) Depress INVERTER TO LOAD pushbutton on 1Y88 control panel

Answer:	**A**

Explanation:

The candidate must be able to determine which Inverter has lost power by knowing which is ESF and which is BOP powered.

SOI P&L 3.4 The Static (automatic) switch <u>will</u> automatically transfer back to the normal supply (inverter supply load) upon restoration of normal power. The ESF Static Inverters **<u>AND</u>** 1Y99 Inverter do **<u>not</u>** have the Auto Retransfer feature back to DC Power. **<u>If</u>** DC Power is lost, the BATTERY INPUT breaker has an under voltage trip associated with it.

1Y79 is powered from 11DD and a BOP inverter, 1Y88 is powered from ESF batteries and an ESF inverter.

‘A’ is correct

Distracters:

‘B’ is wrong, 1Y79 is a BOP inverter and will auto transfer to normal power

‘C’ & ‘D’ are wrong, 1Y88 is an ESF inverter and must be manually transferred back to normal power.

K/A Match

This question requires the applicant to have the knowledge to determine the difference between an ESF and BOP inverter and the effect on loss of an inverter.

Technical References:

04-1-01-L62-1
GLP-OPS-L6200

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-L6200, Obj. 17

Question Source:	**Bank # 1242**	Not used on an NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 After de-energizing an inverter cabinet, **WAIT** at least 60 seconds before restarting inverter **OR** entering cabinet.

3.2 The IN SYNC light Should be on anytime the applicable inverter is in operation **AND** alternate source power is available. **NEVER** operate Manual Bypass switch with Both sources energized unless **EITHER** the IN SYNC light is On **OR** the alternate source is supplying load through Static switch. The synchronizing circuit Will prevent out-of-SYNC transfers unless there is an inverter failure since a lack of transfer May be worse than an out-of-SYNC transfer.

3.3 The inverter Should **NOT** be allowed to operate with a DC input voltage less than 105 Vdc. Maximum of 140 Volts DC on ESF inverters 1Y88, 1Y87, 1Y95, 1Y96.

3.4 The Static (automatic) switch Will automatically transfer back to the normal supply (inverter supply load) upon restoration of normal power. The ESF Static Inverters **AND** 1Y99 Inverter do **NOT** have the Auto Retransfer feature back to DC Power. **IF** DC Power is lost, the BATTERY INPUT breaker has an under voltage trip associated with it.

3.5 Inverters Should **NOT** be operated without the battery connected; **OTHERWISE**, damage to the power supply **AND/OR** loads Could result. Before opening a BATTERY OUTPUT breaker, **ENSURE** the associated inverters are swapped to ALTERNATE SOURCE.

3.6 BOP Inverter 1Y99 has backlit INVERTER TO LOAD pushbutton **AND** a backlit ALTERNATE SOURCE TO LOAD pushbutton.

3.7 BOP Inverters 1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y100 **AND** 1Y101 have trouble alarms On SH13-P807.

3.8 **COORDINATE** **ANY** change of power that affects CAS, SAS **OR** the C83 system with the GGNS Manager Security/designee **AND/OR** the On-duty Security Shift Supervisor, **AND** ensure that security has implemented any compensatory measures **IF** required.

3.9 RE1 **AND** RE2 Core Monitoring computers are powered form 1Y91 **AND** 1Y74. **COORDINATE** with Reactor Engineering panel isolations.

Examination Outline Cross Reference	Level	RO
263000 D.C. Electrical Distribution K2. Knowledge of electrical power supplies to the following: K2.01 Major D.C. loads	Tier	2
	Group #	1
	K/A	263000 K2.01
	Rating	3.1
	Revision	0
Revision Statement:		

Question: 49

The plant is operating at rated conditions.

The following alarms are received:

- P680-9A-D11, GEN SEAL OIL TROUBLE
- P680-10A-C10, GEN H2 SEAL OIL PUMP C FAULT

Which of the following is the DC Electrical bus that was lost?

A. 11DB

B. 11DD

C. 11DE

D. 11DF

Answer: D
Explanation: 04-1-01-1-L11 Attachment 1F shows the power supply for the DC SEAL OIL PUMP C MOTOR. 11DF is the only 250VDC system 'D' is correct
Distracters: All distractors are other DC buses, but all are 125VDC.
K/A Match This question requires the applicant to have the knowledge of the power supply for major DC loads.

Technical References:

04-1-01-L11-1, Attachment 1F, 125V DC BUS 11DF LOAD LIST

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-L1100 Obj 6.2, 8.3

Question Source:	**Bank # 86**	Not used on an NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:

125 VDC ESF is listed as #7 on the System Importance to CDF.

125V DC BUS 11DF LOAD LIST
1DF1

BREAKER NUMBER	COMPONENT NUMBER	DESCRIPTION	SCHEMATIC DRAWING
72-11F01		SPARE	E-1027
72-11F02		SPARE	E-1027
72-11F03		SPARE	E-1027
72-11F04		SPARE	E-1027
72-11F05	1P43-C002	TBCW DC EMER WTR COOLING PMP	E-1027
72-11F07	N34	DC LUBE OIL PUMP	E-1027
72-11F08	1N42-C005C	DC SEAL OIL PUMP C MOTOR	E-1027
72-11F09	N21	RFPT EMER BEARING OIL PUMP N21-C002A	E-1027
72-11F10	N21	RFPT EMER BEARING OIL PUMP N21-C002B	E-1027

Examination Outline Cross Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) A2. Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.01 Parallel operation of emergency generator	**Tier**	2
	Group #	1
	K/A	264000 A2.01
	Rating	3.5
	Revision	0
Revision Statement:		

Question: 50

Div 1 D/G has auto started due to an inadvertent signal and is running unloaded.

The CRS has directed you to parallel the Div 1 D/G to an offsite power source per the SOI.

To adjust Div 1 D/G frequency within range of bus frequency, use _____(1)______ handswitch so that synchroscope is rotating SLOWLY in ___(2)___ direction.

A. (1) DG GOV MAN CONT
   (2) SLOW

B. (1) DG GOV MAN CONT
   (2) FAST

C. (1) DG 11 VR MAN CONT
   (2) FAST

D. (1) DG 11 VR MAN CONT
   (2) SLOW

Answer: **B**
Explanation: Per 04-1-01-P75-1 section 4.2, "**ADJUST** Standby Diesel Generator 11 [12] speed to bring frequency within range of bus frequency by **USING** DG 11[12] GOV MAN CONT so that synchroscope indicator is **ROTATING SLOWLY** in the FAST direction (clockwise). 'B' is correct
Distracters:

‘A’ is wrong, but plausible. The answer states to move in the FAST direction. IF the D/G was carrying the bus alone and an offsite power source was being paralleled to it then the scope would be moving in the FAST direction.

‘C’ is wrong, but plausible. This switch is used but for adjusting voltage and VARS not speed. And The answer states to move in the FAST direction. IF the D/G was carrying the bus alone and an offsite power source was being paralleled to it then the scope would be moving in the FAST direction.

‘D’ is wrong, but plausible. This switch is used but for adjusting voltage and VARS not speed.

K/A Match

This question requires the applicant to have the knowledge of procedure requirements for emergency diesel generators being paralleled to an offsite source.

Technical References:

04-1-01-P75-1, Standby Diesel Generator SOI section 4.2, Rev 112

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P7500 Obj 17

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

Div 1 & 2 EDGs are listed as #11 on the System Importance to CDF.

4.2.2 (Cont.)

d. **SELECT** phase of bus voltage to be **MONITORED** with VM 4.16 kV BUS 15AA [16AB] VOLTMETER handswitch.

e. **PLACE** Standby Diesel Generator 11 [12] Output Breaker Synchronizing switch SYN CONT FDR BKR 152-1508 [152-1608] handswitch to ON.

f. **PLACE** DG 11 [12] PRL CONT handswitch momentarily to PRL (spring return to OFF) to defeat parallel interlock **AND** **PLACES** governor in the "DROOP" mode.

(1) **ENSURE** AMBER Light above DIV 1 (2) LSS TEST MODE SEL Switch Comes ON.

g. **ADJUST** Standby Diesel Generator 11 [12] INCOMING VOLTS DIV 1 [2] about 50 volts above RUNNING VOLTS DIV 1 [2] with DG 11[12] VR AUTO SET PT CONT handswitch.

h. **ADJUST** Standby Diesel Generator 11 [12] speed to bring frequency within range of bus frequency by **USING** DG 11[12] GOV MAN CONT so that synchroscope indicator is **ROTATING SLOWLY** in the FAST direction (clockwise).

> NOTE
>
> Diesel Generator Output Voltage Should be maintained by Automatic Voltage Regulator, **BUT** a **CHECK** Should be kept on Diesel Generator Output voltage **AND** frequency.

i. There May be a need to adjust Diesel Generator speed to obtain proper rotation of synchroscope indicator. Once indicator is rotating in the correct direction (clockwise) **ALLOW** it to make a few revolutions to **ENSURE** frequency stability.

Examination Outline Cross Reference	Level	RO
300000 Instrument Air System (IAS) A4. Ability to manually operate and / or monitor in the control room: A4.01 Pressure gauges	Tier	2
	Group #	1
	K/A	300000 A4.01
	Rating	2.6
	Revision	0
Revision Statement:		

May 2017 NRC Exam Q52

Question: 51

The plant is operating at rated power.

Which of the following identifies the control room panel where operators can read INSTRUMENT AIR SUPPLY HEADER PRESSURE, and what is its value for these conditions?

A. Panel P854; normally reads approximately 110 psig

B. Panel P854; normally reads approximately 135 psig

C. Panel P870; normally reads approximately 110 psig

D. Panel P870; normally reads approximately 135 psig

Answer: C

Explanation:

There is only one Instrument Air Supply Header Pressure indicator in the control room; it is located on P870 and normally reads about 110 psig. See 02-S-01-31, Control Room Rounds two sheets (one for Instrument Air at P870 page 15 of 40, the other for Service Air at P854 page 13 of 40).

‘C’ is correct

Distracters:

‘A’ is wrong - Panel P854 is plausible because that is where operators control two of the Plant Air Compressors (PAC ‘B’ and ‘C’) and is the location of the indicator for Service Air Header Pressure

'B' is wrong – Panel P854 is plausible because that is where operators control two of the Plant Air Compressors (PAC 'B' and 'C') and is the location of the indicator for Service Air Header Pressure. 135 psig is plausible because this is the "unload" setpoint for the in-service ("LEAD") PAC. .

'D' is wrong - normal header pressure is 110psig, 135 psig is plausible because this is the "unload" setpoint for the in-service ("LEAD") PAC

K/A Match

This question requires the applicant to have ability to monitor the Instrument Air pressure in the control room by knowing location of indication and normal value.

Technical References:

GLP-OPS-P5100, Plant Air System lesson plan, Rev. 8
02-S-01-30, Control Room Rounds, Rev. 50

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P5100 Obj 18

Question Source:	**Bank #**	**May 2017 NRC Exam Q52**
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Control Room Rounds

Item	Instruction	Entry	Shift	Day	Min	Max
TBCW Discharge Header Pressure	Record TBCW discharge header pressure on P870 P43R601.	PSIG	All	All	42	
Instrument Air Header Pressure	Record Instrument Air header pressure on P870 P53R602.	PSIG	All	All	90	
PCW Discharge Header Pressure	Record PCW discharge header pressure on P870 P71R600.	PSIG	All	All	76	
CST Level	Record CST level on P870 P11R601.	Ft	All	All	25	29
P870 MOV Test Switches	Are all MOV test switches on P870 in normal?		All	All		

If "No" notify CRS and explain in Comments

Item	Instruction	Entry	Shift	Day
SSW-B Status	Is SSW-B in Standby? (Yes, No)		All	All
SSW B Valve Positions	Are the following valves in their required positions: P41-F006B Open P41-F001B Closed P41-F018B Closed P41-F014B Closed P41-F068B Open P41-F005B Closed (Yes, No) (NA if SSW B is not in Standby)		All	All

If "No" Notify CRS that SSW B valves are not in their required Standby positions.

Lesson Content	Instructor Notes
## Instrument Air Pressure Reducing Station	
7.4 The Instrument Air Pressure Reducing Station is located in the Water Treatment Building, 133' elevation, Area 31.	***EO–7.4*** ***Also called "Letdown Station"***
6.4 The Instrument Air Pressure Reducing Station serves as the interconnect between the Plant Air System (P51), on its upstream side, and the Instrument Air System (P53), on its downstream side.	***EO–6.4*** ***Figure 14***
The station uses one of two in-service pressure control valves to maintain the Instrument Air header at some nominal pressure.	***Figure 15***
18.0 This station has two pressure control valves: a "primary" valve (P51-F503) and a "backup" valve (P51-F504).	***EO-18*** ***M-1126***
The primary valve is an electro-pneumatic one. It is normally in-service, is INFI-90 controlled, and uses a signal from pressure transmitter P53-PT-N060 (on the Instrument Air header side of the station) to automatically maintain header pressure at a 110 psig setpoint.	***M-1067A*** ***♫ Knowledge – Understand system and component purpose and design***
The backup valve is strictly a pneumatic one. It has its own pneumatic controller. The controller is set to maintain 100 psig, as sensed at the downstream side of the backup valve itself.	
This design makes the backup valve essentially self-actuating. Normally kept closed by the proper operation of the primary valve (maintaining the 110 psig header pressure), it will automatically open and begin to control at its setpoint when pressure sensed at the valve drops to 100 psig due to a primary valve/controller malfunction.	
The station also has a manual bypass valve.	
## Service Air Pressure Reducing Station	
7.4,10.3 The Service Air Pressure Reducing Station is located in the Water Treatment Building, 133' elevation, Area 31.	***EO–7.4,10.3*** ***Also called "Letdown Station"***
6.4 The Service Air Pressure Reducing Station serves as the interconnect between the Plant Air System (P51), on its upstream side, and the Service Air System (P52), on its downstream side.	***EO–6.4*** ***Figure 16***
This station has only one pressure control valve (P51-F501). This electro-pneumatic valve is INFI-90 controlled and uses a signal from pressure transmitter P52-PT-N060 (on the Service Air header side of the station) to automatically maintain header pressure at a 110 psig setpoint.	***M-1126C*** ***M-1068D***

Examination Outline Cross Reference	Level	RO
400000 Component Cooling Water System (CCWS) A3. Ability to monitor automatic operations of the CCWS including: A3.01 Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	Tier	2
	Group #	1
	K/A	400000 A3.01
	Rating	3.0
	Revision	0
Revision Statement:		

Question: 52

The plant is operating at rated conditions.

A transient occurs on the CCW Pumps.

Currently only one (1) CCW pump is running.

CCW header temperature is 96°F.

All subsequent actions from the Loss of CCW ONEP have been completed.

Which of the following, in addition to Reactor Recirculation Pumps, describes the components that still have CCW flowing through them?

A. Control Rod Drive Pump oil coolers ONLY

B. Fuel Pool Cooling and Cleanup heat Exchangers ONLY

C. RWCU Non Regen Heat Exchanger and Control Rod Drive Pump oil coolers ONLY

D. RWCU Non Regen Heat Exchanger and Fuel Pool Cooling and Cleanup heat Exchangers ONLY

Answer:	**A**
Explanation: Per the ONEP If only one pump is running then isolate the CCW to Fuel Pool HT EX. and isolate CCW to RWCU non-regen Ht EX. This is RO knowledge because it covers the overall mitigating strategy of the ONEP. Also, with only one pump running a low flow condition will occur and automatically cause the FPCCU heat exchangers to isolate. 'A' is correct	

Distracters:

'B' is wrong – FPCCU would be isolated due to performing the actions of the onep

'C' is wrong – RWCU would be isolated due to performing the actions of the onep

'D' is wrong - RWCU would be isolated due to performing the actions of the onep and FPCCU would already be isolated by performing actions of the onep

K/A Match

This question requires the applicant to have ability to monitor the automatic actions on a loss of a CCW pump.

Technical References:

GLP-OPS-P4200
05-1-02-V-1, Loss of CCW ONEP

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P4200 Obj 10

Question Source:	**Bank # 1205**	2015 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Lesson Content	Instructor Notes
Normal CCW System flow through FPCCU Heat Exchanger B, in flowpath order, is through F105, F203, F028B, F032B, F204, and F205.	*Figures 1 and 5*
7.6 Valves F105, F203, F204, and F205 can be opened only if both F200B and F201B are full closed.	*EO 7.6*
7.6 Valves F028B and F032B can be opened only if both F200B and F201B are full closed.	
7.6 Valves F105, F203, F204, and F205 receive auto close signals if either valve F028A or F032A is not full closed and a low flow condition of <1000 gpm to the FPCCU Heat Exchangers exists.	♫ ***Operator Fundamental; Control, discuss two handed operations; know expected response and consequences***
7.6 Valves F028B and F032B receive auto close signals if a low flow condition of <1000 gpm to the FPCCU Heat Exchangers exists.	
7.6 When initially valving in FPCCU Heat Exchanger B,	
F028A and F032A are confirmed to be full closed, which allows opening of F105, F203, F204, and F205 with a low flow signal present, and	***Holding switches in OPEN prevents auto closure (switch not in AUTO position).***
F028B and F032B handswitches must be held in the OPEN position until low flow condition clears.	
Low flow to FPCCU Heat Exchangers is indicated by receipt of two Control Room panel P870 annunciators, both labeled FP HX CCW FLO LO; however, one receives inputs from FIS-N600A/N601A and the other receives inputs from FIS-N600B/N601B.	
If required, SSW A can be aligned to provide cooling to FPCCU Heat Exchanger A, as follows:	*Figures 1 and 5*
The heat exchanger is isolated from CCW by closing F105, F203, F204, and F205.	***Actions directed in P42 SOI - note that closing F204 also isolates CCW from heat exchanger B.***
CCW to/from FPCCU Heat Exchanger A valves, F028A and F032A, remain in the open position and SSW to/from FPCCU Heat Exchanger A valves, F200A and F201A, are opened.	

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
Continued from Page 13 D. Partial loss of CCW	**Continued from Page 13** ****CAUTION 3**** D6. **ISOLATE** CCW to Fuel Pool Cooling Heat Exchangers **PERFORM** the following actions: a. **CLOSE** 1P42-F105, on 1H13-P870-2C. b. **CLOSE** 1P42-F205, on 1H13-P870-2C. D7. **IF** CCW Temperature CANNOT be maintained <95°F, **THEN ISOLATE** CCW to RWCU; **PERFORM** the following actions: a. **IF** Reactor Water Temperature is ≥ 130°F, **AND** RWCU Demineralizers are IN SERVICE, **THEN TRIP** the RWCU Pumps. b. **IF** Reactor Pressure is > 120 psig, **THEN TRIP** the RWCU Pumps. c. **CLOSE** 1P42-F103 RWCU HX ISOL VLV, Elevation 161' RWCU HX Room Entrance. D8. **IF** the RWCU Inlet Conductivity Monitor is Inoperable, **THEN NOTIFY** Chemistry. D9. **ROTATE** CRD pumps to **MAINTAIN** oil temperature < 135°F, **USE** SOI 04-1-01-C11-1. D10. **IF** CRD pump operation is required for RPV LEVEL control, **THEN ROTATE** CRD pumps to **MAINTAIN** oil temperature < 180°F, **USE** SOI 04-1-01-C11-1. **END OF SECTION**

Examination Outline Cross Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) K1. Knowledge of the physical connections and/or cause effect relationships between HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) and the following: K1.02 Suppression Pool: BWR-5,6 .	**Tier**	2
	Group #	1
	K/A	209002 K1.02
	Rating	3.5
	Revision	0
Revision Statement:		

Question: 53

A valid SUPP POOL LVL HI annunciator is received on the P601 panel at rated power.

Consider the following valves:

- E22-F015, HPCS pump Suppression Pool Suction Valve
- E22-F001, HPCS pump CST Suction Valve

The HPCS system will respond to this event by ________(1)_______ .

After Suppression pool level is returned to normal the operator will ______(2)_______.

A. (1) opening E22-F015 and then closing E22-F001
   (2) manually realign the HPCS pump suction to the CST

B. (1) opening E22-F015 and then closing E22-F001
   (2) verify the HPCS pump suction automatically realigns to the CST

C. (1) closing E22-F001 and then opening E22-F015
   (2) manually realign the HPCS pump suction to the CST

D. (1) closing E22-F001 and then opening E22-F015
   (2) verify the HPCS pump suction automatically realigns to the CST

Answer: **A**
Explanation: Per ARI P601-16A-C5 (SUPP POOL LVL HI), the suction swap occurs by opening E22-F015 and then closing E22-F001. Although it is possible to manually realign the suctions at any point, a caution in the ARI reads "Restore normal suppression pool level before realigning HPCS suction to the CST." ‘A’ is correct

Distracters:

‘B’ is wrong – HPCS suctions will not auto realign

‘C’ is wrong – F001 will not close first the system will maintain a suction path, also it receives a close signal from the F015 valve position not Supp Pool Level

‘D’ is wrong - F001 will not close first the system will maintain a suction path, also it receives a close signal from the F015 valve position not Supp Pool Level and HPCS suctions will not auto realign

K/A Match

This question requires the applicant to have the knowledge of the HPCS’s physical connection with the Suppression Pool and the cause and effect of suppression pool level.

Technical References:

GLP-OPS-E2201
ARI P601-16A-C5

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2201 Obj 9

Question Source:	**Bank # 563**	Not on previous NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:

HPCS is listed as #16 on the System Importance to CDF.

Lesson Content	Instructor Notes
• <u>HPCS Pump Suction from Suppression Pool, F015 - CLOSED</u>	
▪ The HPCS Pump Suction from Suppression Pool Valve, F015, is used to isolate the HPCS pump suction from the Suppression Pool.	Objective 5.3 Figures 1 and 2
- Normally closed, 24-inch, carbon steel, motor-operated, gate valve.	
- Located in the HPCS Pump Room (Area 8, Elevation. 93').	Objective 6.3
▪ With F015 fully open, momentarily placing the handswitch in the CLOSE position will shut the valve.	Objective 9.2 Figure 7
▪ Taking the switch to the OPEN position, opens the valve provided: - the valve is fully closed and - the HPCS Inboard and Outboard Test Return to CST Valves, F010 and F011, are fully closed.	
▪ In the AUTO position, the valve opens upon receipt of a low CST level signal or a high Suppression Pool level signal. - When F015 reaches full open, the CST suction valve, F001, automatically closes.	Objective 9.2 ♫Monitoring: Maintain awareness of critical parameter status.
▪ When CST level drops to 5.0 feet, as indicated by level indicating switch LIS-N654C or G, F015 opens and a CST LVL LO annunciator is received at P601-16A-C4. - CST level is monitored by transmitters LT-N054C and G.	Table 2
▪ When Suppression Pool level rises to 5.9 inches (18' 9" actual level), as indicated by either level indicating switch LIS-N655C or G, F015 opens and a SUPP POOL LVL HI annunciator is received at P601-16A-C5. - The Suppression Pool level is monitored by LT-N055C and G.	Table 2

Lesson Content	Instructor Notes
• HPCS Suction Valve from the CST, F001 - OPEN	Standby lineup position follows the valve name.
▪ The HPCS Pump Suction from CST Valve, F001, isolates the CST from the HPCS pump suction. - Normally open, 18-inch, motor-operated, gate valve, constructed of carbon steel.	Objective 5.2 Figures 1 and 2
- Located outside the HPCS Pump Room (Area 8, Elevation 93').	Objective 6.2
▪ With F001 fully open, momentarily placing its CLOSE/AUTO/OPEN switch in the CLOSE position will shut the valve. - However, if the valve is stroking open, momentarily placing the switch in the CLOSE position will have no effect until it is full open. - The open and close relays in the control circuits for all the HPCS motor operated gate valves are interlocked with each other to prevent reversing the motor's direction while it is stroking the valve.	Objective 9.1 Figure 7
▪ Placing the switch momentarily in the OPEN position, will open the valve provided: - it is in the fully closed position and, - the HPCS Pump Suction from Suppression Pool Valve is NOT fully open.	Objective 9.1
▪ In the AUTO position, the valve opens upon receipt of an initiation signal if the HPCS Pump Suction from Suppression Pool Valve (F015) is not fully open. - Also in AUTO, the valve will close if the HPCS Pump Suction from Suppression Pool (F015) reaches full open.	Objective 9.1

SUPP POOL
LVL
HI

04-1-02-1H13-P601-16A-C5	
Revision: 151	Page 1 of 2

Safety Related

Alarm Device E22-LAH-L603

1.0 POSSIBLE CAUSES

1.1 Suppression pool level of 18'9" due to:

1.1.1 HPCS running on minimum flow to the suppression pool.

1.1.2 RCIC running on minimum flow to the suppression pool.

1.1.3 Improper valve lineup to the suppression pool.

1.1.4 Upper containment pool dump to suppression pool.

2.0 AUTOMATIC ACTION

2.1 HPCS pump suppression pool suction valve F015 opens.

2.2 HPCS pump CST suction valve F001 closes **WHEN** F015 is full open.

2.3 HPCS test return valves to the CST F010 (Breaker 52-170107 normally open) **AND** F011 close.

3.0 IMMEDIATE OPERATOR ACTION

3.1 **VERIFY** that automatic actions occur.

3.2 **CHECK** that HPCS pump flow is normal **IF** HPCS is in operation.

3.3 **CHECK** suppression pool level indication on panel 1H13-P870.

CAUTION

Manually overiding the CST/suppression pool suction back to the CST Will make HPCS inoperable in Modes 1, 2 AND 3 per TS 3.3.5.1 Condition D.

Restore normal suppression pool level before realigning HPCS suction to the CST.

Examination Outline Cross Reference	Level	RO
201001 Control Rod Drive Hydraulic System K4. Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following: K4.08 Controlling control rod drive header pressure	Tier	2
	Group #	2
	K/A	201001 K4.08
	Rating	3.1
	Revision	1
Revision Statement: **Rev 1, Facility Rep, F002 is mentioned in only one answer, added the mention of F002 in answer 'B'**		

Question: 54

Given the following CRD system components:

C11-F003, CRD DRIVE WTR PRESS CONT VLV
C11-F002, CRD FLOW CONT VLV

Which of the following describes how CRD Drive Water D/P is maintained during reactor pressurization?

A. F003 is adjusted manually ONLY to maintain D/P and flow until rated pressure is achieved in the RPV.

B. F003 automatically adjusts position to maintain D/P while the F002 automatically positions to maintain flow during pressurization.

C. F003 is manually set to provide the required D/P initially then the automatic positioning of the F002 will maintain D/P during pressurization.

D. F003 automatically adjusts position to maintain system flow which in turn will maintain the required D/P until pressurization is complete.

Answer: **C**
Explanation: During a reactor startup and pressurization the drive water D/P is set at 250 psid. As reactor pressure rises CRD flow begins to drop due to higher pressure in the RPV, the CRD FCV F002 will open to maintain set flowrate of 60 gpm. When the F002 opens to provide more flow the D/P will rise back to the set value. Therefore, with the flow control valve in automatic, set at 60 gpm, and the pressure control valve set at 250 psid, as reactor pressure increases, the flow control valve opens to provide more flow and maintain the appropriate drive water Δp. 'C' is correct

Distracters:

‘A’ is wrong – Only one adjustment is needed to maintain D/P

‘B’ is wrong – F003 is a manual MOV and does not operate in automatic

‘D’ is wrong - F003 is a manual MOV and does not operate in automatic

K/A Match

This question requires the applicant to have the knowledge of the design features of the CRD system to control D/P.

Technical References:

GLP-OPS-C111A, CRD Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C111A Obj 9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

PRESSURE CONTROL STATION

State the purpose of the Pressure Control Station. (3.7)
Identify the location of the Pressure Control Station. (4.6)
State from where the Pressure Control Station can be operated. (6.6)
Describe the operation of the Pressure Control Station. (8.2)

Figure 1

Per the SOI, normal drive water Δp is 225 to 275 psid and cooling water Δp is <20 psid.

3.7, 4.6 The Pressure Control Station, located in Containment on the 135' elevation, maintains normal drive water and cooling water differential pressures.

Pressure Control Station consists of a single pressure control valve, F003, four sets of stabilizing valves, and two differential pressure transmitters.

Drive water pressure can be increased to 475 psid IAW ONEP CRD Malfunctions. Control Room indicator has a maximum indication of 350 psid. For Δp indications of >350 psid, use the PDS computer.

6.6, 8.2 Pressure Control Valve F003, operated from Control Room Panel P601, is used to maintain CRD drive water pressure ~250 psi above reactor pressure while also maintaining CRD cooling water pressure <20 psi above reactor pressure and allowing 54 – 66 gpm cooling water flow.

Excessive drive water pressures can result in internal damage to the CRD mechanisms.

Cooling water pressures >20 psid can result in control rod drifts.

State the purpose of the stabilizing valves. (3.8)
Identify the location of the stabilizing valves. (4.7)
Describe the operation of the stabilizing valves. (8.3)
Describe pressure and flow control in the CRD Hydraulic System and their relationship. (9.0)

Figure 1

3.8, 4.7, 9.0 Four sets of solenoid-operated stabilizing valves, located in Containment on the 135' elevation, automatically open and close in response to control rod insertion or withdrawal signals to maintain a constant system flow.

Prevents variations in system pressure that would occur if the system flow had changed.

9.0 This design allows a normal flow of 4.2 gpm for one valve in a set and 1 gpm for the other valve to bypass the pressure control valve during normal operations when control rods are not being moved.

9.0 During control rod movement, the valves close, thus allowing flow to the drive water header used for control rod positioning without changing total system flow.

SYSTEM OPERATION

Discuss pressure and flow control in the CRD Hydraulic System and their relationship. (9.0)

Using the drawings of the CRD Hydraulic System and the HCUs, describe a reactor scram in detail. (10.3)

Concerning SOI 04-1-01-C11-1: (12.0)

Identify precautions, limitations, cautions, warnings and notes that apply to a given situation. (12.1)

Identify the reasons for the precautions, limitations, cautions, warnings and notes given in the SOI. (12.2)

From memory, draw each of the following CRD Hydraulic System flow paths: (13.0)

Normal system flowpath (13.1)

HCU flowpath (13.2)

12.1, 12.2 Using the SOI, discuss the precautions and limitations, cautions, warnings, and notes and reasons for each.

Review the various modes of operation for the CRD Hydraulic System, including the following Normal, Infrequent and Abnormal Operations:

13.1, 13.2 Using the SOI and CRD Hydraulic drawings, as appropriate, discuss the following:

CRD Hydraulic System normal flowpath.

HCU flowpath.

Discuss the operational relationship of the Pressure Control Station and Flow Control Station during reactor startup and pressurization.

9.0 With the flow control valve in automatic, set at 60 gpm, and the pressure control valve set at 250 psid, as reactor pressure increases, the flow control valve opens to provide more flow and maintain the appropriate drive water Δp.

10.3 Using the SOI and CRD Hydraulic drawings, as appropriate, discuss the overall operation of the CRD Hydraulic System in response to a reactor scram.

For each HCU, the RPS System de-energizes the scram pilot valve (F417) and vents the air from the scram inlet (F126) and outlet (F127) valves.

The scram outlet valve opens first then the inlet valve to admit accumulator pressure to the drive piston, pushing it inward.

At the same time, the SDV pilot solenoid valves (F009 and F182) open to vent the air header causing the SDV vents and drains to close.

Examination Outline Cross Reference	Level	RO
202001 Recirculation System K1. Knowledge of the physical connections and/or cause effect relationships between RECIRCULATION SYSTEM and the following: K1.15 Nuclear boiler instrumentation (reactor water level/pressure)	Tier	2
	Group #	2
	K/A	202001 K1.15
	Rating	3.2
	Revision	0
Revision Statement:		

Question: 55

The plant is operating at rated conditions.

A RPV level control problem causes reactor water level to lower.

A Reactor Scram occurs on low reactor water level.

Reactor water level reached -32 inches and was recovered to normal level.

Which of the following describes which Reactor Recirculation Pump breakers are CLOSED?

A. CB-3A/B

B. CB-1A/B and 2A/B.

C. CB-1A/B, 2A/B and 5A/B.

D. CB-1A/B, 2A/B, 3A/B and 4A/B.

Answer: **D**
Explanation: The given conditions indicates level less than +11.4 inches which will cause CB-5A/B to trip. The scram will cause reactor power to lower below the bypass for EOC/RPT, therefore 3 & 4 breakers will not open. Reactor parameters are in transient but did NOT exceed the ATWS/ARI setpoints of -41.6”. If this setpoint had been reached then no breakers would be closed ‘D’ is correct

Distracters:

'A' is wrong – This would be correct if a ATWS/ARI system initiation. ATWS/ARI will trip CB-1,2, 4, and 5 leaving 3 closed.

'B' is wrong – CB-3 and CB-4 will still be closed due to +11.4 inches only trips CB-5

'C' is wrong – CB-5 will trip level reaches +11.4 inches

K/A Match

This question requires the applicant to have the knowledge of the Recirc system trip setpoints from the nuclear boiler system (level and pressure).

Technical References:

GLP-OPS-B3300, Reactor Recirc Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-B3300 Obj 27.2, 28.2

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Lesson Content	Instructor Notes
End-of-Cycle Recirculation Pump Trip (EOC-RPT)	
27.1 The End-of-Cycle Recirculation Pump Trip (EOC-RPT) logic automatically downshifts (transfers) the Recirculation Pumps from fast to slow speed in response to a closure of the turbine stop valves or fast closure of the turbine control valves.	***EO-27.1*** ***Figure 20A***
In the event of a turbine control valve fast closure or turbine stop valve closure at the end-of-core life with all of the control rods nearly full out, the control rods alone cannot insert sufficient negative reactivity fast enough to ensure the reactor core thermal limits are not exceeded.	
27.2 Rapidly inserting negative reactivity into the core by reducing core flow raises void content and counters the positive reactivity insertion due to the reactor pressure rise from the turbine stop or control valves closure.	***EO-27.2*** ***Refer to LER 2000-006***
27.3 The EOC-RPT logic is actuated by the Reactor Protection System (RPS) using the same turbine stop and control valve signals that initiate a reactor scram: Turbine stop valve trip fluid < 40 psig, or Turbine control valve fluid pressure < 46.0 psig	***EO-27.3*** ***The actual Reactor Protection System (RPS) sensors monitor N32 system fluid pressures for the respective stop and control valves.***
EOC-RPT logic automatically seals in for three seconds. This ensures that the EOC-RPT logic gets tripped before the trip units reset.	
27.2 The logic is arranged so that only a full scram condition caused by either the turbine stop or control valves closure will cause an EOC-RPT actuation.	***EO-27.2***
The EOC-RPT logic has two divisions; Division 1 trips Recirculation Pump breaker CB3A and B and Division 2 trips CB4A and B.	
Both divisions initiate fast to slow speed transfer sequences for both Recirculation Pumps.	***Same as with "cavitation interlocks."***
Indications of EOC-RPT logic actuation are provided by:	***Annunciators P680-4A1-A4/A5***
Actuation of EOC/RPT A/B TRIP annunciators on Control Room Panel P680.	

Lesson Content	Instructor Notes
Anticipated Transient Without Scram (ATWS) Alternate Rod Insertion/Recirculation Pump Trip (ARI/RPT)	
28.1, 28.2 The ATWS Alternate Rod Insertion/Recirculation Pump Trip (ATWS-ARI/RPT) circuit is designed to trip the Recirculation Pumps in the unlikely event that the control rods do not completely insert when a reactor scram is required.	***EO-28.1; EO-28.2*** ***Note that this is a complete trip, not a transfer to the LFMG, as for EOC-RPT.***
Tripping the Recirculation Pumps causes rapid void formation in the core. Void formation inserts negative reactivity and mitigates the consequences of a control rod insertion failure.	***The ARI portion provides an independent means of venting the scram air header to scram the control rods.***
28.3 The ATWS ARI/RPT circuit actuates if reactor pressure exceeds 1126 psig or reactor water level drops to -41.6 inches or it can be manually initiated by arming and depressing both ARI/RPT pushbuttons on Control Room Panel P680.	***EO-28.3*** ***Additional details of ARI portion included in CRD Hydraulic System Lesson Plan.***
28.2 Upon actuation, the circuit trips the slow and fast speed circuit breakers (CB1, 2, 4, and 5) for both Recirculation Pumps.	***EO-28.2***
The ARI/RPT logic is divided into two trip channels, arranged in a two-out-of-two-once logic so that a single failure of one level, pressure, or manual initiation pushbutton can neither cause nor prevent actuation of the ATWS ARI/RPT circuit. Channel 1 trips recirculation breakers CB2 and 5 for both pumps. Channel 2 trips recirculation breakers CB1 and 4 for both pumps.	***ARI-RPT trips all Recirculation breakers except CB3.***
28.2 When a trip signal is generated, the Recirculation Pump breakers open, and the Alternate Rod Insertion system is energized, as a backup to RPS, to scram the reactor.	***EO-28.2***

Examination Outline Cross Reference	Level	RO
202002 Recirculation Flow Control System K6. Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM : K6.05 Reactor water level	Tier	2
	Group #	2
	K/A	202002 K6.05
	Rating	3.1
	Revision	0
Revision Statement:		

Question: 56

Which of the following describes the RPV water level setpoint required to initiate a Reactor Recirc Flow Control Valve Runback signal?

A. +4 inches

B. +11.4 inches

C. +32.7 inches

D. +40.7 inches.

Answer: C
Explanation: A reactor recirc FCV runback will occur if <2 feedpumps are running and Reactor Recirc pumps are in FAST speed and RPV water level reaches 32.7”. Which is also the low level (level 4) alarm signal. ‘B’ is correct
Distracters: ‘A’ is wrong – Plausible, this is the level signal used after Feedwater level control initiates a Setpoint Setdown. ‘B’ is wrong – Plausible, this is the level that will cause a Recirc pump downshift to slow speed due to being a cavitation interlock. ‘D’ is wrong – Plausible, this is the HIGH level alarm setpoint.
K/A Match This question requires the applicant to have the knowledge of a loss of Reactor Water level will have on the Recirc system FCV runback.

Technical References: GLP-OPS-B3300, Reactor Recirc Lesson Plan.
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-B3300 Obj 24.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:
None.

Lesson Content	Instructor Notes

The controller allows two rates of change, depending on how far left or right the slide lever is moved:

The first stop allows ~0.16%/second change and the second stop allows ~1.6%/second change.™

Mechanical posts in the path of the lever define the point at which the rates change.

Care should be taken that the lever does not get latched behind the post, which would cause the valve to move at high speed until the lever is unlatched.

In accordance with Technical Specifications, individual loop flows are required to be within 10% of each other when core flow is <70% of rated and 5% of each other when core flow is ≥70% of rated.

The speed of valve operation is flow dependent. Valve is set up to provide a linear flow response through a non-linear valve.

Prevents excessive flow imbalances between the recirculation loops that could damage the jet pumps and prevents large temperature stresses on the reactor components.

Stress could occur when flow is raised in a slow loop, causing cool water from a stagnate location to be replaced with hot water.

24.1 The output signal from each loop flow controller feeds a flow limiter designed to help mitigate the consequences of a reactor feed pump trip at high power levels.

EO-24.1

Figure 19

24.2 By running back the FCVs to 52% of rated core flow, this flow limiter lowers the feedwater demand to within the capacity of one reactor feed pump (~67% of rated thermal power).

EO-24.2

24.3 The limiter is bypassed if the Recirculation Pumps are in slow speed, or if reactor water level is above 32.7", or if both reactor feed pumps are running (determined by PSL-N215A(B)).

EO-24.3

A Runback will occur if all the following conditions are met:

1. The associated Recirc Pump is in Fast Speed
2. Less than 2 Feed Pumps are running
3. Reactor water level at or below Level 4 (see E-1163-42)

Note that when the Feed Pumps are Reset (Trip Alarm Clear) the Runback logic circuit considers the feed Pumps running.

24.4 Once activated, the flow limiter prevents increasing flow above 52% until the flow limiter circuitry has been reset using the RECIRC PUMP A/B CAV INTLK RESET pushbutton on Control Room Panel P680.

EO-24.4

Figure 9

Additional cavitation interlocks will be discussed later.

To prevent FCV repositioning upon resetting the FCV runback, the limiter error must be reduced to zero.

From the limiter, the flow demand signal goes to a function generator where the signal is corrected for any non-linear characteristics of the FCV response.

For example, the function generator may adjust a 40% flow signal down to a 38% valve position signal if this is the valve position that actually yields 40% flow.

Examination Outline Cross Reference	Level	RO
201005 Rod Control and Information System (RCIS) K3. Knowledge of the effect that a loss or malfunction of the ROD CONTROL AND INFORMATION SYSTEM (RCIS) will have on following: K3.01 Control rod drive system: BWR-6	Tier	2
	Group #	2
	K/A	201005 K3.01
	Rating	3.3
	Revision	3
Revision Statement: **Rev 1 – Added in Mode 2 to clarify that Pattern controller is in effect.** **Rev 2 – Added Control rod movement is controlled by the RPC to ensure clarity.**		

Question: 57

A Reactor Startup is in progress in Mode 2.

Control Rod movement is being controlled by the RPC.

As a control rod is being moved to position 20, a reed switch fails and a DATA FAULT is received.

Which of the following describes the ability to drive control rods?

A. INSERT block only

B. WITHDRAW block only

C. INSERT AND WITHDRAW block

D. INSERT AND WITHDRAW block for only selected control rod.

Answer: **C**
Explanation: A DATA FAULT will cause an Insert and Withdraw block due to RP&IS doesn’t know the position of the control rod and must be corrected prior to continued movement to ensure Rod Pattern Control. ‘C’ is correct
Distracters: ‘A’ is wrong – Plausible, this will occur but along with a Withdraw block ‘B’ is wrong – Plausible, this will occur but along with a Insert block.

‘D’ is wrong – Plausible, this will occur but for all control rod motion not just one rod.

K/A Match

This question requires the applicant to have the knowledge of a failure of the RCIS system will have on Control Rod Drive system.

Technical References:

GLP-OPS-C1102, RC&IS Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C1102 Obj 13.2, 13.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Depressing the ROD SELECT CLEAR pushbutton on the OCM deselects any selected control rods and inhibits further selection.

One rod is selected and withdrawn with the reactor MODE switch in REFUEL.

Rods selected and driving (motion timer activated).

Analyzer card comparison or self-test malfunction occurs:

Command and acknowledge words do not agree, or

Rods are moving without an operator command or one or more HCU transponder cards do not respond during self-testing mode.

13.2 Either of the following signals will result in a control rod insertion block, prohibiting further control rod insertion:

BLOCK	SETPOINT	WHEN BYPASSED
Rod Position Information	Data Fault	Substitute position data entered (Note 1)
Rod Pattern Controller	Pattern Violation	Power >LPSP (Note 2)

Note 1

Substitute data can be entered only if all of the prerequisites are met:

Rod with the data fault is selected

INDIVIDUAL ROD DRIVE mode is selected

RAW DATA is not selected

The following rules apply to all data substitutions:

Substitute position cannot be entered into both channels for the same rod address

Good position data always replaces substitute position data

Note 2

The LPSP (Low Power Setpoint) is set at 26% RTP. To clear the block, rod position must be bypassed in RACS and the rod returned to its required pattern position. An RPC-enforced insert block is identified by an energized INSERT BLOCK indicator on the ROD MOTION CONTROL section of the OCM and an energized INSERT INHIBIT indicator on the PATTERN CONT section of the OCM.

13.3 Control rod withdrawal blocks and their setpoints are as follows:

BLOCK	SETPOINT	WHEN BYPASSED
SRM Upscale	>10^5cps	Associated IRM ≥ Range 8 or RUN mode
SRM INOP	• Module unplugged • High voltage low • Function switch not in OPERATE	Associated IRM ≥ Range 8 or RUN mode
SRM Downscale	<0.7 cps	Associated IRM ≥ Range 3 or RUN mode
SRM Wrong Position	Any SRM Detector not full in	SRM counts >106 cps or Associated IRM ≥ Range 3 or RUN mode
IRM Upscale	>86% of scale (108/125)	RUN mode

♫ Monitoring – Track inoperable and degraded technical specification and other equipment important to safety and reliable plant operations.

Refer to Notes 1, 2, and 3 on following page.

IRM INOP	• Module unplugged • High voltage low • Function switch not in OPERATE	RUN mode
IRM Downscale	<4% of scale (5/125)	IRM range 1 or RUN mode
IRM Wrong Position	Any IRM not full in	RUN mode
APRM Upscale	$\geq$12%	RUN mode
APRM Upscale	Operating above the simulated thermal power line of the Power /Flow Map. (0.58 Wd + 54.1% Clamped at108%; 0.58 Wd + 31.3% SLO)	Not in RUN mode
APRM INOP	• Loss of Input Power • <21 LPRMs in OPERATE or less than 3 at required detector level. • Mode Switch in INOP • Self Test Critical Fault • Software / Firmware error (Watchdog Timer timed out)	Never
APRM Downscale	<5%	Not in RUN mode
APRM Flow Upscale	>111% rated flow	Not in RUN mode
Rod Pattern Controller	Pattern Violation	Power >LPSP (Note 1)
Scram Discharge Vol. High Level	$\geq$32 inches	Key switch to BYPASS and Mode Switch in SHUTDOWN or REFUEL
Rod Position Information	Data fault	Substitute position data entered (Note 2)
Rod Withdrawal Limiter	(Note 3)	(Note 3)
Reactor Mode Switch	SHUTDOWN position	Other than SHUTDOWN position
Main Hoist Fuel Grapple	Loaded and platform near or over the core	Not in REFUEL mode
Rod Drive Mode	GANG mode selected	Not in REFUEL mode
One-rod-out interlock	Any rod not full in and another rod selected	Not in REFUEL mode
Refueling Platform	Near or over the core	Not in STARTUP mode

Note 1
The LPSP (Low Power Setpoint) is set at 26% RTP. To clear the block, rod position must be bypassed in RACS and the rod returned to its required pattern position. An RPC-enforced withdraw block is identified by an energized WITHDRAW BLOCK indicator on the ROD MOTION CONTROL section of the OCM and an energized WITHDRAW INHIBIT indicator on the PATTERN CONT section of the OCM.

Examination Outline Cross Reference	Level	RO
216000 Nuclear Boiler Instrumentation AI. Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including: A1.02 Removing or returning a sensor (transmitter) to service	**Tier**	2
	Group #	2
	K/A	216000 A1.02
	Rating	2.9
	Revision	1
Revision Statement: **Rev 1 – Validators did not remember the operation of a DP cell, Changed answer 'C' for clarification.**		

Question: 58

I&C are working on Div 1 RPV Wide Range level transmitter and inadvertently opens the equalizing valve across the transmitter.

RPV level will indicate __________.

A. full upscale

B. full downscale

C. last known indication

D. mid-range of the indicator

Answer: **A**
Explanation: With a level instrument in operation, opening the instrument equalizing valve will cause the instrument to fail full scale as the differential pressure across the transmitter goes to zero. 'A' is correct.
Distracters: 'B' is wrong – Plausible, this will occur if d/p goes high on the transmitter, such as failure of the variable leg of a transmitter. 'C' is wrong – with any change in d/p a change in indication will occur. 'D' is wrong – Plausible, if the student thinks that equalizing the pressure on the transmitter will cause an mid-scale reading.

K/A Match This question requires the applicant to have the knowledge of a failure of a level transmitter when removing from service.
Technical References: GLP-OPS-B2101, RC&IS Lesson Plan.
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-B2101 Obj 12.2, 12.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Instructor Notes	Lesson Content

12.1 EP-2, RPV Control Caution 1 provides guidance on RPV Level instrument use during LOCA conditions as follows:

- If the temperature (in Drywell or Containment) near any RPV level instrument leg reaches the saturation temperature for the existing reactor pressure, that instrument may be unreliable due to boiling in the instrument leg
- If temperatures near the reference leg vertical runs of the Wide Range, Upset and Shutdown Level instruments are above a specified elevated temperature and the instrument is indicating below a specified level, the instrument must not be used.

Under these conditions, actual level could be below the variable leg tap yet still indicating on-scale due to the severe off-calibration conditions.

Instruments should not be able to indicate a trend if actual level rises or lowers, under these conditions.

ATWS

After the Recirc Pumps trip, Fuel Zone level will indicate ~48″ lower than actual level while the reactor is at or near rated pressure.

As the reactor is depressurized, Fuel Zone approaches calibration conditions (except for Drywell temperature, if there is no concurrent LOCA) and indicated level increases to approach actual level.

The difference between the Fuel Zone and Wide Range is reduced as pressure lowers.

For conservatism, GGNS operators are to interpret level with respect to Top of Active Fuel using the Fuel Zone indicator only, unless it is known to be in error.

Leaking or Blocked Instrument Lines

Predict the impact of leaking or blocked instrument lines on the RPV Water Level Instrumentation System. (12.2)

Student Question 2

12.2 Instrument line leaks, caused by valve packing or connection leaks, have caused significant level instrument divergence at GGNS.

Figure 4

12.2 The rate of divergence rises as level in the condensing pot lowers to the narrow portion of the pot. Reference leg purge flow should compensate for most small leaks in the reference leg.

Instructor Notes	Lesson Content

Predict the impact of transmitter isolation and equalization valve misalignments on the RPV Water Level Instrumentation System. (12.3)

12.2, 12.3 With a variable leg blocked or isolated from the reactor vessel, the level instrument will not respond to changes in RPV level. If pressure between the blockage or closed isolation valve and the level transmitter lowers, the indicated level will lower.

12.2, 12.3 If the reference leg is blocked or isolated, the level instrument may gradually fail high as the pressure downstream of the blockage or closed isolation valve (high pressure side of the level transmitter) decays.

12.3 With a level instrument in operation, opening the instrument equalizing valve will cause the instrument to fail full scale as the differential pressure across the transmitter goes to zero.

If the instrument equalizing valve is opened with the instrument unisolated, the reference leg level will equalize with the vessel level failing all instruments using that reference leg to their full-scale condition.

Excessive Condensing Chamber Purge Flow

Predict the impact of excessive condensing chamber purge flow on the RPV Water Level Instrumentation System. (12.4)

12.4 The error in indicated level as a result of purge flow to a condensing chamber error is negligible as long as the purge flow is maintained in the range of 2-6 lb_m/hr. Excessive purge flow can result in indicated level being lower than actual level because it can cause a rise in reference leg pressure.

Excessive purge flows create the potential for:

- Conservatively early low level initiation of ECCS
- Non-conservatively delayed high level initiation of Level 8 RPS trips and HPCS/RCIC isolations

Air or Non-condensables In Instrument Lines

Predict the impact of air or non-condensable gases trapped in instrument lines on the RPV Water Level Instrumentation System. (12.5)

Examination Outline Cross Reference	Level	RO
226001 RHR/LPCI: Containment Spray System Mode K2. Knowledge of electrical power supplies to the following: K2.02 Pumps	Tier	2
	Group #	2
	K/A	226001 K2.02
	Rating	2.9
	Revision	0
Revision Statement:		

Question: 59

A tornado has caused a total loss of offsite power.

Division 1 Diesel Generator has failed to start.

Which of the following indicates the systems available for Containment Spray?

A. RHR ‘A’ only

B. RHR ‘B’ only

C. RHR ‘B’ and ‘C’

D. RHR ‘A’ and ‘B’.

Answer: B
Explanation: The only systems that can be used for Containment spray is RHR ‘A’ and ‘B’. With a loss of offsite power and Div 1 Diesel Gen. bus 15AA is not available. Therefore RHR ‘A’ is not available due to 15AA powers RHR ‘A’ and 16AB powers RHR ‘B’. Bus 16AB should be powered by the Div 2 D/G and available for Containment Spray. ‘B’ is correct
Distracters: ‘A’ is wrong –the student must be able to determine with the given losses that RHR ‘A’ has no power. ‘C’ is wrong – RHR ‘C’ does not have a Containment Spray mode. ‘D’ is wrong – RHR ‘A’ has no power

K/A Match

This question requires the applicant to have the knowledge of power supplies to the RHR pumps with a Containment spray mode.

Technical References:

GLP-OPS-E1200, RHR Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200 Obj 7.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Instructor Notes	Lesson Content
Use Figure 2 to review realignment of RHR valves on LPCI initiation. “Build” from standby configuration to show that only one MOV, F042A/B/C, actually changes position, although several valves receive signals to reposition (in event RHR not in standby configuration).	Upon receipt of a LPCS/LPCI initiation signal, The associated RHR Pumps start, taking a suction on Suppression Pool through Suppression Pool suction strainer and pump suction valve, and discharging through discharge piping, including through the A and B heat exchangers and bypass valves, through the system shutoff valves, up to the injection shutoff valves. As Reactor pressure decreases, the injection shutoff valves open, providing an injection flowpath through the testable check valve and Reactor Vessel and to inside the core shroud where the water floods the core to prevent overheating of the fuel.
	Given an RHR System drawing, trace the Containment Spray flowpath. (3.3)
	3.3 CONTAINMENT SPRAY
	A and B loops of RHR, Initiate automatically in the Containment Spray mode in response to: Receipt of associated Containment Spray automatic initiation signal, or Manual initiation signal via manual initiate pushbuttons (with a high Drywell pressure signal present). May be manually initiated in the Containment Spray mode using individual pump and valve control handswitches.
Use Figure 3 to review realignment of RHR valves on Containment Spray initiation, from both LPCI and standby alignments.	Upon receipt of a Containment Spray initiation signal, the associated RHR Pumps start, taking a suction on the Suppression Pool through the Suppression Pool suction strainer and pump suction valve, and discharging through the RHR Heat Exchangers, discharge piping and valves to the Containment Spray headers and spargers.
“Build” from LPCI configuration to show that only three MOVs change position (F042A/B and F048A/B close and F028A/B open), and only one other valve receives signal to close (F024A/B).	Once initiated, Containment Spray: Reduces Containment pressure by condensing steam, Provides hydrogen control by mixing air volume, Assists in Suppression Pool cooling, and Helps suppress airborne iodine, thus reducing the release of fission products to the environment.

Instructor Notes	Lesson Content
	State the power supplies for the RHR Pumps. (7.1)
	Each pump is driven by a 1000 horsepower, squirrel-cage induction motor powered from the ESF 4160V AC buses.
	7.1 C002A is powered from Bus 15AA; C002B and C are powered from Bus 16AB.
	Each pump motor is equipped with a heater to prevent moisture accumulation in the motor windings when the motor is not running.
	Heater deenergizes when the associated pump breaker closes and re-energizes when the pump breaker opens.
Refer to Figures 16 and 17 and Table 3 to review pump pressure and flow indications.	Table 3 lists RHR Pump/Loop instrumentation: Indication of RHR Pump discharge flow is provided on Control Room panel P601 flow indicators FI-R603A/B/C and on Remote Shutdown Panel flow indicators FI-R200A/B.
ADS to be covered in separate lesson plan.	PIS-N655A/B/C and PIS-N656A/B/C provide ADS pressure permissive signal at 125 psig.
P601-20A-C4, 17A-C1, 17A-C4	PIS-N653A/B/C activates the RHR PMP A/B/C DISCH PRESS ABNORMAL annunciator if discharge pressure increases to 458 psig. Indicates pressurization occurring through leaking valves and alerts the operator of a possible piping overpressurization condition.
P601-20A-C4, 17A-C1, 17A-C4	PIS-N654A/B/C activates the same RHR PMP A/B/C DISCH PRESS ABNORMAL annunciator if discharge pressure decreases to 40 psig for the A and B loops, or 28 psig for the C loop. Indicates a possible leak in the system or problem with the associated jockey pump.
	Local pump suction and discharge pressure is provided by pressure indicators PI-R002A/B/C and PI-R008A/B/C, respectively.
	<u>RHR Pump Minimum Flow Lines</u>
	State the purpose of the RHR Pump Minimum Flow Lines. (4.3) State the location of the RHR Pump Minimum Flow Lines and Valves. (5.3)
	4.3, 5.3 A minimum flow line, located just downstream of each pump and returning to the Suppression Pool, is provided to protect the pump from overheating during low pump flow rate conditions.

Examination Outline Cross Reference	Level	RO
245000 Main Turbine Generator and Auxiliary Systems A4. Ability to manually operate and/or monitor in the control room: A4.05 Generator megawatt output	Tier	2
	Group #	2
	K/A	245000 A4.05
	Rating	2.7
	Revision	0
Revision Statement:		

Question: 60

The plant is operating at rated conditions.

Generator Megawatt output begins to rise.

Which of the following indicates a possible cause?

A. Failed open SRV

B. Failed open Main Steam Bypass Valve

C. Reactor Recirc Flow Control Valve fail closed

D. Feedwater Heater 6A level control failure, level reaches Hi-Hi alarm setpoint

Answer: **D**
Explanation: For MWE to rise at rated conditions a rise in reactor power must also occur. Feedwater heater 6A level reaching the Hi-Hi setpoint will cause a heater isolation in turn causing a loss of feedwater heating and lower inlet subcooling in the core and causing a rise in reactor power which in turn causes a rise in Generator MWE. 'D' is correct
Distracters: 'A' is wrong – A failed open SRV will cause a re-direction of steam flow to the suppression pool and a lowering of steam pressure to the turbine causing the Turbine Control valves to close to maintain pressure and MWE will lower. 'B' is wrong – Same reason as a failed open SRV. 'C' is wrong – A Recirc FCV failing closed will cause a lowering of reactor power therefore a lowering of MWE

K/A Match This question requires the applicant to have the ability to monitor the Generator output MWE and determine the cause and the knowledge of a HI HI level in a feedwater heater.
Technical References: GLP-OPS-ONEP, ONEP Lesson Plan.
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-ONEP Obj 7.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

PLANT OPERATIONS MANUAL

Volume 05
Section 02

05-1-02-V-5
Revision: 120
Date: 04/12/2019

REFERENCE USE

OFF-NORMAL EVENT PROCEDURE

LOSS OF FEEDWATER HEATING

SAFETY RELATED

ALARMS

• HTR DR PMP A TRIP	1H13-P680-1A-A7
• HTR DR PMP B TRIP	1H13-P680-1A-A8
• HTR DR TK LVL HI-HI	1H13-P680-2A-E1
• HTR DR TK LVL LO-LO	1H13-P680-1A-C7
• FW HTR 1A, 1B, 1C LVL HI	1H13-P680-2A-D6, D7, D8
• FW HTR 2A, 2B, 2C LVL HI	1H13-P680-2A-C6, C7, C8
• FW HTR 3A, 3B, 3C LVL HI	1H13-P680-2A-B6, B7, B8
• FW HTR 4A, 4B, 4C LVL HI	1H13-P680-2A-A6, A7, A8
• FW HTR 5A, 5B LVL HI	1H13-P680-2A-B9, B10
• FW HTR 6A, 6B LVL HI	1H13-P680-2A-A9, A10
• FW HTR 1A, 1B, 1C LVL HI-HI	1H13-P870-6A-D1, D2, D3
• FW HTR 2A, 2B, 2C LVL HI-HI	1H13-P870-6A-C1, C2, C3
• FW HTR 3A, 3B, 3C LVL HI-HI	1H13-P870-6A-B1, B2, B3
• FW HTR 4A, 4B, 4C LVL HI-HI	1H13-P870-6A-A1, A2, A3
• FW HTR 5A, 5B LVL HI-HI	1H13-P870-6A-B4, B5
• FW HTR 6A, 6B LVL HI-HI	1H13-P870-6A-A4, A5

SYMPTOMS

- HP Feedwater Heater Hi-Hi level isolation of extraction steam
- LP heater string condensate flow isolation
- APRM / LPRM Upscale alarm coincident with Feedwater Heater Hi-Hi level alarm(s)
- Feedwater heating malfunction results in reduction of final feedwater temperature of ≥10° F
- Heater Drain Pump(s) trip
- RCIC injection into the reactor in Mode 1
- HPCS injection into the reactor in Mode 1
- Main Turbine Bypass Valve Open with Main Turbine On-Line
- Main Steam Safety / Relief Valve Open with Main Turbine On-Line

ADDITIONAL INFORMATION

- A low pressure feedwater heater isolation signal is valid when the heater condensate isolation valves begin to close.
- The RCIC discharge into feedwater is located downstream of the final feedwater temperature indication.
- The severity of a loss of feedwater heating condition will vary. It is dependent upon which heater(s) are isolated, reactor power, and Main Turbine load. At higher load conditions, the resultant feedwater temperature reduction will be sufficient to result in a significant reactor power increase.
- Operation in Region II of Attachment 2 is outside of the licensing basis analyses, for MCPR limits. The MCPR limits cited in TS 3.2.2 are NOT valid when the combination feedwater temperature and reactor core power is in Region II of the Feedwater Temperature vs Percent Rated Core Thermal Power curve in Attachment 2.
- The OPRM setpoints and the backup stability protections (alternate methods) are NOT valid when the combination feedwater temperature and reactor core power is in Region II of the Feedwater Temperature vs Percent Rated Core Thermal Power curve in Attachment 2.

Monitored Parameters	**Monitoring Location**
• ONEP Guide	PDS Menu
• Reactor Power	PDS Point C34NA011
• Total Core Flow	PDS Point B33NA001 OR 1B33-UR-R613 (1H13-P680-3B)
• Final Feedwater Temperature A	PDS Point B21N041A.C88
• Final Feedwater Temperature B	PDS Point B21N041B.C88
• RFP A Suction Pressure	PDS Point N21N078A
• RFP B Suction Pressure	PDS Point N21N078B

Examination Outline Cross Reference	Level	RO
256000 Reactor Condensate System K4. Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following: K4.08 Dedicated ECCS water supply: Plant-Specific	Tier	2
	Group #	2
	K/A	256000 K4.08
	Rating	3.6
	Revision	0
Revision Statement:		

Question: 61

How much water is reserved in the CST for HPCS and RCIC operation?

What design feature ensures this amount is maintained?

A. 50,000 gallons
Level control system

B. 50,000 gallons
Internal CST standpipe

C. 170,000 gallons
Level control system

D. 170,000 gallons
Internal CST standpipe

Answer: **D**
Explanation: Of the total CST capacity, approximately 170,000 gallons is reserved for use by HPCS and the Reactor Core Cooling System because of physical piping design and layout. A standpipe is used inside the CST for other systems that use the CST as a source. 'D' is correct
Distracters: 'A' is wrong – Plausible, 50,000 is the amount of water left in the CST when the Low CST level HPCS/RCIC suction swap occurs. The CST does not have a level control system. 'B' is wrong – Plausible, 50,000 is the amount of water left in the CST when the Low CST level HPCS/RCIC suction swap occurs.. 'C' is wrong – The CST does not have a level control system.

K/A Match This question requires the applicant to have the ability to monitor the Generator output MWE and determine the cause and the knowledge of a HI HI level in a feedwater heater.
Technical References: GLP-OPS-E2201, HPCS Lesson Plan. GLP-OPS-P1100, Condensate and Refueling Water Storage and Transfer Lesson Plan
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-E2201 Obj 5

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	
PRA Applicability:		
HPCS and RCIC are #16 and #17 on the System Importance to CDF.		

NOTES	LESSON BODY

The Refueling Water Storage and Transfer Subsystem has one large capacity storage tank, two transfer pumps, piping, and instrumentation necessary for operation.

The Refueling Water Storage and Transfer Subsystem provides the water handling requirements for the Upper Containment Pool and the Cask Storage Pool during refueling and cask loading. The water from either pool is drained to the RWST for temporary storage.

After refueling, any excess water in the RWST is drained to the waste surge tanks to be processed by the Liquid Radwaste System before being transferred to the CST.

The Refueling Water Transfer Pumps can be lined up to take suction on the Suppression Pool and pump it to either the RWST or the Suppression Pool Cleanup System.

MAJOR COMPONENT DESCRIPTION

Condensate Storage Tank (CST)

- State the purpose of the Condensate Storage Tank. (4.1)
- State the location of the CST. (5.1)

Figures 1, 2, 5, 6, and 7

4.1 The CST, with a capacity of ~300,000 gallons, stores the water reserves necessary for makeup during plant operations.

5.1 Located south of the Auxiliary Building in a pit, the CST is the normal makeup water source for the main condenser hotwell.

> Of the total CST capacity, approximately 170,000 gallons is reserved for use by HPCS and the Reactor Core Isolation Cooling System because of physical piping design and layout. The CST level transmitters for RCIC are tapped off of the CST suction line in lieu of a static sensing line. As a result of friction losses, the low CST suction swaps to the suppression pool are dependent on flow and occur higher than intended, resulting in less usable volume. Less usable volume is not an issue since the suppression pool is the normally credited water source.

The water required for the RCIC and HPCS Systems is supplied through one of the many lines coming from and going to the CST.

Lesson Content	Instructor Notes
1. Leak Detection System (E31)	Objective 13.4
▪ The Leak Detection System monitors the valve stem leakoff from testable check valve, F005. - Leak detection is determined by a temperature element that measures the temperature in the piping connected to the air-operated check valve.	Figure 2
▪ The Leak Detection System is also used to monitor for breaks in the HPCS injection line (external to the reactor vessel shroud). - Instrumentation measures differential pressure between the HPCS injection line and the above core plate instrument tap. - HPCS Line Break status lamp is only valid above 80% power and 90% core flow. The HPCS System will still operate when the Leak Detection System is inoperable.	Figure 1
▪ ECCS differential pressure instrumentation is classified as providing indication only and is not required for system operability.	
2. Condensate and Refueling Water Storage and Transfer System (P11)	Objective 13.5
▪ The Condensate Storage Tank provides the normal source of water to the HPCS Pump suction. - Of the total CST capacity, approximately 170,000 gallons is reserved for use by HPCS and the Reactor Core Cooling System. The CST level transmitters for HPCS are tapped off of the CST suction line in lieu of a static sensing line. As a result of friction losses, the low CST suction swaps to the suppression pool are dependent on flow and occur higher than intended, resulting in less usable volume. Less usable volume is not an issue since the suppression pool is the normally credited water source.	Figure 2

Examination Outline Cross Reference	Level	RO
259001 Reactor Feedwater System K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM : K5.03 Turbine operation: TDRFP's-Only	Tier	2
	Group #	2
	K/A	259001 K5.03
	Rating	2.8
	Revision	0
Revision Statement:		

Question: 62

The plant is operating at rated thermal power when the Main Turbine trips.

The operator at the controls aligns for startup level control using Reactor Feed Pump ‘A’.

10 minutes after the Scram the transient is complete; all plant parameters are stable again.

Which of the following describes the Governor Control Valve position for RFPT ‘A’?

A. One-half the original indication due to the Scram.

B. Slightly lower than original indication due to the Scram.

C. Slightly higher than original indication due to more load on the feed pump turbine.

D. More than double the original indication due to a swap to its high pressure steam source.

Answer: **D**
Explanation: Rated position for the governor valve is 20% after a scram. The governor will slowly move to high pressure steam and governor valve will be 60%. ‘D’ is correct
Distracters: ‘A’ is wrong – Plausible, just the opposite from the actual . ‘B’ is wrong – Plausible, Due to the swap to high pressure steam that actually have less energy due to not being superheated the gov will have to be farther open.

‘C’ is wrong – Due to the swap to high pressure steam that actually have less energy due to not being superheated the gov will have to be farther open by a large amount not slightly.		
K/A Match This question requires the applicant to have the knowledge of the implications of the effects on Reactor feed pump turbines when the steam supply swaps after a plant shutdown.		
Technical References: GLP-OPS-N2100, Feedwater Lesson Plan. 04-1-01-N21-1, Feedwater SOI		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-N2100 Obj 17, 18		
Question Source:	**Bank # 761**	2013 NRC Exam Q62
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Simulator actions.

Examination Outline Cross Reference	Level	RO
272000 Radiation Monitoring System A3. Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including: A3.06 Ventilation system isolation indications	Tier	2
	Group #	2
	K/A	272000 A3.06
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 63

At time = 0 minutes: All 4 Control Room Vent Radiation Monitors are indicating 6 mR/hr.

At time = 5 minutes: A LOCA occurs.

At time = 10 minutes: Reactor water level reaches a low of -51" wide range before HPCS and RCIC begin to restore it.

Which of the following describes operation of the Fresh Air Inlet valves, Z51-F007 and F016, for the Control Room Standby Fresh Air Ventilation System?

Auto-isolated at time:

A. 10 minutes; can be manually re-opened at time = 20 min.

B. 0 minutes; can be manually re-opened at time = 10 minutes.

C. 0 minutes; are interlocked closed until all signals are manually reset.

D. 10 minutes; are interlocked closed until all signals are manually reset.

Answer: B
Explanation: On the receipt of a control room isolation signal, -41.6" Reactor water level or 1.23 psig Drywell pressure or >5mr on Control Room rad monitor, the SBFA unit will auto start in Recirc mode. The fresh air inlet valves F007 and F016 are interlocked closed for 10 min after isolation signal. A subsequent LOCA signal has no effect on the 10 min time delay. 'B' is correct
Distracters: 'A' is wrong. The isolation occurred at T-0

'C' is wrong. The fresh air valves can be opened 10 min after isolation signal.

'D' is wrong. The isolation occurred at T-0

K/A Match

This question requires the applicant to have the ability to monitor ventilation isolations from radiation signals.

Technical References:

GLP-OPS-Z5100, Control Room HVAC Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-Z5100 Obj 9, 11

Question Source:	**Bank # 251**	Not used on NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

The Standby Fresh Air Unit Fans automatically trip if a fire is detected in the filter train.

Fresh air operation should not be confused with the name fresh air unit.

Fresh air operation refers to drawing in fresh air into the system via the fresh air valves F007 or F016.

Fresh air is unfiltered air from the outside via Z77 HVAC penetrations to the control building.

There is only one fresh air MOV in each fresh air line.

The lines are not monitored for radiation.

♫ Monitoring – Ensure effective plant monitoring by operators.

Opening either F007 or F016 requires entry into LCO 3.0.3 when in Operational Conditions 1, 2, or 3 to ensure a single failure during a LOP/LOCA will not result in the unfiltered in leakage to the control room.

Figure 5

Each fresh air unit has two motor-operated inlet valves:

The recirculation valve (F008/F014) that diverts a portion of the Control Room Air Conditioner return air to the Fresh Air Unit Filter for purification.

The fresh air inlet valve (F007/F016) that allows outside air, from the safeguard switchgear and battery room ventilation intake, to be drawn in to be filtered by the filter train prior to routing it to the Control Room.

State the location from which the Control Room Standby Fresh Air Units can be operated. (6.4)

6.4 The two inlet valves (F008/F014 and F007/F016) are operated by the same three-position, FRESH AIR/AUTO/RECIR, spring return to AUTO hand switch, located on Control Room Ventilation Panel SH13-P855.

Red/green position indicating lights are provided for the valves on P855.

If the fresh air unit fan is running, placing the hand switch in FRESH AIR will open the fresh air inlet valve (F007/F016) and close the recirculation valve (F008/F014).

Placing the switch in RECIR will open the recirculation valve (F008/F014) and close the fresh air inlet valve (F007/F016).

Describe the actions that occur on a Control Room HVAC isolation. (8)

8 The recirculation valve (F008/F014) will automatically open on a Control Room isolation signal or if the associated fresh air unit fan is started.

The recirculation valve (F008/F014) will automatically close if its associated fresh air unit fan is stopped.

8 Fresh air inlet valve (F007/F016) automatically closes on a Control Room isolation signal or if the associated Fresh Air Unit Fan is stopped.

NOTES	LESSON BODY
	Motor operated dampers F008/F014, and F007/F016 fail as is on a loss of AC power.
The control room basically becomes a confined space and over time CO_2 will build up and oxygen concentration will drop.	[8] The Control Room isolation signal has a 10-minute time delay interlock.
	After the isolation signal automatically opens the recirculation valve (F008/F014) and closes the fresh air inlet valve (F007/F016), if open, the valves cannot be repositioned for 10 minutes.
	Prevents contaminated air from entering the Control Room.
	After the 10-minute time delay, the operator may operate the fresh air valves if air analysis proves it is safe to do so. RP personnel sample the control room atmosphere for oxygen and CO_2 concentrations if operating in the isolate mode for a prolonged period of time after an event.
	The fresh air unit motor-operated discharge damper (F005/F013) automatically opens and closes as its associated fan is started and stopped.
SOI P&L 3.9: Ref TS 3.7.3 if F007 & F016 not fully closed.	The damper does not have a hand switch but does have red/green position indicating lights on Control Room Ventilation Panel SH13-P855.
	An MOV Test Circuit is provided for the Control Room Ventilation System motor-operated valves (F007/F016 and F008/F014).
	A two-position key lock NORM/TEST (key removable in NORM) hand switch on Control Room Ventilation Panel SH13-P855 operates the test circuit and is maintained in NORM during plant operation.
	When the hand switch is in NORM, the motor protection devices are bypassed to allow the motor-operated valves to operate under emergency conditions (possible motor damage may result since protection devices are bypassed).
	When the hand switch is in TEST, the motor protection devices are not bypassed which allows the valves to be tested.
Annunciator P855-1A(2A)-C5	If either division MOV Test hand switch is in the TEST position, Control Room Ventilation Panel SH13-P855 annunciator CONT RM DIV 1(2) MOVS IN TEST MODE is actuated.
MCP 91/1052 abandoned the humidity controls on the standby fresh air units and requires compensatory measures.	The humidity control feature of the heater has been abandoned in place. Compensatory measures have been put in place in the Z51 SOI which include taking Control Room humidity readings when outside ambient temperature drops below 40ºF and placing the Z51 system in isolate mode while below 25% humidity.
	A humidity measuring device checked out from M&TE is normally maintained in the control room for this purpose during periods of low humidity.
	A manual water deluge system is provided for each filter train
Annunciator P855-1A(2A)-B1	On high temperature, a temperature sensor for each filter train actuates Control Room Panel P855 FRESH AIR UNIT A(B) CHAR TEMP HI annunciator.

Examination Outline Cross Reference	Level	RO
290002 Reactor Vessel Internals A2. Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 †Control rod drop accident	Tier	2
	Group #	2
	K/A	290002 A2.03
	Rating	3.6
	Revision	0
Revision Statement:		

Question: 64

Plant startup is in progress.

Control rods are being withdrawn to achieve critical condition.

One control rod reaches position 48, a coupling check is performed.

The following indication occurs.

P680-4A2-E5, CONT ROD OVERTRAVEL
P680 Status light "ROD UNCOUPLED" backlight"

(1) Which of the following describes the immediate impact on the plant?

(2) What action is performed by the crew?

A. (1) Inability to drive other control rods
(2) Perform a local single control rod scram.

B. (1) Inability to drive other control rods
(2) Move the control rod by normal insertion to full in position

C. (1) Possible control rod drop and fuel damage
(2) Perform a local single control rod scram.

D. (1) Possible control rod drop and fuel damage
(2) Move the control rod by normal insertion to full in position

Answer:	**D**
Explanation:	

The given indications are a control rod that has become uncoupled from the CRD mech. This could be the setup for a control rod drop accident. Procedures state to fully insert the control rod to attempt to re-couple.

'D' is correct

Distracters:

'A' is wrong. A stuck rod would be no movement of the CRD mech. Single rod scrams is not used due to damage to the CRD or Mech.

'B' is wrong. A stuck rod would be no movement of the CRD mech.

'C' is wrong. Single rod scrams is not used due to damage to the CRD or Mech.

K/A Match

This question requires the applicant to have the ability to predict the impact on a Control rod drop accident and have knowledge of the procedure to mitigate.

Technical References:

GLP-OPS-C111B, Control Rod Mech Lesson Plan.
P680-4A2-E5, CONT ROD OVERTRAVEL
05-1-02-IV-1, CRD Malfunction ONEP

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C111B Obj 9, 11

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Instructor Notes	Lesson Content

Nitriding is a method of increasing the surface hardness of a metal while still maintaining its ductile core.

The index tube, threaded to the coupling spud on the upper end and the drive piston on the lower end, is a long hollow shaft made of nitrided stainless steel with circumferential locking grooves every six inches for positioning of the control rod.

The upper surface of the collet finger engages the square shoulder to effectively lock the index tube to prevent inadvertent withdrawal.

Of the twenty-five notched positions on the index tube, twenty-four have square shoulders on the top for locking with the collet fingers and tapered surfaces on the bottom which cause the collet fingers to cam out on rod insertion.

The double tapered uppermost notched position does not provide a locking surface; it serves only as an indentation for the collet fingers to relax when the rod is fully withdrawn.

Describe the method of detecting an uncoupled control rod. (5.2)

CR Guide Tube is discussed in the Nuclear Boiler and Vessel Internals Lesson Plan.

When fully withdrawn, the control rod firmly backseats on the CR guide tube due to its weight.

With the control rod attached, the drive piston can not travel to the "overtravel" position.

5.2 If the coupling spud is uncoupled from the control rod, the index tube settles and the upper sloping surface of the double tapered uppermost notch cams the collet fingers outward.

The drive piston thus withdraws the small additional distance required to reach its lower mechanical end stop.

Annunciator P680-4A2-E5. Operators are required to verify the absence of this alarm anytime a control rod is fully withdrawn.

In this "overtravel" position, the attached ring magnet actuates the "overtravel" reed switch in the position indicator probe, thus providing an indication of control rod and drive separation.

Receive associated Control Room Panel P680 annunciator CONT ROD OVERTRAVEL.

Collet Locking Mechanism

State the purpose of the Collet Locking Mechanism. (3.4)

Figure 1 and Vendor Manual Figure 1-1. Collet housing located at the upper end of the outer tube.

3.4 Contained in the collet housing portion of the outer tube, the collet locking mechanism prevents the index tube from accidentally moving downward while allowing for incremental positioning of the index tube.

The collet locking mechanism consists of a guide cap, barrel, collet and piston assembly, and collet spring.

Instructor Notes	Lesson Content
Note that the numerical position indication is only available when the control rod is selected or the ALL RODS pushbutton is depressed, as discussed in the RC&IS Lesson Plan.	Switch 52, located opposite and slightly higher than switch 00, closes simultaneously with 00 to provide the “full-in” indication. “00” displayed and the associated green full-in indicator light illuminated on the Control Room Panel P680 full core display portion of the Rod Display Module. Switch 51, located immediately above switch 52, is closed at the extreme upper end of the control rod insertion to provide the “overtravel full-in” indication.
Same green light without numeric display.	Only the green full-in indicator light illuminated on the Control Room Panel P680 full core display portion of the Rod Display Module.
	Switch 49, located near the bottom of the probe at the same level as switch 48, closes simultaneously with switch 48 to provide the “full-out” indication. “48” displayed on the Control Room Panel P680 full core display portion of the Rod Display Module with the associated red full-out indicator light illuminated. Switch 50, located 2 inches below the normal full out position, is closed only when the rod and drive mechanism are uncoupled.
Annunciator P680-4A2-E5.	When closed, causes Control Room Panel P680 annunciator CONT ROD OVERTRAVEL to alarm.
The limit of drive piston down travel is provided by the backseat of the control rod on the guide tube.	With the control rod and drive mechanism coupled, the control rod rests on the guide tube seating surface, preventing the index tube/drive piston from withdrawing to the overtravel position which prevents the ring magnet from passing over and actuating the switch.
	For each CRD mechanism, a thermocouple, located in the top of the position indicating probe on the “A” Channel switch assembly, provides indication on multi-point temperature recorder C11-TJR-R018.
Annunciator P680-4A2-A4.	This recorder, located in Area 7, elevation 139′ of the Auxiliary Building, inputs to Control Room Panel P680 common annunciator CRD HYD TEMP HI, which actuates when any of the 193 drive mechanisms reaches a temperature of 400ºF.

04-1-02-1H13-P680-4A2-E5	
Revision: 100	Page 1 of 1

CONT ROD
OVERTRAVEL

L616

Non-Safety Related

Alarm Device 1C11-ZA-

1.0 POSSIBLE CAUSES

1.1 A control rod has become uncoupled and the CRD drive piston is at the overtravel out position (control rod position cannot be determined)

2.0 AUTOMATIC ACTION

2.1 None

3.0 IMMEDIATE OPERATOR ACTION

3.1 Determine the rod that has become uncoupled by depressing the ROD UNCOUPLED pushbutton and observe the red LEDs of the uncoupled rod on the rod display module.

3.2 Refer to ONEP 05-1-02-IV-1, Control Rod/Drive Malfunction.

4.0 SUBSEQUENT OPERATOR ACTION

4.1 Refer to Technical Specifications 3.1.3 AND 3.10.8.

3.6 Uncoupled Control Rod

CAUTION

Do NOT attempt to couple a Control Rod by scramming. This Could result in Control Rod OR CRD drive damage.

3.6.1 **OBTAIN** Reactor Engineering permission to **ATTEMPT RECOUPLING** Control Rod **AND WITHDRAWING** drive to full-out position.

3.6.2 **ENTER** Tech Spec 3.1.3 **AND ATTEMPT** to **RECOUPLE** affected Control Rod by **INSERTING** the drive.

3.6.3 **AFTER ATTEMPTING** to **RECOUPLE** drive, **THEN WITHDRAW** Control Rod to full-out position to **DETERMINE IF** drive is uncoupled.

3.6.4 **IF** drive **CANNOT** be recoupled, **THEN FULLY INSERT** the Control Rod **AND DISARM** per LCO 3.1.3 CONDITION C.

3.7 Control Rod Drop

3.7.1 **IF** a scram did **NOT** occur **AND IF** permitted by RPCS, **THEN FULLY INSERT** drive for affected Control Rod.

3.7.2 **MONITOR** radiation monitors for signs of fuel damage.

Examination Outline Cross Reference	Level	RO
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode 2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	Tier	2
	Group #	2
	K/A	219000 2.4.49
	Rating	4.6
	Revision	0
Revision Statement:		

Question: 65

RHR pump 'A' is operating in Suppression Pool cooling.

The following occurs:

- SUPP POOL LVL HI/LO alarms on P870
- RHR RM A SMP LVL HI-HI alarm on P680
- RHR A PMP RM FLOODED alarm on P870

Current Suppression Pool level is 18.2 ft.

Which of the following is the required action?

A. Isolate Suppression Pool Cooling only

B. Trip RHR A pump only
Enter EP-4 only

C. Trip RHR A pump only
Enter EP-3 only

D. Isolate Suppression Pool Cooling
Enter EP-4
Enter EP-3

Answer: D
Explanation: The given indications are indications of a suppression pool leak into the RHR A room. EP-3 is entered on low suppression pool level <18.34 ft., EP-4 is entered from RHR room sump level and RHR room flooded. 02-S-1-43, Transient Mitigation Strategy, step 6.4.1, "Leaks should be isolated **WITHOUT** informing SRO" 'D' is correct

Distracters:

'A' is wrong. EP-3 & 4 shall be entered.

'B' is wrong. EP-3 shall also be entered. RHR A system needs to be isolated to stop the leak

'C' is wrong. EP-4 shall also be entered. RHR A system needs to be isolated to stop the leak

K/A Match

This question requires the applicant to have the ability to perform actions without CRS direction.

Technical References:

GLP-OPS-EP3 & EP4, Lesson Plan.
02-S-01-43, Transient Mitigation Strategy, Rev 7

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3 & EP4, OBJ 3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

6.3 EP-3 Strategy

6.3.1 **IF** CTMT spray is required, **THEN** **BOTH** loops of CTMT spray should be initiated, **UNLESS** RHR pumps are required to maintain adequate core cooling.

a. It may be necessary to place one RHR loop in CTMT spray at a time to ensure remaining pump will maintain adequate core cooling.

6.3.2 **IF** Suppression Pool temperature and level are stable

AND

Suppression Pool level is NOT low,
THEN Suppression Pool Make Up (SPMU) can be overridden **PER** SOI 04-1-01-E30-1.

6.3.3 **IF** Suppression Pool level is lowering **OR** Suppression Pool temperature is unable to be controlled,
THEN **AVOID** overriding SPMU.

6.4 EP-4 Strategy

6.4.1 Leaks should be isolated **WITHOUT** informing SRO.

a. **IF** leak involves personnel **OR** equipment damage **AND** **IF** required due to a hazard,
THEN SRO should be informed as soon as possible.

6.4.2 In conditions where there is an un-isolable leak **AND** adequate Reactor water level inventory exists, it may be appropriate to **REDUCE** Reactor driving head to reduce leakage **AND** therefore secondary effects on plant. In this situation, reactor pressure bands may be reduced using guidance provided in Section 6.6.6 Band Control Strategies.

6.4.3 Confirmation that Main Steam Line radiation level is above 3 x Normal background is performed by observance of both:

- MSL A/D HiHi/INOP (1H13-P601-18A-C4)
- MSL B/C HiHi/INOP (1H13-P601-19A-C4) annunciators in ALARM

a. Additional confirmation can be obtained by observing upward trends in:

- Offgas Pre-Treatment radiation readings
- MSL radiation monitors

6.4.4 **IF** RCIC steam leak with failure to isolate occurs,
THEN RCIC will be shutdown and isolated unless required for Adequate Core Cooling.

Entry Conditions

SP temperature above **95°F**	**DW temperature** above **135°F**	**CTMT temperature** above **95°F**
DW pressure above **1.23 psig**	**SP level** below **18.34 ft**	**SP level** above **18.81 ft**

Discussion

Entry Conditions

The *Containment Control* entry conditions are symptomatic of an emergency or a condition that could become an emergency. The specified parameters are the same as the key parameters controlled by the procedure. The specified setpoints have been chosen to be operationally significant, unambiguous, readily identifiable, and familiar to plant operators. In general, they are also setpoints for scrams, alarms, trips, ECCS initiation, or Technical Specification limits. These action levels provide advance warning of potential emergency conditions, allowing action to be taken sufficiently early to prevent more severe consequences.

Containment Control must be entered at the beginning of the flowchart whenever any entry condition occurs, or clears and then occurs again, even if the procedure is already in use.

SP temperature

The suppression pool temperature entry condition corresponds to the most limiting Technical Specification LCO (95°F). Adverse effects of a high suppression pool temperature include reduced capacity for condensing steam, loss of ECCS pump NPSH, and pressurization of the containment. Events which may cause this entry condition include:

Loss of coolant

SRV actuation

DW temperature

The drywell temperature entry condition corresponds to the higher of the drywell temperature LCO and the maximum normal operating temperature. At GGNS, both of these values are 135°F.

A high drywell temperature is a symptom of events which may jeopardize primary containment integrity and the operability of equipment in the containment, including:

Loss of coolant

Loss of drywell cooling

SRV actuation

Entry Conditions

Any Operating Limit exceeded (Table SC-1)

Differential pressure above 0 in.

Offsite Rad Release declaration of **Alert** expected

Spent Fuel Pool Level below **207.49 ft.**

Spent Fuel Pool Temperature above **120°F**

Discussion

Entry Conditions

The *Auxiliary Building Control* entry conditions are symptomatic of an emergency or a condition that could become an emergency. Adverse effects on the operability of equipment located in the secondary containment and conditions directly challenging secondary containment integrity were specifically considered in the selection of these entry conditions. The specified parameters are the same as the key parameters controlled by the procedure or are closely related to them. The specified setpoints have been chosen to be operationally significant, unambiguous, readily identifiable, and familiar to operators.

High differential pressure is indicative of a potential loss of secondary containment structural integrity and could result in uncontrolled release of radioactivity to the environment. While differential pressure is used as an entry condition to EP-4, no separate “Differential Pressure” branch is included in the procedure. Since the margin between the normal differential and the structural building strength is small, the only practical control methods are those specified for controlling secondary containment temperature and radiation level:

Operating area coolers

Operating SGTS

Isolating breaks

High area temperatures are indications of fires and of breaks into the secondary containment. Either may jeopardize the habitability of the secondary containment and the operability of the equipment in it.

The high area radiation, high HVAC exhaust radiation, and high area water level entry conditions provide additional indications of breaks into the secondary containment. High radiation levels may limit access to the secondary containment and to the equipment in it. Water collecting in secondary containment spaces could flood equipment necessary for safe operation of the plant.

Spent fuel pool temperature above the high temperature alarm setpoint is indicative of a loss of spent fuel pool cooling. Continued heatup of the spent fuel pool may result in release of volatile fission products, increased secondary containment humidity, and eventual loss of spent fuel pool inventory due to boiling.

Table SC-1
Aux Building Area Parameters

Area	Operating Limit	Max Safe Value
TEMPERATURE		
MSL PIPE TUNNEL TEMP	185°F (P601-19A/18A-A3/A4)	250°F (E31-N604A,B,C,D,E,F)
RHR-A EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608A, N610A)
RHR-B EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608B, N610B)
RCIC EQUIP AREA TEMP	185°F (P601-21A-G3)	212°F (E31-N602A/B)
RWCU-PUMP ROOM 1 TEMP	170°F (P680-11A-A1)	NA
RWCU-PUMP ROOM 2 TEMP	170°F (P680-11A-A2)	NA
HVAC EXHAUST RADIATION LEVEL		
AUX BLDG FHA VENT EXHAUST	3.6 mr/hr (P601-19A-B9/C9)	NA
AUX BLDG FHA POOL EXHAUST	30 mr/hr (P601-19A-B10/C10)	NA
RADIATION LEVEL		
RHR ROOM A	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR ROOM B	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR HX A HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RHR HX B HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RCIC ROOM	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
MAIN STEAM LINE RAD MONITOR	Set Point Log (P601-19A-D4)	8×10^4 mr/hr
SGTS FLTR TRN	2.5 mr/hr (P844-1A-C5)	8×10^2 mr/hr
WATER LEVEL		
RHR RM A	89 ft 8 in. (P680-8A1-A2)	93 ft 6 in. (P870-2A-E1)
RHR RM B	89 ft 8 in. (P680-8A1-B2)	93 ft 6 in. (P870-10A-G1)
RHR RM C	90 ft 6 in. (P680-8A1-C2)	93 ft 6 in. (P870-10A-G2)
RCIC RM	90 ft 6 in. (P680-8A1-C4)	93 ft 6 in. (P870-2A-A1)
LPCS RM	90 ft 6 in. (P680-8A1-A4)	93 ft 6 in. (P870-2A-F1)
HPCS RM	90 ft 6 in. (P680-8A1-B4)	93 ft 6 in. (P870-5A-H1)

Examination Outline Cross Reference	Level	RO
2.1.8 Ability to coordinate personnel activities outside the control room.	Tier	3
	Group #	0
	K/A	2.1.8
	Rating	3.4
	Revision	0

Revision Statement:

New Question changed to Attachment 21 to avoid confusion.

Question: 66

You have been directed by the CRS to have the safe shutdown operator install the following Emergency Procedure attachment:

- Att. 21 – Deenergizing Scram Solenoids

Where will you send the operator to perform this attachment?

A. Main control room back panels and upper control room

B. Turbine Building area 3, elevation 113 ft and area 4, elevation 166 ft.

C. ARI/RPT test panel in area 8, elevation 139 ft in the Auxiliary Building

D. Reactor Protection System motor generator rooms in the Control Building

Answer: D
Explanation: EP Att. 21 is performed by opening two breakers on each of the RPS distribution panels. The Distribution panels are located in the respective RPS Motor/Generator rooms. 'D' is correct
Distracters: 'A' is wrong but is plausible since RPS trips are defeated by EP Attachment 19 by installing relay jumpers in the listed panels on elevations 166 ft and 189 ft of the Control Building, and RPS is related in function to ARI. 'B' is wrong but is plausible since this would be Att. 18 and it also opens breakers to achieve the task. 'C' is wrong but is plausible since the ARI/RPT test panel could be used to manipulate ARI/RPT system, but the panel is not used for Attachment 18.

K/A Match

This question requires the applicant to coordinate the crew during an emergency outside the control room.

Technical References:

05-S-01-EP-1 Att. 18
05-S-01-EP-1 Att. 19
05-S-01-EP-1 Att. 21

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP01 Obj. 10

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

1. DEENERGIZING SCRAM SOLENOIDS

1.0 PURPOSE

1.1 To provide instructions for de-energizing all scram solenoids during an ATWS.

2.0 INSTRUCTIONS

		INITIAL / DATE
2.1	**OPEN** breaker 1C71-CB2A (52-1C71102) at panel 1C71-P001 (Area 25A, EL 189').	/
2.2	**OPEN** breaker 1C71-CB8A (52-1C71108) at panel 1C71-P001 (Area 25A, EL 189').	/
2.3	**OPEN** breaker 1C71-CB2B (52-1C71202) at panel 1C71-P002 (Area 25A, EL 148').	/
2.4	**OPEN** breaker 1C71-CB8B (52-1C71208) at panel 1C71-P002 (Area 25A, EL 148').	/
2.5	**WHEN** control rod motion Stops, **THEN AUTHORIZE** to Re-Close breakers 1C71-CB2A, 1C71-CB8A, 1C71-CB2B, **AND** 1C71-CB8B.	/ Shift Supv

Examination Outline Cross Reference	Level	RO
2.1.19 Ability to use plant computers to evaluate system or component status.	Tier	3
	Group #	0
	K/A	2.1.19
	Rating	3.9
	Revision	0
Revision Statement:		

Question: 67

Which of the following computer systems can be used to determine current system / component status and trend parameters?

A. PDS

B. EOOS

C. CYCLOPS

D. INFORM

Answer: **A**
Explanation: PDS, Plant Display System, is used for up to date information and current information on system and components. 'A' is correct
Distracters: 'B' is wrong. EOOS is Equipment Out Of Service, even though it gives information on equipment that is Out of service, the information is manually input and does not supply up to date information. 'C' is wrong. CYCLOPS is used by the STA/RE to determine core information, not system status. 'D' is wrong. INFORM is used to inform the Emergency Response Organization for plant emergency status, no system or component status.
K/A Match This question requires the applicant to have the knowledge of and use of computer systems that provide Component / System status.

Technical References: 04-1-01-C91-1, Plant Display System (PDS), SOI, Rev. 21		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GSMS-RO-IN002, Obj. 8		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

5.2 NORMAL OPERATIONS – System Operation

5.2.1 Prerequisites

1. Inverters in service per Procedure 04-1-01-L62-1, Static Inverters System.
2. PDS is operational.

5.2.2 Instructions

> **NOTE**
>
> Some PDS machines may not have all listed functions.

1. **PERFORM** following functions to view plant data:

    - Process Diagram – Set of interactive P&ID displays which provide dynamic components and values based on most recent plant data.
    - Operator Guide – Display both static and dynamic data for a previously defined set of computer points.
    - Realtime Trend – Flexible means to trend digital or analog data with ability to control ranges, alarm limits, output format, and time base parameters.
    - Historical Trend – Review a history of digital or analog data.
    - Point Detail – Review a point from a list to display point ID, description, current value, current state, and quality.
    - Point Mode – Allows request to substitute value, out of service, maintenance mode, or normal processing mode change on a point selected.
    - Point Data List – Means to create or modify a list of points to be used in a Trend or an Operator Guide.
    - Status Log – Predetermined set of information to view including Plant Transient Log and Sequence of Events Log.
    - Special Function – Allows selection of Quality Display status or a Reactor Heatup and Cooldown Trend.
    - ONEPs – Predefined process data which can be selected for use during entry into an Off Normal Event Procedure.
    - ERFIS – Selection of Operations Guides containing information on points potentially needed during an emergency event.
    - Exit – PDS logout capability.

NOTE

FEEDS server is repository to store high resolution data reports for post trip reviews, scram time testing and turbine valve testing.

2. **IF** data from FEEDS server is required to be acquired,
THEN CONTACT Computer Services Group to assist.

3. **MONITOR** plant parameters as follows:

   a. **USE** Process Diagrams.

   b. **USE** Operator Guides.

   c. **USE** Realtime Trends.

   d. **USE** Historical Trends.

   e. **USE** Point Details.

   f. **USE** Point Mode.

   g. **USE** Point Data List.

   h. **USE** Status Log.

   i. **USE** Special Functions.

   j. **IF** ONEP entry required,
   THEN USE ONEPs function.

   k. **IF** emergency event,
   THEN USE ERFIS function.

   l. **IF** required to exit PDS,
   THEN USE EXIT.

Examination Outline Cross Reference	Level	RO
2.2.40 Ability to apply Technical Specifications for a system.	Tier	3
	Group #	0
	K/A	2.2.40
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 68

Which of the following requires IMMEDIATE entry into TS 3.4.1 Recirculation Loops Operating?

A. TOT JP FLO 1B33-UR-R613 is 64 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 24.9 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 28.9 KGPM

B. TOT JP FLO 1B33-UR-R613 is 74.8 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 26.4 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 30.5 KGPM

C. TOT JP FLO 1B33-UR-R613 is 80.5 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 29.4 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 32.8 KGPM

D. TOT JP FLO 1B33-UR-R613 is 104 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 42.5 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 40.8 KGPM

Answer: **C**

Explanation:

Tech Specs 3.4.1 states “Verify recirculation loop jet pump flow mismatch with both recirculation pumps in operation is:

< = 10% of rated core flow when operating at <70% core flow
< = 5% of rated core flow when operating at > = 70% core flow

70% core flow = 112.5 * 0.7 = 78.75 mlbm/hr - 5% = 2.2 KGPM, 10% = 4.4 KGPM

‘C’ is >70% core flow at 80.5 Mlbm/hr and mismatch is 3.4 Kgpm which is >2.2 Kgpm.

‘C’ is correct

Distracters:

'A' is wrong. The indication is <70% core flow (78.75 Mlbm/hr) so flows must be within 4400 gpm. Actual mismatch is 4000gpm.

'B' is wrong. The indication is <70% core flow (78.75 Mlbm/hr) so flows must be within 4400 gpm. Actual mismatch is 4100gpm..

'D' is wrong. The indication is >70% core flow (78.75 Mlbm/hr) so flows must be within 2200 gpm. Actual mismatch is 1700gpm..

K/A Match

This question requires the applicant to have the knowledge of entry into Tech Specs.

Technical References:

Tech Specs 3.4.1

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-B3300 Objective: 44

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop shall be in operation provided the plant is not operating in the MELLLA+ domain defined in the COLR and provided the required limits are modified for single loop operation as specified in the COLR.

---------------------------NOTE---------------------------
Required limit modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation loop jet pump flow mismatch not within limits.	A.1 Shutdown one recirculation loop.	2 hours

(continued)

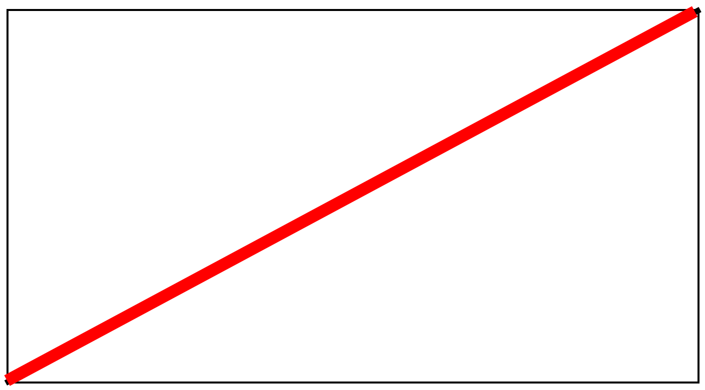
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. --	
	Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. $\leq$ 10% of rated core flow when operating at < 70% of rated core flow; and b. $\leq$ 5% of rated core flow when operating at $\geq$ 70% of rated core flow.	24 hours

Examination Outline Cross Reference	**Level**	**RO**
2.2.43 Knowledge of the process used to track inoperable alarms.	**Tier**	3
	Group #	0
	K/A	2.2.43
	Rating	3.0
	Revision	0
Revision Statement:		

Question: 69

A Control Room alarm window has a single red tape diagonally across the window.



Which of the following describes the status of the control room annunciator represented in the figure?

A. OPERABLE ANNUNCIATOR – Multiple inputs, one input is bypassed

B. PROBLEM ANNUNCIATOR – Faulty Alarm Circuitry, alarm card removed

C. PROBLEM ANNUNCIATOR – Faulty Alarm Circuitry, alarm card still installed

D. OPERABLE ANNUNCIATOR – Multiple inputs, more than one input is bypassed

Answer: C
Explanation: With the alarm card still installed, only a single strip of tape (a diagonal) is used (see section 6.2.1.d). (as indicated the above figure). **IDENTIFY** the alarm window that is **NOT** functioning properly by **PLACING** a length of red tape diagonally across the window, except for alarms bypassed per section 6.3, as shown on Attachment IV. ‘C’ is correct

Distracters:

'B' is incorrect because Per 02-S-01-25, section 6.2 and Attachment IV. Specifically, section 6.2.1.g…when the Problem Annunciator has its alarm card removed, a second strip of red tape is placed on the window, to form an X

'A' & 'D' are incorrect because for an OPERABLE ANNUNCIATOR, with any number of multiple inputs bypassed, two vertical strips of tape are used (see section 6.3.5.a and Attachment IV).

K/A Match

This question requires the applicant to have the knowledge of indications of INOP alarms.

Technical References:

02-S-01-25

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC

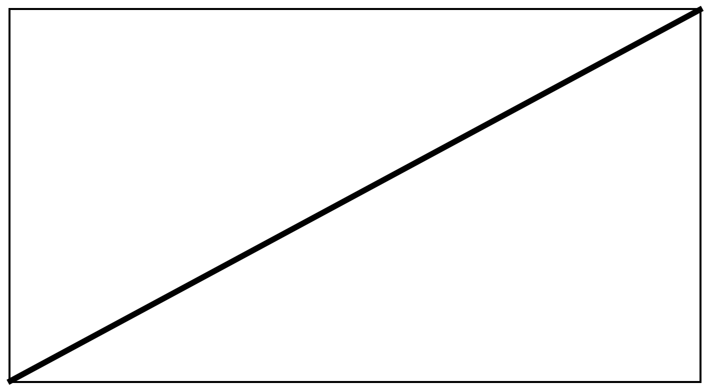
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

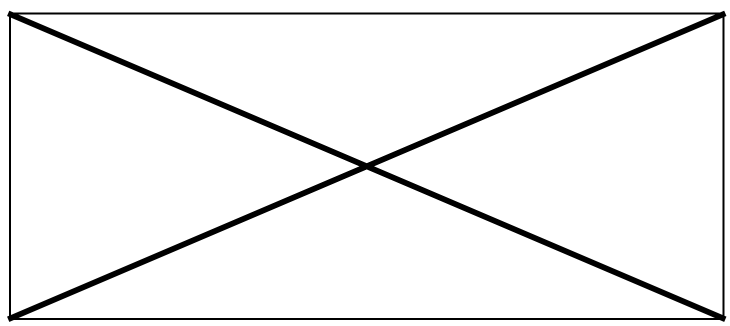
None.

DEFICIENT EQUIPMENT IDENTIFICATION

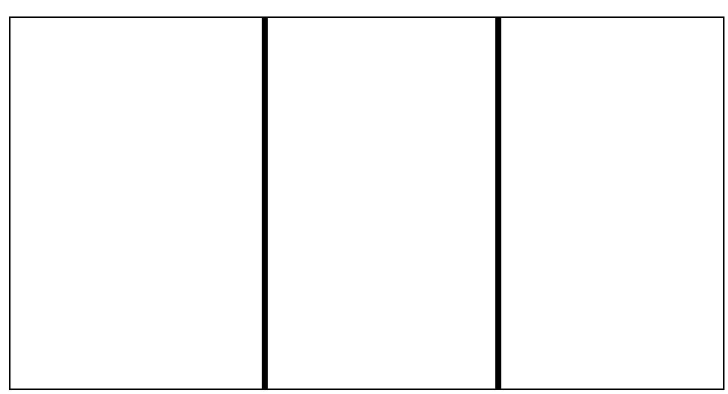
PROBLEM ANNUNCIATOR IDENTIFICATION



PROBLEM ANNUNCIATOR



PROBLEM ANNUNCIATOR WITH A PULLED CARD



OPERABLE ANNUNCIATOR WITH PROBLEM INPUT(S) BYPASSED

Examination Outline Cross Reference	Level	RO
2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	Tier	3
	Group #	0
	K/A	2.2.44
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 70

The following alarm was received.

Per 04-1-01-C82-1, Annunciator System:

(1) Which of the following describes the meaning of the color?

(2) What actions should be taken?

A. (1) Conditions which require Immediate action.
(2) Refer to ARI to determine which input is in alarm due to multiple inputs

B. (1) Conditions which require Immediate action.
(2) Verify automatic actions have occurred

C. (1) Conditions which require corrective action in a timely manner.
(2) Verify automatic actions have occurred

D. (1) Conditions which require corrective action in a timely manner.
(2) Refer to ARI to determine which input is in alarm due to multiple inputs

Answer:	**B**
Explanation:	

Per 04-1-01-C82-1, Annunciator System , section 4.1

Amber alarms (warning alarms) apply to conditions which require Immediate action

Alarms which have the lower right hand corner blacked out are associated with an automatic isolated condition. **REFER** to the" Automatic Isolated" ONEP 5-1-02-III-5 for initiating conditions and affected components

'B' is correct

Distracters:

'A' is wrong - Per 04-1-01-C82-1, Annunciator System Alarms which have a black square box in the lower right hand corner of the annunciator window have reflash capability and reannunciate upon other conditions which initiate the alarm.

'C' is wrong - Per 04-1-01-C82-1, Annunciator System White alarms (caution alarms) apply to conditions which require corrective action in a timely manner, this is an Amber alarm

'D' is wrong - Per 04-1-01-C82-1, Annunciator System White alarms (caution alarms) apply to conditions which require corrective action in a timely manner, this is an Amber alarm and Annunciator System Alarms which have a black square box in the lower right hand corner of the annunciator window have reflash capability and reannunciate upon other conditions which initiate the alarm.

K/A Match

This question requires the applicant to have the ability to interpret control room alarm colors and designations to determine operator actions and/or procedure entry.

Technical References:

04-1-01-C82-1, Annunciator System SOI, Rev 29

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		

None.

4.0 <u>NORMAL OPERATIONS</u>

4.1 <u>System Startup</u>

4.1.1 <u>Prerequisites</u>

a. Attachment III of this instruction completed.

4.1.2 <u>Instructions</u>

a. With the prerequisites completed, the system is started up.

4.2 <u>Response to Annunciation</u>

4.2.1 <u>Prerequisites</u>

a. The system has been started up per Section 4.1 of this instruction.

4.2.2 <u>Instructions</u>

<u>NOTE</u>

Section 4.2.2 is written addressing Control Room annunciators only The differences between local panels and Control Room panels are <u>**NOT**</u> discussed.

a. <u>**WHEN**</u> an annunciator alarms with an audible alarm, the operator <u>Should</u> **OBSERVE** a fast flashing annunciator window.

(1) Alarms are color coded as follows:

(a) Red alarms (critical alarms) apply to conditions which indicate the plant is in an unsafe condition <u>**OR**</u> direct scram.

(b) Amber alarms (warning alarms) apply to conditions which require <u>Immediate</u> action.

(c) White alarms (caution alarms) apply to conditions which require corrective action in a timely manner.

(d) Blue alarms (general alarms) apply to alarms which provide the operator with system status <u>**AND/OR**</u> information.

(e) Glowing cerise - a color bordering the annunciators encompassing those alarms associated with EP-4 entry conditions. This provides Operators with a quick reminder to **REFER** to EP-4 for applicable entry conditions.

4.2.2 (Cont.)

b. The operator, observing the window and scanning for any simultaneous alarms pushes the SILENCE pushbutton. He Should **NOT** see any change in the speed of the flashing annunciator. The audible alarm Should stop.

c. After reviewing all the annunciation that is in, the operator acknowledges the annunciation by **PUSHING** the ACKNOWLEDGE pushbutton.

IF the alarm has cleared, the annunciator window flashes at a slower rate.

IF the alarm is still in an alarm condition, the annunciator window shows a solid light.

Alarms which have a black square box in the lower right hand corner of the annunciator window have reflash capability and reannunciate upon other conditions which initiate the alarm.

Alarms which have the lower right hand corner blacked out are associated with an automatic isolated condition. **REFER** to the" Automatic Isolated" ONEP 5-1-02-III-5 for initiating conditions and affected components.

Alarms which have a '$' in the bottom left corner are Tech Spec/TRM required alarms. **REFER** to "Deficient Equipment Identification" 02-S-01-25 for cross-reference of alarms and associated TS/TRM.

d. **WHEN** the alarm condition clears, the annunciator window begins to flash slowly.

The operator Should **REVIEW** all annunciation before proceeding with any other steps.

Care Should be taken by the operator to ensure he knows which annunciation light goes out when he resets the annunciator.

WHEN he is assured of annunciation status, the operator Should **PUSH** the RESET pushbutton **AND** **OBSERVE** the annunciator window light go out.

Examination Outline Cross Reference	Level	RO
2.3.11 Ability to control radiation releases.	Tier	3
	Group #	0
	K/A	2.3.11
	Rating	3.8
	Revision	0
Revision Statement:		

Question: 71

Consider the following processes:

1. Batch release of liquid effluents during normal plant operations
2. Continuous discharge of liquid
3. Discharge of solid radioactive waste
4. Continuous discharge of gaseous effluents during normal plant operations

Which of the above processes are controlled using a Discharge Permit per 01-S-08-11, Radioactive Discharge Controls?

A. 1, only

B. 4, only

C. 1 and 2, only

D. 2 and 3, only

Answer: A
Explanation: 'A' is correct 01-S-08-11, Radioactive Discharge Controls, Rev 116 Section 5.4.3 - Batch Liquid Discharge Permit similar to Attachment 1, BATCH LIQUID RADWASTE DISCHARGE PERMIT should be completed for all batch liquid releases.
Distracters: 'B' & 'C' are wrong because, per Section 5.3, continuous discharges do not require a permit. 'D' is wrong per Section 6.2 which states that solid radwaste must not be discharged but rather handled Reference 3.5 (EN-RP-11, Radioactive Material Control).

K/A Match

This question requires the applicant to have the ability to interpret the different types of radioactive releases and ability to control the process.

Technical References:

01-S-08-11, Radioactive Discharge Controls, Rev 116

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC Obj.51

Question Source:	**Bank # 195**	2008 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(13)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

5.0 INSTRUCTIONS

5.1 General

1. All releases of radioactive materials shall be monitored to: **[O-1] [O-9] [O-23]**

   Demonstrate compliance with TRM/ODCM Specifications, appropriate effluent concentration based on 10CFR20, Appendix B, Table 2 and 10CFR50, Appendix I.

   Allow evaluation of performance of containment, waste treatment and effluent controls.

   Permit evaluation of environmental impact and estimation of potential annual radiation doses to public.

2. Minimum dilution factor for all liquid releases will be calculated in accordance with the ODCM.

3. All releases should be held to As Low As Reasonably Achievable criteria of 10CFR50, Appendix I. **[O-23]**

5.2 Solid Waste Discharges

1. Solid waste must not be discharged.

   Handling of solid waste will be performed as directed in Procedure EN-RP-121, Radioactive Material Control.

5.3 Continuous Discharges

1. Continuous discharges do not require discharge permit.

   Data for annual report will be taken from periodic sampling and flow monitors.

NOTE
As ODCM Specifications limits are approached, sampling frequency will be increased.

2. Sample of gaseous effluents will be taken at least monthly for isotopic analysis.

3. Sample for particulate and iodine concentrations will be taken at least weekly.

4. Dose from continuous gaseous discharges will be calculated in accordance with requirements of ODCM.

5.1 Batch Releases [O-11]

5. All planned batch releases of radioactive material shall be analyzed for radioactivity before discharge to determine identity and quantity of specific radionuclides.

5.4.2 Batch Gaseous Releases

1. **IF** condition exists in which Chemistry Manager determines a batch of gaseous release would have radiological impact on environment,
   THEN batch gaseous Release Permit WILL BE devised similar to liquid release permit.

5.**4.3** Batch Liquid Releases

1. Batch Liquid Discharge Permit similar to Attachment 1, BATCH LIQUID RADWASTE DISCHARGE PERMIT should be completed for all batch liquid releases.

NOTE

Steps 5.4.3.2 through 5.4.3.7 describe content and requirements of Attachment 1, BATCH LIQUID RADWASTE DISCHARGE PERMIT.

All spaces on Batch Liquid Radwaste Discharge Permit not used (e.g., provisions for inoperative flow or radiation process monitoring instrumentation and associated operator/technician signatures or initials) may be marked as N/A (not applicable) if not otherwise filled in.

2. Part 1, Discharge Request:

   a. Completed and signed by Operator.

   b. Verified by second Operator.

   c. Reviewed and signed by Operations Supervision.

   d. Will include:

      1) Release Number

      2) The tank to be released

      3) The volume of the tank

         FDST A/B Volume (gallons) = Ft. in Tank * 1904 gal/ft.

         EDST A/B Volume (gallons) = Ft. in Tank * 2592 gal/ft.

      4) Date and Time tank recirculation started

      5) Dilution Flow from (in order of preference):

         a) Circulating water blowdown flow (N71-R603)

         b) Plant service water, PSW, flow (Discharge Canal Flow)

         c) Other dilution water source flow.

Examination Outline Cross Reference	Level	RO
2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	Tier	3
	Group #	0
	K/A	2.3.14
	Rating	3.4
	Revision	0
Revision Statement: **NEW QUESTION**		

Question: 72

Which of the following room radiation levels must be posted as a High Radiation Area per 10CFR?

A. >5 mrem/hr

B. >10 mrem/hr

C. >100 mrem/hr

D. >500 rem/hr

Answer: C

Explanation:

Per 10CFR20.1003 Definitions

Radiation area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

High radiation area means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surfacethat the radiation penetrates.

Very high radiation area means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in 1 hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates.

These same definitions are located in EN-RP-100.

'C' is correct

Distracters:

'A' is wrong – This only meets the criteria for a Radiation area.

‘B’ is wrong - This only meets the criteria for a Radiation area, up to 100 mrem/hr

‘D’ is wrong – This meets the criteria for a Very High Radiation area.

K/A Match

This question requires the applicant to have the knowledge of possible radiation hazards during an emergency and results from actions.

Technical References:

10CFR20 1003

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-CFR01

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(13)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

3.0 continued

[4] **Declared Pregnant Woman** - A woman who has voluntarily informed the licensee, in writing, of her pregnancy and the estimated date of conception. The declaration remains in effect until the declared pregnant woman withdraws the declaration in writing or is no longer pregnant. [10CFR20]

[5] **Self-Reading Dosimeter (SRD)** – A self-reading quartz fiber, electronic, or other type of radiation measuring device used to measure exposures to x-ray or gamma radiation which can be read directly by the individual.

[6] **Dosimeter of Legal Record (DLR)** – A device used to determine an individual's accumulated external occupational radiation exposure including DDE, LDE and SDE. This device is inclusive of, but not limited to, OSLDs (optically stimulated luminescent dosimeters) and TLDs (thermoluminescent dosimeters).

[7] **High Contamination Area (HCA)** – An area where the majority of the area has removable surface contamination equal to or greater than 100,000 dpm/100cm2 beta-gamma, or equal to or greater than 500 dpm/100 cm2 alpha.

[8] **High Radiation Area** - an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates. [10CFR20]

[9] **Hot Spot** - A localized area such as valves, bends in pipe, drains, etc. which have radiation levels:

- Greater than or equal to 100 mRem/hr on contact, **AND**
- The contact radiation levels are greater than five times the 30 cm (≈ 12 inches) reading.

[10] **Locked High Radiation Area** - An area, accessible to individuals, in which radiation levels from sources external to the body could result in an individual receiving a deep dose equivalent greater than or equal to 1 Rem (10 mSv) in 1 hour at 30 cm (≈ 12 inches) from the radiation source or from any surface that the radiation penetrates.

[11] **Radiation Area** - An area, accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent greater than or equal to 5 mRem (0.05 mSv) in one hour at 30 cm (~ 12 inches) from the radiation source or from any surface that the radiation penetrates. [10CFR20

3.0 continued

[12] **Radiation Worker Self-Briefing** – A process whereby radiation workers can brief themselves on work radiological conditions without having to interface directly with Radiation Protection personnel. This process places part of the responsibility for radiological safety on the worker and their direct supervisor. [INPO 05-008, Rev. 2]

[13] **Radioactive Material Area** - An area in which licensed radioactive material in an amount exceeding 10 times the quantity specified in Appendix C, 10CFR20, is used or stored. This does not apply to radioactive materials contained within process equipment or materials in transport and packaged and labeled in accordance with appropriate regulations. (The terms Radioactive Material, Radioactive Materials, and Radioactive Materials Area are synonymous at Fleet facilities).

[14] **Radiologically Controlled Area (RCA)** - An area where full radiological controls (such as, contamination monitoring and controlled access/egress) are in effect for the purpose of providing protection and/or information to the individual. At Fleet Nuclear facilities, the main RCA normally includes parts or all of the Auxiliary, Fuel, Radwaste, Reactor Buildings and the Turbine Buildings at BWRs.

[15] **Self-Reading Dosimeter (SRD)** - This is equivalent to a direct reading dosimeter (DRD).

[16] **Very High Radiation Area** - An area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess 500 rads (5 grays) in 1 hour at 1 meter (≈ 3 feet) from the radiation source or 1 meter from any surface that the radiation penetrates. [10CFR20]

[17] **Whole Body Contamination Monitor (WBCM)** – Eberline PCM or equivalent automated personnel contamination monitor.

RESPONSIBILITIES

[1] **All Individuals:**

- Comply with all requirements of this procedure.
- Cooperate fully with RP personnel in all matters pertaining to radiological safety.
- Know personal dose margin.
- Know electronic dosimeter alarm setpoints.

]

Examination Outline Cross Reference	Level	RO
2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	Tier	3
	Group #	0
	K/A	2.4.2
	Rating	4.5
	Revision	0
Revision Statement:		

Question: 73

Which of the following coincides with an EOP entry condition?

A. DRWL PRESS HI/LO alarm on P870

B. Any Suppression pool temp high alarms on P870

C. HPCS system automatic suction swap on Suppression Pool Level

D. Initiation of Reactor Recirculation Pump cavitation interlock on Reactor water level.

Answer: D

Explanation:

Per EP-2 entry conditions a RPV water level of 11.4”

The Reactor Recirculation Pump Cavitation interlock is 11.4”

‘D’ is correct

Distracters:

‘A’ is wrong – P870-9A-D4, DRWL PRESS HI/LO, states the high pressure setpoint 1 psig. Entry condition for EP-2 or EP-3 is 1.23 psig drywell pressure

‘B’ is wrong – Two high temp alarms for suppression pool temp setpoint is 85.5°F, while the other two is set at 115°F. 85.5 is less than the entry condition for EP-3 of 95°F

‘C’ is wrong – P601-16A-C5, SUPP POOL LVL HI, corresponds with the HPCS suction swap, states Suppession pool level of 18’9” or 18.75 feet. EP-3 entry condition for High Suppression pool level is 18.81 feet.

K/A Match

This question requires the applicant to have the knowledge of possible radiation hazards during an emergency and results from actions.

Technical References: GLP-OPS-B3300, Reactor Recirc System lesson plan, Rev. 30 02-S-01-40, EP Technical Bases, Rev. 9 04-1-02-1H13-P870-9A-D4, Rev 130 04-1-02-1H13-P870-9A-E4, Rev 105		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-EPTS26 Obj. 40		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability: None.		

Lesson Content	Instructor Notes

momentarily, several times if needed to free the FCV.

If needed the recirculation loop flow controller output is raised slightly resulting in a slight servo error, and then resetting FCV motion inhibit. This results in a high pressure fluid pulse (or hammer) to the FCV ram effectively bumping it open further reducing FCV dp until it can be positioned with normal HPU pressure.

System Engineering guidance says not to hammer the FCV when HPU pressure is above 2250 psig.

Normal pressure is 1850 to 1950 psig as set by adjustment of Discharge Relief Valve F151A1, or F151A2 on HPU A or F151B1 or F151B2 on HPU B, depending on which subloops are running. After adjustment, a lock nut is tightened to maintain a setting.

Cavitation Interlocks

25.1, 25.2, 25.3 Cavitation interlocks prevent fast speed pump operation without adequate NPSH by automatically shifting the pumps to slow speed operation upon receiving any of the following signals:

EO-25.1; EO-25.2; EO-25.3

Figure 20B

Referred to as Recirc Pump Downshift or Transfer to Slow Speed.

Saturation temperature is calculated from steam dome pressure.

- < 7.4°F differential temperature, for 15 seconds, between the reactor saturation temperature and the pump suction lines.
- < 3.0 mlbm/hr total feedwater flow for 15 seconds.
- < 11.4 inches reactor water level.
- Low differential temperature and low feedwater flow are indicative of inadequate subcooling that could allow cavitation in the jet pumps, FCV, or the Recirculation Pumps.

For all downshifts, a trip in either division will actuate both trip systems.

Annunciator RECIRC PUMP A/B AUTO TRIP XFER TO LO SP on Control Room Panel P680 actuates to alert the operator that the pumps have transferred to slow speed.

Annunciators P680-3A-D4/D10. In addition, annunciators could have been initiated by an EOC-RPT trip, discussed later.

- If the transfer was due to low differential temperature or low feedwater flow, white lights above the RECIRC PMP A/B CAV INTLK RESET pushbuttons will illuminate.
- If the transfer was due to low reactor water level, white lights above the RX WTR LVL LO INTLK A/B RESET pushbuttons will illuminate.

Entry Conditions

RPV level
below **11.4 in.**

RPV pressure
above **1064.7 psig**

DW pressure
above **1.23 psig**

Reactor scram required
and any control rod withdrawn past **position 02** or unknown

Discussion

EP-2 must be entered at the beginning of the flowchart whenever any entry condition occurs, or clears and then reoccurs, even if the procedure is already in use. The entry conditions are symptomatic of an emergency affecting the RPV or a condition that could become an emergency. The specified parameters are the same as the key parameters controlled by the procedure or are closely related to them. The specified setpoints have been chosen to be operationally significant, unambiguous, readily identifiable, and familiar to operators. In general, the entry conditions are also setpoints for scrams, alarms, trips, and ECCS initiation. While each of the entry conditions requires a scram, a scram by itself is *not* an entry condition for the procedure.

RPV level below 11.4 in.

The RPV water level entry condition corresponds to the low RPV water level scram setpoint (11.4 in.). A low RPV water level is a symptom of events which may jeopardize adequate core cooling, including:

Loss of coolant

Loss of feedwater

RPV pressure above 1064.7 psig

The RPV pressure entry condition corresponds to the high RPV pressure scram setpoint (1064.7 psig). A high RPV pressure is a symptom of events which may jeopardize RPV integrity (and thereby adequate core cooling) such as:

SRV failure

Turbine trip with bypass failure

MSIV closure

ATWS

Drywell pressure above 1.23 psig

The drywell pressure entry condition corresponds to the high drywell pressure scram setpoint (1.23 psig). Although drywell pressure is not directly controlled by EP-2, high drywell pressure is a symptom of a break in the drywell, and thus of events that may challenge adequate core cooling.

Entry Conditions

SP temperature above **95°F**	**DW temperature** above **135°F**	**CTMT temperature** above **95°F**
DW pressure above **1.23 psig**	**SP level** below **18.34 ft**	**SP level** above **18.81 ft**

Discussion

Entry Conditions

The *Containment Control* entry conditions are symptomatic of an emergency or a condition that could become an emergency. The specified parameters are the same as the key parameters controlled by the procedure. The specified setpoints have been chosen to be operationally significant, unambiguous, readily identifiable, and familiar to plant operators. In general, they are also setpoints for scrams, alarms, trips, ECCS initiation, or Technical Specification limits. These action levels provide advance warning of potential emergency conditions, allowing action to be taken sufficiently early to prevent more severe consequences.

Containment Control must be entered at the beginning of the flowchart whenever any entry condition occurs, or clears and then occurs again, even if the procedure is already in use.

SP temperature

The suppression pool temperature entry condition corresponds to the most limiting Technical Specification LCO (95°F). Adverse effects of a high suppression pool temperature include reduced capacity for condensing steam, loss of ECCS pump NPSH, and pressurization of the containment. Events which may cause this entry condition include:

Loss of coolant

SRV actuation

DW temperature

The drywell temperature entry condition corresponds to the higher of the drywell temperature LCO and the maximum normal operating temperature. At GGNS, both of these values are 135°F.

A high drywell temperature is a symptom of events which may jeopardize primary containment integrity and the operability of equipment in the containment, including:

Loss of coolant

Loss of drywell cooling

SRV actuation

DRWL PRESS HI/LO

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Safety Related

Alarm Device 1M71-PDA-L601B

1.0 POSSIBLE CAUSES

1.1 High pressure due to leakage in the drywell.

1.2 High pressure due to loss of drywell cooling.

1.3 Low pressure due to excessive cooling in drywell.

1.4 Loss of coolant accident.

1.5 Air rupture in the drywell.

1.6 High pressure setpoint 1 psig above Ctmt pressure, low pressure setpoint 2.1 psig below Ctmt pressure.

2.0 AUTOMATIC ACTION

2.1 None

3.0 IMMEDIATE OPERATOR ACTION

3.1 Check recorder 1M71-PDR-R601B to determine whether high or low.

3.2 Check the air header pressures, instrument air on 1H13-P870 and service air on 1H13-P854.

3.3 Check drywell cooling is operating in accordance with SOI 04-1-01-M51-1.

3.4 The drywell may need venting due to heat up. Vent the drywell to the containment cooling ventilation system using the following steps:

3.4.1 Ensure that the containment cooling system is operating normally with containment cooling fans running and at least one containment cooling system charcoal filter train in service recirculating flow back to the containment cooling fans suction plenum.

3.4.2 Check the drywell atmosphere monitor on 1H13-P870 for abnormal particulate, or gaseous activity. If the readings are abnormally high, contact Chemistry to draw a sample from the drywell for analysis before venting.

SUPP POOL
CH-D TEMP
HI

Safety Related

Alarm Device 1M71-TAH-L605D

1.0 POSSIBLE CAUSES

1.1 Suppression pool temperature 85.5°F due to:

1.1.1 Safety/relief valve lifted or weeping.

1.1.2 RCIC running without suppression pool cooling in operation

1.1.3 Loss of coolant accident

1.1.4 One of the six channel D instruments greater than 85.5°.

> NOTE
>
> Any of the six channel D suppression pool temperature detectors can initiate this alarm.

2.0 AUTOMATIC ACTION

2.1 None

3.0 IMMEDIATE OPERATOR ACTION

> NOTE
>
> The six channel D temperature detectors read out on the PDS Computer and back-panel temperature indicating switches only. Recorder 1M71-R605D is fed from channel B instruments.

3.1 Check all safety/relief valves closed.

3.2 Check PDS Computer points M71N606C and M71N612C through M71N616C for channel C temperatures.

3.3 Check PDS Computer points M71N606D and M71N612D through M71N616D for channel D temperatures.

3.4 If average suppression pool temperature exceeds 95°F then enter EP-3, Containment Control.

Examination Outline Cross Reference	Level	RO
2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	Tier	3
	Group #	0
	K/A	2.4.21
	Rating	4.0
	Revision	1

Revision Statement:

Rev 1 Per NRC review (10 free view) changed wording in stem of parameters, HPCS system is injecting. Changed the answers to be clear and deleted reference to HPCS.

Per Facility Rep, added "low pressure " to stem before ECCS systems.

Question: 74

A grid disturbance has occurred causing a loss of offsite power.

Division 2 Diesel Generator has failed to start.

A LOCA has now occurred with the following parameters:

- Drywell pressure 1.58 psig
- Reactor water level -56 inches
- HPCS has started and is injecting
- At the time of initiation, a logic power failure occurred on RHR 'A'

The CRS has directed you to perform an ECCS Status.

Which of the following describes the current available Low Pressure ECCS systems?

A. RHR 'A' only, manual operation required.

B. LPCS only, automatic operation has occurred.

C. RHR 'A' and LPCS available for manual start and injection

D. RHR 'A' manual operation required and LPCS automatic operation has occurred

Answer: **C**

Explanation:

With a RHR 'A' logic power failure RHR 'A' pump can be started from the control room and the injection valve, E12-F042A can be manually opened from the Remote shutdown panel,

LPCS system can be manually started and injection valve manually opened from the main control room.

'C' is correct

Distracters:

‘A’ is wrong – With a RHR ‘A’ logic power failure RHR pump can be started from the control room and the injection valve can be opened from the Remote shutdown panel, LPCS system can be manually started and injection valve opened from the main control room

‘B’ is wrong – See description above

‘D’ is wrong – LPCS has not auto started due to the logic power failure, but can be manually started and manually open the injection valve.

K/A Match

This question requires the applicant to have the knowledge of parameters and ability to use to determine status of emergency systems.

Technical References:

04-1-02-1H13-P601-20A-H6, Rev. 169

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200 Obj. 15

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

4.3 **INVESTIGATE** logic **AND** relay problems at 1H13-P629 to determine **AND** correct trip unit **OR** logic problem.

4.4 **IF** a status light does **NOT** accompany the "OUT OF SERVICE" annunciator, **INVESTIGATE** the RHR A jockey pump breaker (52-153132) for overload or power loss.

4.5 **IF** the RHR loops **CANNOT** be returned to service, **MONITOR** RPV pressure water level, **AND** loop temperatures, **MAINTAIN** conditions stable **USING** alternate methods.

4.6 **REFER** to Technical Specifications 3.5.1, 3.5.2, 3.3.5.1, 3.3.6.1, 3.4.9, 3.4.10, 3.6.1.3, 3.6.1.7, 3.6.1.8, 3.6.2.3, 3.9.8, 3.9.9, 3.3.6.3 and Technical Requirements Manual 6.8.2.

4.7 **IF** RHR A Logic Power Failure status light is illuminated, the following Manual start of RHR A are effected as follows:

4.7.1 RHR A Can be manually started from 601 **AND** RSP.

4.7.2 Manual opening of E12-F042A is from RSP Only.

4.7.3 RHR A May be placed in Suppression Pool Cooling from P601 **AND** RSP with manual alignment of SSW A.

4.7.4 **INJECT** with LPCS from P601 by **MANUALLY STARTING** pump, E21-F005 Will operate normally.

4.7.5 RCIC Will operate normally in Auto **OR** Manually.

4.7.6 Containment spray **CANNOT** be initiated automatically **OR** manually.

4.7.7 No DIV I LOCA signal is received by LSS, D/G11, ADS A, CGCS A, **OR** SSW A.

4.7.8 E12-F064A Will **NOT** open **OR** close automatically.

4.7.9 Injection through RHR A feedwater is available by **OPENING** E12-F053A

4.8 **IF** annunciator is in alarm due to line break with Rx Power >80% **AND** Core Flow >90%, instrumentation Should be calibrated with on demand task.Line break instrumentation is no longer Tech Spec/ TRM related.

Examination Outline Cross Reference	Level	RO
2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.	Tier	3
	Group #	0
	K/A	2.4.45
	Rating	4.1
	Revision	0
Revision Statement:		

Question: 75

A Reactor Scram has occurred on MSIV closure.

The CRS has entered EP-2 and implemented transient alarm response.

Which of the following valid alarms should be immediately reported to the CRS?

A. ADS/SRV LEAK, P601-18A-G2

B. DRWL PRESS HI, P601-21A-E7

C. RECIRC PUMP A AUTO TRIP XFER TO LO SP, P680-3A-D4

D. SCRAM PILOT VLV AIR HDR PRESS LO, P680-5A-C4.

Answer: B

Explanation:

Per EN-OP-115-08, Annunciator Response, section 9, The announcement of transient alarms during Abnormal/ONEP and EOP is not required. In such cases, the operators are expected to announce those alarms that are of significance to the implementation of the applicable Abnormal/ONEP and EOP.

The Drywell Pressure Hi alarm should be immediately reported to the CRS. He is required to enter EP-3 and Re-enter EP-2. Also this alarm could be indicative of a LOCA in the drywell.

'B' is correct

Distracters:

'A' is wrong – This alarm will be in due to pressure control on the SRVs from the MSIV closure.

'C' is wrong – This alarms indicates an auto transfer to slow speed and is an action that should have already happened due to turbine trip.

'D' is wrong – This alarm will be in due to a scram signal and will not reset until the scram is reset.

K/A Match

This question requires the applicant to have the ability to recognize, interpret and prioritize control room alarms during a transient.

Technical References:

EN-OP-115-08, Annunciator Response
EN-OP-200, Rev. 5
04-1-02-1H13-P601-18A-G2, Rev. 166
04-1-02-1H13-P601-21A-E7, Rev 110
04-1-02-1H13-P680-5A-C4, Rev. 100
04-1-02-1H13-P680-3A-D4, Rev. 160

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

5.0 INSTRUCTIONS

5.1 TRANSIENT RESPONSE EXPECTATIONS

1. Parameter Control

NOTE

- Control bands are preferred to be as small as reasonably achievable commensurate with equipment availability and plant conditions without excessively burdening the Operator.
- Once controlling bands are established, operation of controls to maintain established bands is authorized without subsequent direction from the individual with command and control.

a. Control Room Supervisor (CRS) **IDENTIFY** critical parameters based on transient conditions.

b. CRS **COMMUNICATE** critical parameter control bands and trigger points as appropriate.

c. **ENSURE** directed control bands are within the limits set by applicable governing procedures.

d. **IF** operators determine that parameters are out of band,
THEN **NOTIFY** the CRS.

e. **IF** parameter can **NOT** be controlled in band,
THEN CRS **PROVIDE** a new control band.

2. Alarm Response

NOTE

- Announcement of transient alarms during abnormal/emergency operating procedures is not required.
- It is understood that not all annunciator response procedures are consulted during complex casualties and that abnormal and emergency operating procedures direct operator response under these circumstances.

a. **WHEN** numerous alarms are received,
THEN **PRIORITIZE** response procedure usage among procedures in effect.

b. During emergency/abnormal conditions, **ENSURE** alarms are silenced and acknowledged as soon as practical so as not to interfere with the transient response.

c. Operators **ANNOUNCE** alarms significant to implementation of the applicable abnormal or emergency operating procedure.

d. **USE** annunciators during events to guide operators and establish priorities.

DRWL
PRESS
HI

04-1-02-1H13-P601-21A-E7	
Revision: 110	Page 1 of 2

Safety Related

Alarm Device B21-PAH-L601A

1.0 POSSIBLE CAUSES

1.1 Drywell pressure 1.39 psig.

1.1.1 Steam leak inside the drywell.

1.1.2 Instrument or service air leak inside drywell.

1.1.3 Loss of coolant accident.

1.1.4 Loss of drywell cooling.

1.1.5 D/W purge compressor running.

2.0 AUTOMATIC ACTION

2.1 The following conditions exist:

2.2.1 ADS high drywell pressure permissive.

2.2.2 RHR A, B, C (LPCI) initiation.

2.2.3 LPCS initiation.

2.2.4 HPCS initiation.

2.2.5 RHR and LPCS test lines isolate (Group V).

2.2.6 Containment and drywell isolations (Group VI).

2.2.7 RCIC Exhaust vacuum breaker isolation (Group IX).

2.2.8 Containment spray permissive.

2.2.9 Division 1 and 2 Diesel Generators start.

2.2.10 HPCS Diesel Generator starts.

2.2.11 RHR (A, B, C), LPCS, HPCS room coolers start.

2.2.12 A and B loops of Standby Service Water start.

3.0 IMMEDIATE OPERATOR ACTION

3.1 If a scram has occurred, proceed with Reactor Scram ONEP 05-1-02-I-1. If a scram should have occurred but did not, proceed to EP-2.

3.2 Check that automatic actions have occurred.

3.3 Check drywell cooling is operating in accordance with SOI 04-1-01-M41-1.

3.4 Refer to EP-2, RPV Control, and EP-3, Containment Control.

3.5 Check narrow range drywell pressures on 1H13-P691, P692, P693, P694 on 1C71-N605A, B, C, D pressure indicator switches.

3.6 Refer to Attach 15 of 05-S-01-EP-2, if necessary.

4.0 SUBSEQUENT OPERATOR ACTION

4.1 If only one channel of drywell pressure indicated high, investigate the problem on 1H13-P629 and continue to monitor Drywell pressure.

4.2 Increase purge of the drywell with the ventilation system if Drywell pressure is high, in accordance with SOI 04-1-01-M41-1, if pressure is not due to a coolant leak.

4.3 Check that Service and Instrument Air Pressure is normal.

4.4 When the alarm condition is cleared refer to ONEP 05-1-02-III-5 to restore the systems isolated.

4.5 Refer to IOI 03-1-01-4, Scram Recovery.

ADS/SRV
LEAK

04-1-02-1H13-P601-18A-G2	
Revision: 166	Page 1 of 7

Safety Related

Alarm Device 1B21-TAH-L627

1.0 **POSSIBLE CAUSES**

1.0 ADS **OR** safety/relief valve discharge line temperature ≥ 155°F. as printed out on TJRS-R614 on Panel 1H13-P614.

2.0 Leaking **OR** misaligned Reactor Head Vent Valves B21-F001 **AND** B21-F002.

AUTOMATIC ACTION

1.0 None.

IMMEDIATE OPERATOR ACTION

1.0 None.

SUBSEQUENT OPERATOR ACTION

1.0 **OBSERVE** TJRS-R614 to determine affected valve.

2.0 **DETERMINE** degree of valve leakage by observing generator load meter **AND**/**OR** Suppression Pool temperature charts on Panel 1H13-P870.

3.0 **REFER** to EP-3, Containment Control.

4.0 **CHECK** downstream pressure switches to determine which ADS/SRV is leaking.

5.0 **IF** suppression pool temperatures are climbing due to an ADS **OR** safety/relief valve leak, **THEN START** RHR system in Suppression Pool Cooling mode in accordance with SOI 04-1-01-E12-1 **PRIOR** to suppression pool temperatures reaching 110 degrees.

GRAND GULF NUCLEAR STATION — ALARM RESPONSE INSTRUCTION

RECIRC PUMP A AUTO TRIP XFER TO LO SP

L631A

04-1-02-1H13-P680-3A-D4	
Revision: 160	Page 1 of 1

Safety Related

Alarm Device <u>B33-XA-</u>

1.0 POSSIBLE CAUSES

1.1 Turbine stop valve closure or control valve closure (EOC-RPT) with reactor power > 30%

1.2 Feed flow < 3.0 Mlbm/hr for 15 seconds

1.3 Differential temperature between reactor dome and recirc suction < 7.4°F for 15 seconds

1.4 Reactor water level < 11.4"

2.0 AUTOMATIC ACTION

2.1 Recirc Pump A transfers to slow speed.

2.1.1 If the fault is sensed by both pump circuits, both recirc pumps will transfer to slow speed.

3.0 IMMEDIATE OPERATOR ACTION

3.1 Check that affected recirc pumps Trip or transfer to slow speed.

3.2 Refer to ONEP 05-1-02-III-3, Reduction in Recirculation System Flow Rate.

4.0 SUBSEQUENT OPERATOR ACTION

4.1 Determine the cause of the transfer to slow speed or the trip of Recirc Pump A.

4.2 Refer to Technical Specifications 3.4.1.

SCRAM PILOT VLV
AIR HDR PRESS
LO

Safety Related

Alarm Device 1C11-PAL-L606

1.0 POSSIBLE CAUSES

1.1 Low pressure due to loss of instrument air

1.2 Low pressure due to plugged instrument air filter D006 or D026

1.3 Instrument air pressure of 75 psig

1.4 Instrument air header rupture

2.0 AUTOMATIC ACTION

2.1 None

NOTE

Low air pressure may cause control rods to drift.

3.0 IMMEDIATE OPERATOR ACTION

3.1 Check the instrument air header pressure indication on 1H13-P870.

3.2 Refer to ONEP 05-1-02-V-9, Loss of Instrument Air.

3.3 Observe rod position indicators.

4.0 SUBSEQUENT OPERATOR ACTION

4.1 Ensure the Instrument Air System is operating in accordance with SOI 04-1-01-P53-1.

GGNS ILT 2/2020 **NRC** INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

SRO EXAM

ANSWER KEY

76	A
77	C
78	B
79	B
80	B
81	C
82	C
83	D
84	B
85	C
86	A
87	A
88	D
89	D
90	C
91	B
92	C
93	B
94	D
95	B
96	D
97	B
98	D
99	C
100	A

Examination Outline Cross Reference	Level	SRO
295003 Partial or Complete Loss of A.C. Power	Tier	1
	Group #	1
AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :	K/A	295003 AA2.02
	Rating	4.3
AA2.02 Reactor power / pressure / and level.	Revision	2
Revision Statement: **Rev 1 – Validation comment add MSIVs have closed instead of the student determine that they closed** **Rev 2 – Validation comment Added No other EP Entry conditions have been exceeded.**		

Question: 76

The plant is operating at rated conditions.

MSIVs have closed

Current conditions:

- Rx power is 11%
- Rx pressure is maintaining between 926 and 1073
- Rx water level is +10” and lowering
- No other EP entry conditions have been exceeded

With the given conditions,

(1) What is the CRS’s direction for RPV level control?

(2) What is the CRS’s direction for RPV pressure band?

A. (1) Lower level to -70 inches Wide Range
(2) Lower Pressure band 450 to 600 psig

B. (1) Lower level to -70 inches Wide Range
(2) Maintain pressure band 800 to 1060 psig

C. (1) Lower level to -167 inches Fuel Zone Range or <5% power
(2) Lower Pressure band 450 to 600 psig

D. (1) Lower level to -167 inches Fuel Zone Range or <5% power

(2) Maintain pressure band 800 to 1060 psig

Answer:	**A**

Explanation:

An ATWS occurs due to power at 11% after MSIVs are closed and with water level at 10 inches.

SRVs are open due to the pressure being maintained is 926 to 1073 psig which is the opening and closing setpoints for the lo-lo set SRVs.

The CRS should enter EP-2 then to EP-2A and lower level to -70" Wide Range. Pressure control should be manually controlled at a band of 450 to 600 psig due to loss of feedwater and lowering band will allow Condensate to feed and control level.

Level should be lowered to -70 inches per EP-2A, the lowering to TAF is a Step 8 terminate action and that criteria is not met yet. Step L5 has to have suppression pool temperature >110.

'A' is correct.

Distracters:

'B' is wrong, the pressure band is incorrect. Pressure must be lowered to ensure the Condensate system is able to feed.

'C' is wrong, Level should be lowered to -70 inches per EP-2A, the lowering to TAF is a Step 8 terminate action and that criteria is not met yet. Step L5 has to have suppression pool temperature >110.

'D' is wrong, Level should be lowered to -70 inches per EP-2A, the lowering to TAF is a Step 8 terminate action and that criteria is not met yet. Step L5 has to have suppression pool temperature >110. Also, the pressure band is incorrect. Pressure must be lowered to ensure the Condensate system is able to feed.

K/A Match

The SRO candidate must be able to interpret the given information with a loss of power and decide on the correct actions to take.

Technical References:

04-1-02-1H13-P601-19A-E4, ARI, Rev. 162
GLP-OPS-B1300, Rev 14, Nuclear Boiler and Vessel Internals Lesson Plan
02-S-01-40, EP Bases, Rev. 9
04-1-01-C71-1, RPS SOI, Rev. 37

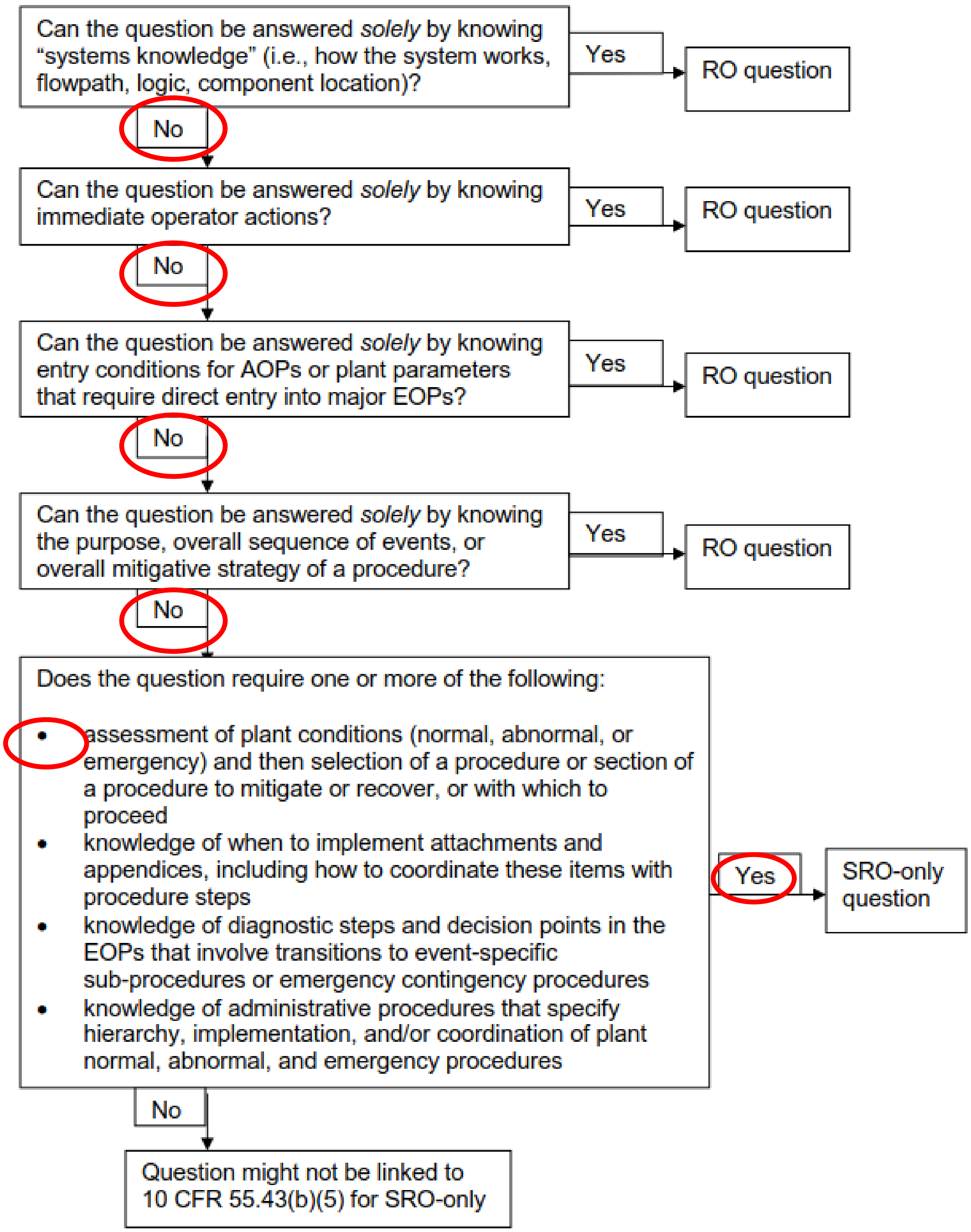
Handouts to be provided to the Applicants during exam:

None

Learning Objective: GLP-OPS-B1300, Rev 14 OBJ. 11.1 GLP-OPS-EP2A		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
R21 ESF is listed as #8 on the System Importance to CDF. RPS is listed as #5 on the System Importance to CDF.		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



GRAND GULF NUCLEAR STATION   ALARM RESPONSE INSTRUCTION

MSIV/DR VLV TRIP INIT

04-1-02-1H13-P601-19A-E4	
Revision: 162	Page 1 of 1

Safety Related

Alarm Device 1B21-FAH-L710

1.0 POSSIBLE CAUSES

1.1	Reactor Vessel Water Level 1	-150.3"
1.2	Main Condenser Vacuum Low	9"H
1.3	Main Steam Tunnel Temperature High	185°F
1.4	Main Steam Line Pressure Low	<849 psig (RUN MODE ONLY)
1.5	Main Steam Line Flow High	252.5 psid
1.6	Loss of ESF Bus 15AA **OR** 16AB	(De-energizes NSSSS Isolation Logic)

2.0 AUTOMATIC ACTION

2.1 Main Steam Line Isolation (Group 1)**, IF** Both NSSS trip systems are tripped.

2.2 **IF** full isolation occurred, **THEN** the following valves Will close:

B21-F022A, B, C, D   B21-F067A, B, C, D
B21-F028A, B, C, D   B21-F016, B21-F019

3.0 IMMEDIATE OPERATIOR ACTION

3.1 **CHECK** that appropriate automatic actions occur.

3.1.1 **CHECK** MAIN STEAM LINE PILOT SOLENOID A/B indicating lights on 1H13-P622 **AND CHECK** that half of Group 1 isolation is present.

3.1.2 **DETERMINE** cause of alarm **AND RESET** half isolation as soon as possible.

3.2 **IF** reactor scram has occurred, **THEN ENTER** scram ONEP 05-1-02-III-5.

4.0 SUBSEQUENT OPERATOR ACTION

4.1 **WHEN** plant status has returned to normal, **RESET** isolation signal by **DEPRESSING** NSSS OTBD **AND** NSSS INBD ISOLATION RESET pushbuttons on 1H13-P601.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 **WHEN** performing an RPS power supply transfer, there is an interruption of power to the bus.

3.1.1 RPS power supply transfer Will cause a half scram, MSIV half isolation **AND** effects EOOS Risk.

3.2 On a loss of power to either RPS bus, **REFER** to ONEP 05-1-02-III-2, Loss of One **OR** Both RPS Buses, to determine which RPS bus is lost.

3.3 **WHEN** attempting to start **OR** shut down an RPS MG set, MOTOR ON **OR** OFF pushbutton Must be depressed until MOTOR ON RED light lites **OR** goes out.

3.4 **IF** Both A **AND** B RPS buses are to be transferred, Half-Scram Must Be reset after first bus is transferred (before transfer of second RPS bus).

3.5 RPS preferred power source Should be RPS MG set, But **EITHER** alternate supplies Can be used **IF** necessary. **HOWEVER** in Modes 1 **AND** 2, Both RPS buses Must **NOT** be fed concurrently by their respective alternate supplies **EXCEPT** for limited emergent plant situations which require Both RPS buses to be connected to their alternate supplies for short periods. Alignment of Both RPS buses to their alternate supplies is **NOT** the normal line up because of increased vulnerably to grid perturbations that Could result in inadvertent trip of Both divisions of RPS connected loads.

3.5.1 Short periods are **NOT** to exceed the time required to correct the limit emergent plant situation(s) to restore at least one RPS buss to normal power supply.

3.5.2 GMPO approval is required prior to placing Both RPS buses on Alternate Supply. Reference CR-2016-0221.

3.6 Before transferring power on an energized RPS bus, **ENSURE** NO SCRAM **OR** ISOLATION surveillances are in progress **AND** **ALL** MSIV solenoids are energized. (MSIV solenoids can be verified energized by all MSIV solenoid lights are energized **AND** all pilot solenoids indicate amperage on 1H13-622 **AND** 1H13-P623).

4.0 NORMAL OPERATIONS

4.1 Motor/Generator Startup

4.1.1 Prerequisites

a. Handswitch Alignment Checksheet, Attachment V, completed except switches with * (asterisk), which line up in Section 5.2.

b. Electrical Lineup Checksheet, Attachment III, completed except breakers with * (asterisk), which are closed in Sections 4.1 **AND** 4.3, **IF** breakers are open.

c. Manual Valve lineup in 04-1-01-N32-1 is complete for isolation valves to C71-PT-N005A-D (N32-FX001; FX002; FX003 **AND** FX003) **AND** C71-PT-N006A-H (N32-FX005; FX006; FX007; FX010; FX008; FX009; FX011 **AND** FX012).

Step L-5

L-5

IF	AND	AND	AND	THEN
Reactor power is above **5%** or unknown	SP temperature is above **110°F**	Any SRV is open **OR** DW pressure is above **1.23 psig**	RPV level is above **-167 in.**	Go to ⑧
	- - - - - - - - - -	- - - - - - - - - -	RPV level is above **-70 in.**	Go to ⑦

Discussion

Step L-5 defines conditions under which RPV water level is intentionally lowered to mitigate ATWS events.

The first row of the override defines conditions for lowering RPV water level to reduce reactor power, thereby minimizing the energy addition to the primary containment.

The second row of the override defines conditions for lowering RPV water level to prevent thermal-hydraulic instabilities.

The condition “reactor power is above 5% or unknown” is common to both rows. This power level corresponds to the APRM downscale setpoint. If power is “unknown,” it must be assumed to be above 5%.

Lowering RPV water level to reduce reactor power

A combination of high reactor power, high suppression pool temperature, and an open SRV or high drywell pressure indicates that heat is being added to the suppression pool faster than it is being removed by available suppression pool cooling. This condition could ultimately result in overpressurization and loss of primary containment integrity. Loss of containment integrity could, in turn, lead to a loss of adequate core cooling and uncontrolled release of radioactivity to the environment. RPV water level is therefore lowered to reduce reactor power while efforts to shut down the reactor continue.

Step L-7

7

Lower level to reduce subcooling:

1. Terminate and prevent all RPV injection except:
   - Boron injection
   - CRD
   - RCIC

   OK to defeat interlocks.

2. Let level drop to **-70 in.** (Wide Range).

Lowered level: -70 in.

L-7

Nominal Band
-70 in. to -130 in.

Maximum Band
-70 in. to -191 in.

Discussion

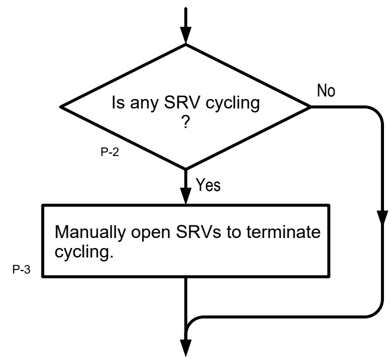
If the second override in Step L-5 is satisfied, RPV water level must be lowered to reduce subcooling and prevent instabilities. The level reduction is accomplished by terminating and preventing all injection into the RPV except from boron injection systems, CRD, and RCIC. Level then decreases by boil-off or losses through any primary system breaks that may exist.

Injection from boron injection systems and CRD is allowed to continue because the flow rates are relatively small and the systems may be needed to shut down the reactor. RCIC injection need not be terminated since its flow rate is also relatively small and because continued RCIC operation may facilitate reestablishing control of RPV water level when injection is restarted. The marginal decrease in the rate of water level reduction due to RCIC operation is expected to have a negligible impact on core inlet subcooling and suppression pool temperature.

When RPV water level is lowered significantly below the normal operating range with the reactor still at power, short term bundle-to-bundle flow instabilities may occur and induce power oscillations of noticeable magnitude. These oscillations are expected and analyses have demonstrated that the resulting thermal transients are well within the design capabilities of the fuel. RPV water level should be lowered as required by EP-2A even if such oscillations are observed.

The final level reached in Step L-7 defines the upper end of the RPV water level control band established in Element L-9. Since the previous instruction is to allow level to drop to -70 in. this blank is pre-completed. This format is to be consistent with step L-8 when this value is required in Step L-9.

Steps P-2 and P-3



Discussion

"SRV cycling" is defined to be multiple, closely sequenced valve actuations with the valves opening as RPV pressure exceeds the lift setpoints and closing as pressure drops below the reset setpoints. Cycling is undesirable and warrants prompt manual action for the following reasons:

It exerts significant dynamic loads upon the RPV, the SRV tail pipes and supporting structures, and the primary containment.

Swell and shrink associated with the valve actuations cause RPV water level fluctuations that complicate level control actions.

Under failure-to-scram conditions, the consequent level and pressure oscillations can result in significant power transients.

The potential for a stuck open relief valve is increased.

SRV cycling is terminated by manually opening any SRVs that are either open or cycling on high pressure. This action establishes direct operator control of the SRVs and RPV pressure and precludes further automatic actuation cycles.

With the SRVs open, RPV pressure will decrease toward some equilibrium value dependent on reactor power. No pressure control band is specified here, since the purpose of the instruction is simply to terminate cycling. Nor is any direction to close the SRVs after they are manually opened intended; the control and stabilization of RPV pressure after SRV cycling is terminated is addressed in Step P-4.

The frequency at which SRV actuations are judged to constitute "cycling" is dependent on the effect of the actuations upon RPV water level and pressure, the effectiveness of automatic control systems, the practicability of manual control, and the availability of personnel for performance of manual control functions. If SRV actuations are interfering with RPV water level control actions or are judged too rapid for any other reason, the valves should be opened manually.

Step P-4

❺

P-4

Stabilize RPV pressure below **1064.7 psig** using one or more Table P-1 systems.

- If the ADS receiver is below **95 psig**, place all ADS control switches in OFF.

Normal Band
800–1060 psig

Reduced Band
450–600 psig

Discussion

Stabilize RPV pressure below 1064.7 psig

RPV pressure is stabilized to facilitate control of RPV water level and reactor power. Pressure oscillations may result in RPV water level fluctuations due to the effects of shrink and swell and power transients due to changes in the void fraction. Pressure is stabilized below the high RPV pressure scram setpoint (1064.7 psig) to avoid SRV actuation and to permit the scram logic to be reset (provided no other scram signal exists). No minimum value is specified since the RPV pressure at which the EPs are entered cannot be predefined. A target pressure should be selected close to the initial value and below the RPV pressure scram setpoint that permits 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 100°F/hr cooldown rate limit is not exceeded. A large pressure reduction, such as from an intermediate pressure of 400 psig to below the shutdown cooling RPV pressure interlock (135 psig), should not be required and is not considered a permissible adjustment even if the depressurization is within the LCO cooldown rate.

A Nominal Band of 800 psig to 1060 psig and a Reduced Band of 450 psig to 600 psig are recommended as typical operational targets within the procedural control band. While other target ranges may be appropriate under unusual circumstances or extremely degraded conditions, in most cases the recommended bands are expected to provide the desired operational flexibility.

“Stabilize” means to hold RPV 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:

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

Examination Outline Cross Reference	Level	SRO
295006 SCRAM AA2. Ability to determine and/or interpret the following as they apply to SCRAM : AA2.03 Reactor water level	Tier	1
	Group #	1
	K/A	295006 AA2.03
	Rating	4.3
	Revision	1
Revision Statement: **Rev 1 - Validation comment, add "RCIC has tripped on overspeed" to ensure SRO knows level is being controlled by a high volume system.**		

Question: 77

A complete loss of Feedwater occurred at rated power.

Both Reactor Feed Pumps are tripped due to a spurious high water level signal that remains sealed in.

RCIC has tripped on overspeed.

The CRS directs level control using HPCS.

Reactor water level is +20 inches Narrow Range.

Which of the following Scram ONEP steps should the CRS direct the RO to **skip**?

A. Inserting IRMs.

B. Restart RWCU.

C. Resetting RPS Scram logic.

D. Restarting Reactor Recirc Pumps.

Answer: **C**
Explanation: The level band when HPCS is used for level control is -30" to +50" per 02-S-01-43 and EP-2. Scram ONEP 05-1-02-I-1 step 3.7 directs resetting the RPS scram only if level is stable, to ensure subsequent scrams are not received. Since level is expected to go below 11.4" for the current level band, resetting the scram is not appropriate.

‘C’ is correct.		
Distracters: Candidate must assume that RPV level lowered to less than level 2 (-41.6”). At rated conditions and a total loss of Feedwater level will lower to approximately -60” to -80” All distracters are plausible because they are Scram ONEP subsequent actions, but they are wrong because they are advantageous and are not affected by current conditions. ‘A’ is wrong, Step 3.6.2 of the SCRAM ONEP states to insert IRMs, there is no reason not to insert the IRMs at this time. ‘B’ is wrong, Step 3.10 states in the CAUTION . **IF** no Reactor Recirculation pumps are running, **THEN** section 3.11 **AND** 3.12 **Should** be performed concurrently without delay to prolong the onset of thermal stratification. Step 3.10 states **IF** Reactor Water Cleanup System (RWCU) has tripped, **THEN RESTART** RWCU per Attachment I after Reactor water level **AND** pressure have stabilized. ‘D’ is wrong, See 3.10 CAUTION from above. Step 3.11 states, **IF** no Reactor Recirculation pumps are running, **THEN PERFORM** the following without delay. Step 3.11.4 **START** Both Reactor Recirculation pumps, **IF** available, in slow speed per Attachment II.		
K/A Match The candidate must be able to interpret the given information on reactor water level and conditions and decide on the correct actions to take.		
Technical References: 05-1-02-I-1, Reactor Scram ONEP, Rev. 132 02-S-01-43, Transient Mitigation Strategy, Rev. 7		
Handouts to be provided to the Applicants during exam: None		
Learning Objective: GLP-OPS-PROC, OBJ. 11.1 GLP-OPS-ONEP, OBJ. 2, 3		
Question Source:	**Bank # 679**	Bank not on any NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	

	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

NOTE

SRMs May be inserted simultaneously with IRMs.

3.6.2 **INSERT** IRMS as follows;

a. **IF** Reactor SCRAM signal has **NOT** been reset
THEN INSERT all IRMs

b. **IF** Reactor SCRAM signal has been reset
THEN:

1. **INSERT** one RPS division of IRMs (A,C,E,G **OR** B,D,F, H).

2. **WHEN** the first RPS division of IRMs are fully inserted,
THEN INSERT the second RPS division of IRMs.

3.6.3 **INSERT** all SRMs.

CAUTION

IF no Reactor Recirculation pumps are running, **THEN** the Reactor Scram **Should** be reset without delay **WHEN** conditions allow to help delay thermal stratification of Reactor coolant.

3.7 **IF** Reactor Scram signal Can be cleared **AND** Reactor vessel level **AND** pressure are stable,
THEN RESET Reactor Scram **AND RETURN** CRD System to normal as follows:

3.7.1 **BYPASS** Scram Instrument Volume High Level scram by **PLACING** CRD DISCH VOL HI TRIP BYP switches RPS Div 1, 2, 3, 4 to BYPASS.

3.7.2 **IF** ATWS ARI/RPT has initiated,
THEN PERFORM the following to reset ATWS ARI/RPT initiation signal:

a. **VERIFY** Reactor pressure is less than 1126 psig **AND** Reactor level is Greater than - 41.6”.

b. **VERIFY** ATWS ARI/RPT READY TO RESET status light is illuminated on 1H13-P680.

c. **DEPRESS** the ATWS ARI/RPT RESET pushbutton on 1H13-P680.

6.6.6 (Cont)

a. A target band as allowed by EP-2/2A may be given even **IF** present operation is **NOT** within the target band.

Operator **DOES NOT** have to report outside of band, but should report **IF** he anticipates target **CAN NOT** be attained/maintained.

This **DOES NOT** mean desired/target band has to be changed; acceptable expanded control band should be assigned **UNTIL** target band can be maintained.

b. Following are recommended nominal bands **AND** applicable control/instrument ranges for given conditions:

Level Control

Nominal level bands provided are recommended bands based on assumed plant conditions. Should these bands **NOT** be practical based on actual plant conditions,
THEN Control Room Supervision should modify bands according to steps provided in EOPs.

Condition	**Nominal Level Band**	**Instrument Range**
Normal	+11.4 inches to +53.5 inches	• P680 Narrow Range **OR** • P601 Wide Range
Widened (Expanded) ECCS **OR** SRVs (non-ATWS)	-30 inches to +50 inches	• Wide Range
ATWS with Initial Reactor Power less than or equal to 5 percent.	+11.4 inches to +53.5 inches **OR** +15 inches to +50 inches	• P680 Narrow Range **OR** • P601 Wide Range
ATWS with initial reactor power greater than 5 percent **AND** Suppression Pool less than 110F	-70 inches to -130 inches	• Wide Range **OR** • Density Compensated Fuel Zone Computer Point
ATWS with initial reactor power greater than 5 percent **AND** Suppression Pool greater than **OR** equal to 110°F	Level Which Lowered to -191 inches	• Density Compensated Fuel Zone Computer Point **OR** • Figure 1 Density Compensated Fuel Zone Nomograph

Examination Outline Cross Reference	Level	SRO
295025 High Reactor Pressure EA2. Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: EA2.02 Reactor power	Tier	1
	Group #	1
	K/A	295025 EA2.02
	Rating	4.2
	Revision	1
Revision Statement: **Rev. 1, Per Ops Rep, Changed correct answer to match current operations philosophy.**		

May 2017 NRC Exam Q 79

Question: 78

HANDOUT PROVIDED

An ATWS has occurred.

Current conditions are:

- Reactor power is 40%
- Reactor pressure is 1045 psig and steady
- Two SRVs are currently open
- Suppression pool level is at the Tech Spec low level limit
- Suppression pool temperature is 150°F and rising.
- RHR ‘A’ and RHR ‘B’ systems have just been placed in Suppression Pool Cooling

Which of the following describes the next action the CRS will direct?

A. Emergency Depressurize the Reactor when HCTL is exceeded

B. Lower pressure band to prevent entering unsafe zone of HCTL

C. Immediately Emergency Depressurize the Reactor when exceeding HCTL is predicted.

D. Anticipate an Emergency Depressurization and fully open the Main Turbine Bypass valves

Answer: B

Explanation:

With the plant maintaining current conditions, reactor power exceeds the capacity of the bypass valves therefore SRVs will continue to be opened to maintain pressure. With SRVs open Suppression pool temp will rise.

In the pressure branch of EP-2A, it states "If SP temp cannot be maintained in the Safe Zone of the HCTL THEN, maintain RPV pressure in the Safe Zone of the HCTL."

SRO must know EP steps without reference.

'B' is correct.

Distracters:

All other distractors are plausible, they all are different paths that could be taken if the student forgets the step in the EOPs and that HCTL can be exceeded if it can be restored to the safe zone.

'A' is wrong - Emergency Depress is required only if RPV pressure cannot be restored and maintained in the Safe Zone of the HCTL.

'C' is wrong: Emergency Depress is required only if RPV pressure cannot be restored and maintained in the Safe Zone of the HCTL. Since the RHR systems have just been started in Suppression Pool cooling the SRO should wait and determine if the limit can be restored.

'D' is wrong - Reactor power is too high, the Bypass valves are already fully open and procedures do not allow to anticipate an ED while in EP-2A ATWS.

K/A Match

The candidate must be able to interpret the effects of the given information with reactor power and reactor pressure to determine the next action.

Technical References:

EP-3 Rev date 3/23/2016
02-S-01-40 Rev 8, EP Technical Bases, Attachment V, pages 41, 44, and 45

Handouts to be provided to the Applicants during exam:

HCTL Curve

Learning Objective:

GLP-OPS-EP01

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #1318**	**May 2017 NRC Exam Q79**
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

May 2017 NRC Exam Q79

Question 79

An ATWS has occurred.

Current conditions are:

- Reactor power is 40%
- Reactor pressure is 1045 psig and steady
- Two SRVs are currently open
- Suppression pool level is at the Tech Spec low level limit
- Suppression pool temperature is 150°F and rising.
- RHR ‘A’ and RHR ‘B’ systems have just been placed in Suppression Pool Cooling

Which of the following describes the next action the CRS will direct?

A. Immediately Emergency Depressurize the Reactor when exceeding HCTL is unavoidable.

B. Anticipate an Emergency Depressurization and fully open the Main Turbine Bypass valves.

C. Initiate Suppression Pool Makeup System prior to Emergency Depressurization.

D. Emergency Depressurize the Reactor when HCTL is exceeded.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following:

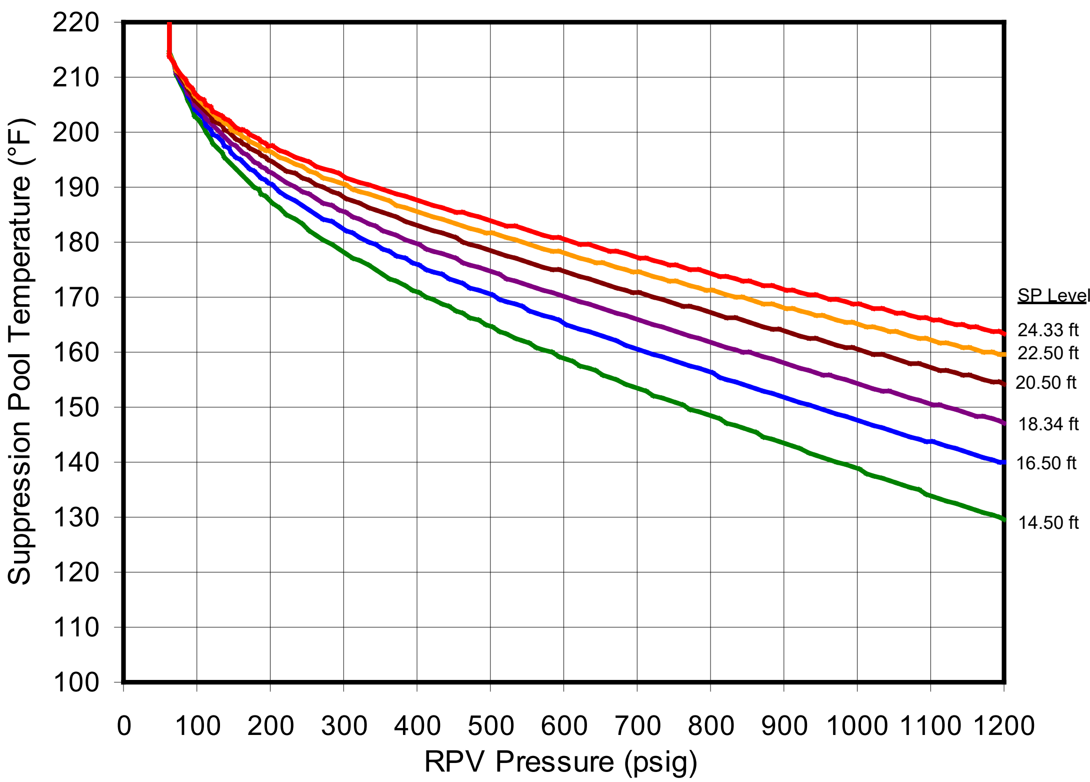
- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Fig 1
Heat Capacity Temperature Limit (HCTL)



Step P-1

PRESSURE

IF	THEN
DW pressure is above **1.39 psig**	Prevent injection from any ECCS pump not required for adequate core cooling before depressurizing below its maximum injection pressure.
Emergency depressurization is or has been required	Exit Level and Pressure → Enter **Emergency Depressurization**
SP temperature cannot be maintained in the **Safe** zone of the **HCTL** (Figure 1)	Maintain RPV pressure in the **Safe** zone of the **HCTL** (Fig 1). IF.........RCIC is required for adequate core cooling THEN... Maintain RPV pressure adequate for RCIC operation OK to exceed **100°F/hr** cooldown.
Boron injection is required at 11 (Q-4), **AND** The main condenser is available, **AND** There has been no indication of a steam line break outside CTMT	Use the main condenser as a heat sink. Open MSIVs if necessary. OK to defeat isolations................**Att 7, 8**

2

P-1

Discussion

PRESSURE

The Pressure branch first stabilizes RPV pressure below the high RPV pressure scram setpoint and, if necessary, depressurizes and cools down the RPV to cold shutdown conditions. Discharging steam to the main condenser using the main turbine bypass valves is the preferred method for controlling RPV pressure. Alternate methods are identified should the bypass valves or main condenser be unavailable.

DW pressure is above 1.39 psig

LPCS and LPCI initiate automatically on high drywell pressure (1.39 psig) and automatically inject when RPV pressure decreases below the shutoff head of the pumps. If these systems are not needed for core cooling, injection may be prevented to facilitate RPV water level control and avoid power excursions. Subsequent use of these systems is not prohibited by the override if plant conditions change such that system operation is required to assure adequate core cooling.

Emergency depressurization is or has been required

If emergency depressurization is required by any EP instruction while EP-2A is in use, RPV pressure control transfers from the Pressure branch to the Emergency Depressurization contingency. The Emergency Depressurization contingency provides appropriate direction for

Discussion (continued)

rapidly depressurizing the RPV and preventing RPV repressurization. The consequences of not depressurizing the RPV under conditions which require emergency RPV depressurization could include a loss of adequate core cooling or failure of the primary containment.

An emergency depressurization requirement supersedes the RPV pressure control strategies implemented in the Pressure branch of EP-2A. Once emergency RPV depressurization is required, the Emergency Depressurization contingency remains in effect until an emergency no longer exists, RPV flooding is required, or the SAPs are entered. This structure ensures that guidance on preventing repressurization of the RPV remains applicable following the initial blowdown. If EP-2A must be re-entered after emergency depressurization has been performed, the condition “Emergency RPV Depressurization…has been required” returns control of RPV pressure to the Emergency Depressurization contingency.

SP temperature cannot be maintained in the Safe zone of the HCTL

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature at which initiation of emergency RPV depressurization will not result in exceeding either:

- The maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized, or,
- The Primary Containment Pressure Limit,

before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. The derivation of the HCTL is discussed in Section 12.0.

Suppression pool temperature is controlled concurrently in the SP Temperature branch of EP-3. If the actions being taken to limit suppression pool temperature increase are ineffective, RPV pressure may be reduced to provide additional operating margin. Full depressurization, however, may result in loss of RCIC. RPV depressurization to mitigate primary containment challenges must therefore be coordinated with core cooling strategies. Full depressurization is appropriate only if adequate core cooling will not be sacrificed as a result. Loss of adequate core cooling would compound containment challenges and increase any resulting radioactivity release. Core cooling is thus prioritized over other EPG objectives. If, at any time during RPV depressurization, it is anticipated that continued pressure reduction will result in loss of injection flow required for adequate core cooling, the depressurization is terminated and pressure stabilized at its existing value.

The normal cooldown rate limit may be exceeded to the extent necessary to maintain RPV pressure below the HCTL. If RPV pressure *cannot* be maintained below the HCTL, emergency RPV depressurization will be required, possibly resulting in an even more rapid cooldown.

Examination Outline Cross Reference	Level	SRO
295027 High Containment Temperature (Mark III Containment Only) 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	Tier	1
	Group #	1
	K/A	295027 2.2.25
	Rating	4.2
	Revision	2

Revision Statement:

Rev 1, Facility Rep. grammar on stem question, added “is to” and changed answers to fit the format.

Rev 2, Added clarity to answer ‘B’ by adding “during a DBA.

Question: 79

One bases for the Tech Spec ACTION to immediately suspend RCIC flow testing based on high Suppression Pool temperature is to….

A. Preclude having to enter EP-3, Containment Control.

B. Ensure the Containment temperature limit is not exceeded during a DBA.

C. Prevent RCIC turbine exhaust check valve chatter that can damage RCIC piping.

D. Prevent elevated RCIC turbine lube oil temperature that could damage RCIC turbine.

Answer: B

Explanation:

In the stem, RCIC flow testing implies power is above 1%. TS 3.6.2.1 Action C.1 requires immediately suspending testing that adds heat to the Supp Pool when Supp Pool average temperature exceeds 105°F. The implication in the stem is Supp Pool temperature has just exceeded 105°F.

Suppression pool average temperature is allowed to be > 95°F with THERMAL POWER > 1% RTP when testing that adds heat to the suppression pool is being performed. However, if temperature is > 105"F, the testing must be immediately suspended to preserve the pool's heat absorption capability. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

TS bases 3.6.2.1 states the Supp Pool temperature limit was developed to address technical concerns that include limiting containment average air temperature to < 185°F during the DBA.

‘B’ is correct.

Distracters:

'A' is wrong - is plausible because entry into an Emergency Procedure is undesirable and implies elevated risk and challenge to plant safety, but it is wrong because TS 3.6.2.1 Action C that requires suspending testing that is adding heat to the Supp Pool at 105°F, so the Supp Pool Temperature EP-3 entry condition of 95°F would have already been met, so EP-3 entry cannot be avoided.

'C' is wrong - is plausible because exhaust check valve operation could theoretically be affected mechanistically by Supp Pool parameters, but it is wrong because 04-1-01-E51-1 step 3.2 states this is a concern for operation of RCIC at speeds below 2000 rpm and because RCIC operability is not a concern in the DBA analysis and is not mentioned in the bases for TS 3.6.2.1.

'D' is wrong - is plausible because it describes undesired effects of elevated Supp Pool temperature on RCIC. It is each wrong because 04-1-01-E51-1 step 3.3 states RCIC suction temperature is limited to 140°F for non-emergency operation to prevent bearing/turbine seals overheating, and during emergencies, EP caution 4 states RCIC equipment damage could occur if Supp Pool temperature exceeds 225°F, well above the 105°F condition implied in the stem and because RCIC operability is not a concern in the DBA analysis and is not mentioned in the bases for TS 3.6.2.1.

K/A Match

The candidate must have the knowledge of the bases in Technical Specifications for limiting conditions for operations on Suppression Pool Temp.

Technical References:

TR 3.6.2.1 and bases
EP-3
EP Caution 4

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

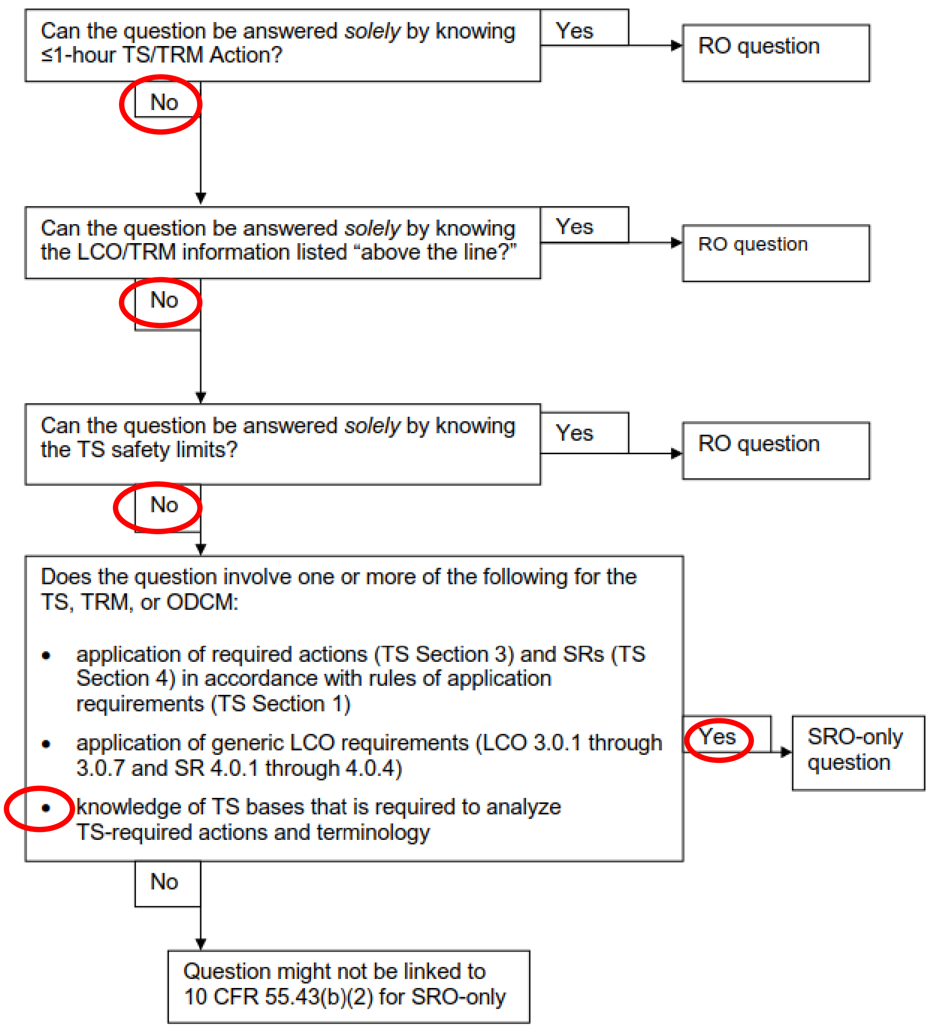
GLP-OPS-TS001, OBJ. 39

Question Source:	**Bank # 694**	2013 NRC Exam Q79
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	2.0	
PRA Applicability:		

RCIC is listed as #17 on the System Importance to CDF.

SRO JUSTIFICATION

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression pool is a concentric open container of water with a stainless steel liner that is located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve discharges or from a loss of coolant accident (LOCA). The suppression pool must also condense steam from the Reactor Core Isolation Cooling System turbine exhaust and provides the main emergency water supply source for the reactor vessel. The amount of energy that the pool can absorb as it condenses steam is dependent upon the initial average suppression pool temperature. The lower the initial pool temperature, the more heat it can absorb without heating up excessively. Since it is an open pool, its temperature will affect both primary containment pressure and average air temperature. Using conservative inputs and methods, the maximum calculated primary containment pressure during and following a Design Basis Accident (DBA) must remain below the primary containment design pressure of 15 psig. In addition, the maximum primary containment average air temperature must remain < 185°F.

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation;

b. Primary containment peak pressure and temperature;

c. Condensation oscillation (CO) loads; and

d. Chugging loads.

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial suppression pool water volume and suppression pool temperature (References 1 and 2). An initial pool temperature of 95°F is assumed for the Reference 1 and 2 analyses. Reactor shutdown at a pool temperature of 110°F and vessel

(continued)

BASES

ACTIONS (continued)

C.1

Suppression pool average temperature is allowed to be > 95°F with THERMAL POWER > 1% RTP when testing that adds heat to the suppression pool is being performed. However, if temperature is > 105°F, the testing must be immediately suspended to preserve the pool's heat absorption capability. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

D.1, D.2, and D.3

Suppression pool average temperature > 110°F requires that the reactor be shut down immediately. This is accomplished by placing the reactor mode switch in the shutdown position. Further cooldown to MODE 4 is required at normal cooldown rates (provided pool temperature remains ≤ 120°F.) Additionally, when pool temperature is > 110°F, increased monitoring of pool temperature is required to ensure that it remains ≤ 120°F. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high pool temperature in this Condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

E.1 and E.2

If suppression pool average temperature cannot be maintained ≤ 120°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 200 psig within 12 hours and the plant must be brought to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

Continued addition of heat to the suppression pool with pool temperature > 120°F could result in exceeding the design

(continued)

Examination Outline Cross Reference	Level	SRO
295031 Reactor Low Water Level EA2.Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : EA2.03 Reactor pressure	Tier	1
	Group #	1
	K/A	295031 EA2.03
	Rating	4.2
	Revision	1
Revision Statement: **Rev 1 Per NRC review (10 free view) changed wording in stem of question, added "procedurally required". Added more explanation about entering alternate level control leg of EOPs.**		

Question: 80

The plant has experienced a LOCA

All High pressure injection systems are not available.

Reactor Water level has reached the lowest indication for Wide Range.

Which of the following describes the procedurally required RPV pressure control being directed by the CRS?

A. LO-LO Set SRVs.

B. Emergency Depressurization.

C. Pressure band 450 – 600 psig using SRVs.

D. Pressure band 800 – 1060 psig using Main Bypass Valves.

Answer: **B**
Explanation: In the stem, when all high pressure feed is lost the CRS/SRO should enter the alternate control leg of the EOPs. There, other systems are started in an attempt to restore level. If RPV water level continues to lower as Reactor water level reaching the lowest indication is -160" Wide Range. Per EP-2 When level drops to -160 in. Emergency Depressurization is Required. The SRO must recognize that the reactor must be depressurized. 'B' is correct.

Distracters:

'A' is wrong - is plausible because during a LOCA it is permissible to use LO-LO Set system to control pressure within band. However, the given information requires an ED.

'C' is wrong - is plausible because this is a band that is used to feed with Condensate. However, with no high pressure feed bases states not to lower the pressure band with nothing to feed.

'D' is wrong – With level indicated at -160" Wide Range the MSIVs would have closed at -150.3", therefore, Main Bypass Valves are unavailable.

K/A Match

The candidate must have the knowledge of how to control reactor pressure with a low reactor water level.

Technical References:

02-S-01-40, EP Bases, Rev. 9

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-EP02

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following:

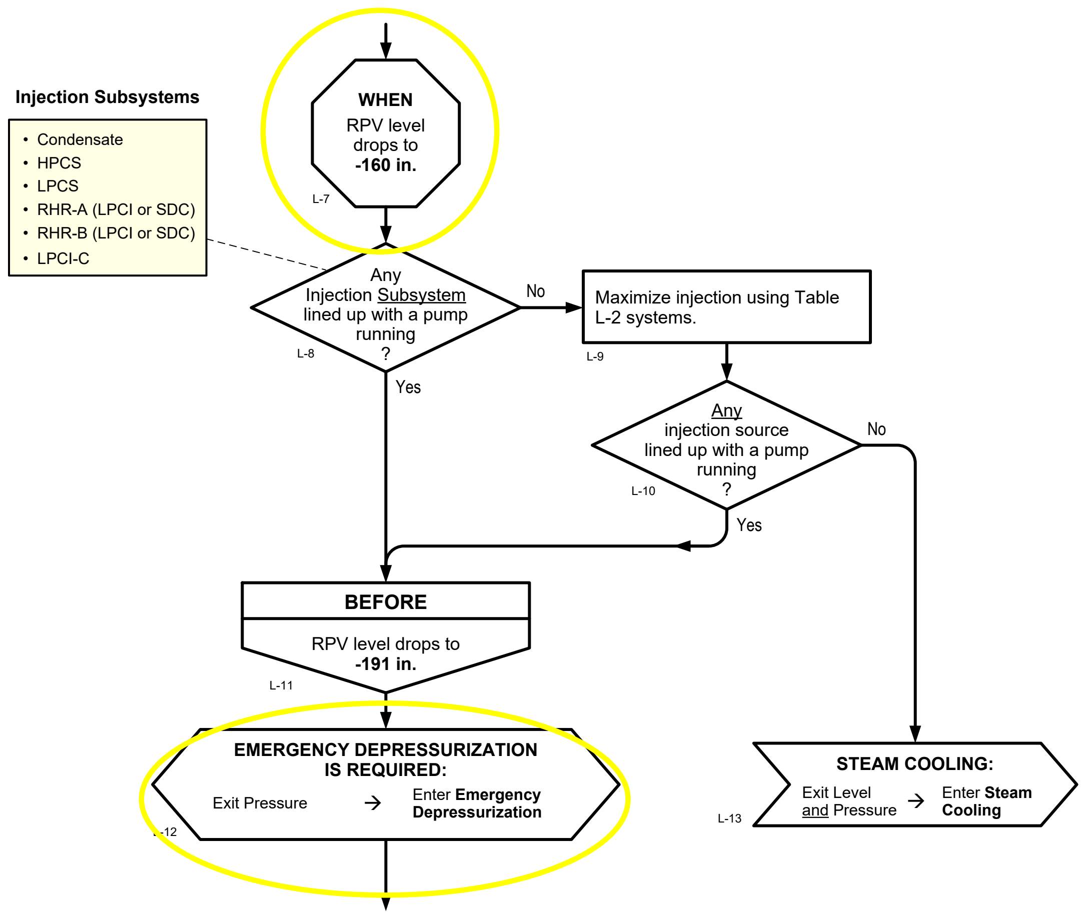
- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Steps L-7 through L-13



Discussion

WHEN RPV level drops to -160 in.

If no Injection Subsystem is available and the decreasing RPV water level trend cannot be reversed by the time the level drops to the top of the active fuel, injection from Alternate Injection Systems must be initiated. While the core is expected to remain adequately cooled as long as RPV water level remains above the Minimum Steam Cooling RPV Water Level (-191 in.), permitting the core to become partially uncovered when it can be prevented is undesirable unless specific benefits can be gained. If available injection sources have not been sufficient to reverse the decreasing RPV water level trend before level drops to the top of the active fuel, it is unlikely that they will be sufficient after level drops *below* the top of the active fuel. Delaying use of the Alternate Injection Systems and permitting water level to drop below the top of the active fuel are therefore unwarranted. In addition, Alternate Injection System pumps must be started to support evaluations of available makeup capacity required in subsequent steps.

Examination Outline Cross Reference	Level	SRO
295038 High Off-Site Release Rate 2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	Tier	1
	Group #	1
	K/A	295038 2.4.4
	Rating	4.7
	Revision	0
Revision Statement:		

Question: 81

The Plant was shut down due to indications of fuel failure.

RCIC system experienced a steam leak with a failure to isolate.

All RCIC Max Safe Values have been met.

Which of the following would require the CRS to direct an Emergency Depressurization?

A. FHA or FPS exhaust radiation levels above trip setpoints

B. SGTS FLTR TRN radiation level above the operating limit

C. Dose Calculation calls for a declaration of a General Emergency

D. MSL radiation level is more than 3 times full power background

Answer: C

Explanation:

Per EP-4 step 11 states IF a discharge cannot be isolated Declaration of a General Emergency is expected due to radioactivity release dose calculation, THEN, Emergency Depressurization is required.

'C' is correct.

Distracters:

'A' is wrong - is plausible this is used in a retention step for EP-4 step 1. IF FHA / FPS exhaust radiation level is above 3.6 / 30 mr/hr, THEN Verify SGTS initiation and isolation.

‘B’ is wrong - is plausible because this would cause a ED however, it requires the Max Safe Value to be exceeded before an ED is required not just exceeding the operating limit.

‘D’ is wrong – is plausible this is used in a retention step for EP-4 step 1. MSL radiation level is more that 3 times full power background, THEN, Close the MSIVs and MSL drain valves.

K/A Match

The candidate must have the knowledge to recognize system operating parameters that are entry-level conditions for emergency procedures.

Technical References:

02-S-01-40, EP Bases, Rev. 9

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-EP4

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(4)	
Level of Difficulty:	3.0	

PRA Applicability:

RCIC is listed as #17 on the System Importance to CDF.

SRO JUSTIFICATION

**Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)**

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?
→ Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?
→ Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?
→ Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
→ Yes → RO question

No ↓

Does the question require one or more of the following:

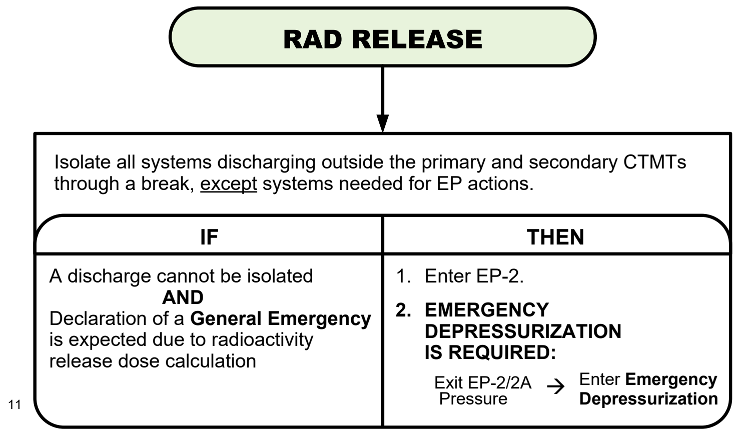
- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

→ Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Step 11



Discussion

Isolate all systems discharging outside the primary and secondary CTMTs

Primary systems comprise the pipes, valves, and other equipment connected directly to the RPV such that a reduction in RPV pressure will decrease the flow of steam or water being discharged through an unisolated break in the system.

Any primary system discharging into the environment should be isolated, except for systems that must be operated to accomplish the objectives of other EPs. Operation of these systems takes precedence over release rate concerns and is addressed in other procedures. Systems needed for other EP steps may include those being used to shut down the reactor, cool the core, depressurize the RPV, or vent the primary containment.

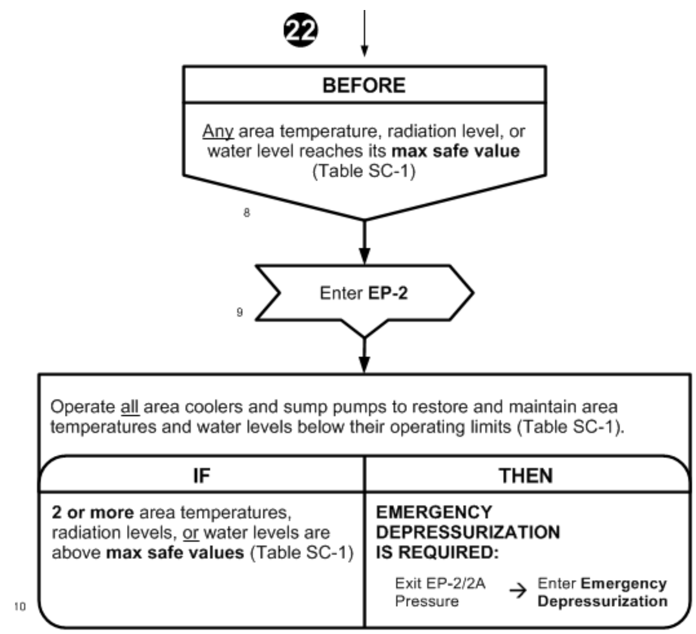
A discharge cannot be isolated

An offsite release rate above the General Emergency level is an indication of degrading conditions and presents a more immediate threat to the health and safety of the public. If a primary system is discharging outside the primary and secondary containments and it is expected that the release rate will reach the General Emergency level, emergency depressurization is performed to reduce the discharge rate. Before the depressurization is initiated, EP-2 is entered and the reactor is scrammed. A scram is performed to reduce the rate of energy production and thus the rate of radioactivity release. The scram is prescribed indirectly, through entry of EP-2, because:

EP-2 coordinates control of RPV water level and pressure following the scram.

EP-2 branches to EP-2A, *ATWS RPV Control*, to provide appropriate guidance if all rods are not inserted by the scram.

Steps 8 through 10



Discussion

BEFORE Any area temperature, radiation level, or water level reaches its max safe value

The “max safe value” of a parameter is defined to be the highest value at which:

- Equipment necessary for the safe shutdown of the plant will operate, and
- Personnel can perform any actions necessary for the safe shutdown of the plant.

If a parameter exceeds its max safe value, plant safety may be jeopardized. If a primary system is known to be discharging into the secondary containment but it cannot be isolated, EP-2 should be entered and the reactor scrammed “before” any max safe temperature, radiation level, or water level is reached.

The scram is performed to reduce the rate of energy production and thus the heat input, radioactivity release, and break flow into the secondary containment. The scram is prescribed indirectly, through entry of EP-2, because:

- EP-2 coordinates control of RPV water level and pressure following the scram.
- EP-2 branches to EP-2A, *ATWS RPV Control*, to provide appropriate guidance if all rods are not inserted by the scram.
- An override in the Pressure branch EP-2 allows rapid depressurization through the main turbine bypass valves in anticipation of a possible emergency depressurization.

Step 1

MSL Radiation Alarms

MSL A / MSL D HI HI	P601-18A-C4
MSL B / MSL C HI HI	P601-19A-C4

While in this procedure:

IF	THEN
FHA exhaust radiation level is <u>above</u> **3.6 mr/hr** **OR** FPS exhaust radiation level is <u>above</u> **30 mr/hr**	• Verify SGTS initiation. • Verify isolation of FHA, FPS, and Aux Building HVAC.
FHA and FPS HVAC are isolated **AND** No FHA or FPS HVAC high radiation isolation has occurred	1. Secure SGTS. 2. Restore FHA, FPS, and Aux Bldg HVAC. OK to defeat isolations...................... **Att 31**
MSL radiation level is more than 3 times full power background	Close the MSIVs and MSL drain valves.

Discussion

FHA HVAC exhaust radiation level is above 3.6 mr/hr OR FPS exhaust radiation level is above 30 mr/hr

If FHA HVAC exhaust radiation level exceeds 3.6 mr/hr or FPS exhaust radiation level exceeds 30 mr/hr, FHA, FPS, and Aux Building HVAC should isolate and SGTS should automatically start. These actions should be verified to limit radioactivity release to the environment and maintain the desired negative differential pressure. If either action did not occur automatically, it should be initiated manually.

SGTS is the normal mechanism employed under post-transient conditions to maintain secondary pressure negative with respect to the atmosphere. The exhaust from SGTS is processed and directed to an elevated release point before being discharged to the environment.

FHA and FPS are isolated AND No FHA or FPS HVAC high radiation isolation has occurred

FHA and FPS also automatically isolate on high drywell pressure and low RPV water level signals. These isolations are anticipatory, designed to prevent release of radioactivity following a primary system break. If radiation levels are not high, however, the isolations may be defeated to permit restarting HVAC. Running normal HVAC will provide more air flow than can be obtained using SGTS.

MSL radiation level is more than 3 times full power background

A main steam line radiation level of 3 times full power background is considered an indication of possible fuel failure and requires isolation of the main steam lines. Since the high main steam line radiation isolation logic has been removed at GGNS, the MSIVs and MSL drains must be closed

Examination Outline Cross Reference	Level	SRO
600000 Plant Fire On Site 2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	Tier	1
	Group #	1
	K/A	600000 2.4.34
	Rating	4.7
	Revision	0

Revision Statement:

Changed stem #1 to ensure clarity of SRO directions

Changed stem #4 to ensure clarity of SRO directions.

Question: 82

Fire in the control room

Which of the following actions should the SRO direct to be performed for this event?

1. Dispatch operators to Alternate Shutdown Panels
2. Depressurize the RPV
3. Manually start Div 1 D/G and energize 15AA
4. Perform Upper Containment Pool lineup to RCIC Suction

ONLY:

A. 1, 3 and 4

B. 1, 2 and 4

C. 1, 2 and 3

D. 2, 3 and 4

Answer: **C**

Explanation:

Per 05-1-02-II-1, Shutdown from the Remote Shutdown panel ONEP, the following actions are time critical action for a fire in the control room.

IF a Main Control Room fire is in progress, the Reactor Operator dispatched to perform Attachment XX Should complete the actions to manually start Div I Diesel Generator (**IF NOT** auto started)

within 8 minutes of reactor being scrammed **AND** Should complete actions to Close the Div I Diesel Generator output breaker within 12 minutes of reactor being scrammed.

IF a Main Control Room fire is in progress, the CRS **AND** FSS (Remote Shutdown Panel) to perform Attachment XXI will complete the actions to Defeat LSS **AND** open the 15 Bus breakers within 8 minutes of reactor being scrammed.

IF a Main Control Room fire is in progress, the CRS **AND** FSS at the Remote Shutdown Panel to perform Attachment XXI will complete the actions to open the SRVs to depressurize the Reactor within 12 minutes.

IF a Main Control Room fire is in progress, the CRS **AND** FSS at the Remote Shutdown Panel to perform Attachment XXI will complete the actions to depressurize **AND** inject RHR A to the reactor within a maximum of 14.3 minutes.

Per step 3.1.3 "actions prior to leaving the control room does not include RCIC.

'C' is correct.

Distracters:

'A' is wrong - Depressurize the RPV is also required.

'B' is wrong – Initiating RCIC is not required prior to leaving the Control Room during a Fire but it is required if evacuating for any other reasons.

'D' is wrong – This action is from 05-S-01-FSG-002, Alternate RCIC Suction Source, that is to be used during an extended loss of AC power, not from the remote shutdown panel.

K/A Match

The candidate must have the knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Technical References:

05-1-02-II-1, Shutdown from the Remote Shutdown Panel ONEP, Rev. 51

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	

	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

1.13 Actions are listed in order of assumed importance and normally Should be performed in order **OR** May be performed concurrently depending on actual plant conditions. Subsequent action May be performed out-of-sequence at the discretion of an On-Shift SRO.

1.14 EC 24584 **AND** Safe Shutdown Analysis (SSA) credit the use of Emergency Procedures 05-S-01-EP-1 **AND** 05-S-01-EP-2 for failure to SCRAM during a control room fire.

1.15 Most Division I components are Safe Shutdown components. Those that are Safe Shutdown are indicated by “SSD” designation as they Can be isolated from Control room in event of a Control Room fire.

1.16 A scram report per Ops Section Guidelines OPG-37 is **NOT** required **IF** a Main Control Room Fire is in progress.

1.17 The actions listed below are actions that will be completed within the times listed to be successful in depressurizing **AND** injecting RHR A to the vessel within maximum 14.3 minutes based on feasibility study **AND** timed walk through of this procedure. Operators are to proceed briskly to complete the actions to provide as much added margin as possible to meet the maximum 14.3 minute time to inject RHR A to the reactor.

1.17.1 **IF** a Main Control Room fire is in progress, the Reactor Operator dispatched to perform Attachment XX Should complete the actions to manually start Div I Diesel Generator (**IF** **NOT** auto started) within 8 minutes of reactor being scrammed **AND** Should complete actions to Close the Div I Diesel Generator output breaker within 12 minutes of reactor being scrammed.

1.17.2 **IF** a Main Control Room fire is in progress, the CRS **AND** FSS (Remote Shutdown Panel) to perform Attachment XXI will complete the actions to Defeat LSS **AND** open the 15 Bus breakers within 8 minutes of reactor being scrammed.

1.17.3 **IF** a Main Control Room fire is in progress, the CRS **AND** FSS at the Remote Shutdown Panel to perform Attachment XXI will complete the actions to open the SRVs to depressurize the Reactor within 12 minutes.

1.17.4 **IF** a Main Control Room fire is in progress, the CRS **AND** FSS at the Remote Shutdown Panel to perform Attachment XXI will complete the actions to depressurize **AND** inject RHR A to the reactor within a maximum of 14.3 minutes.

3.0 SUBSEQUENT OPERATOR ACTIONS

NOTE

Section 3.1 applies to Fires in the Control Room Only **AND** Steps 3.1.1 through 3.1.6 should be performed in parallel.

Digital capable radios pre-set on channel one are primary means of communications. They are located in dedicated charging station in Control Room **AND** designated as Emergency Use Only.

RSDP Locker has required keys on colored key chains hung on inside of drawer.

A controlled copy of this procedure is located in panel 1H22-P298, P295 **AND** P296 for NOB Operator.

Operator designations listed in this section are recommended. Shift Manager/ CRS may designate personal as desired.

WHEN shutdown is controlled from RSP, habitability may be challenged due to limited ventilation through adjacent switchgear rooms. Available heat stress countermeasures should be implemented as necessary per EN-IS-108.

3.1 **IF** decision is made by the Shift Manager/Control Room Supervisor (CRS) to evacuate due to a Control Room fire, **THEN** **PERFORM** the following:

3.1.1 **DISPATCH** a Reactor Operator (with digital capable radio) to **PERFORM** Attachment XX (DG1 & 15AA actions. (Hard Card Available)

3.1.2 **DISPATCH** the Safe Shutdown Operator (with flashlight **AND** digital capable radio) (Reactor Operator **OR** NOB) to the Auxiliary Building Alternate Shutdown Panels to **PERFORM** Attachment IV (Controlled Copy of this ONEP is located in the panels).

3.1.3 **PERFORM** the following actions prior to leaving the Control Room:

a. **PLACE** Reactor Mode switch in SHUTDOWN position.

b. **ENSURE** **ALL** Control Rods are fully inserted.

c. **CLOSE** Inboard **AND** Outboard MSIVs.

d. **PLACE** **ALL** Division II MSRV control hand switches in the OFF position. [MSO related item.]

3.1.4 **DISPATCH** a Second Reactor Operator (with digital capable radio) to PERFORM Attachment XXII (RPS Breakers to Scram Pilot Solenoids **AND** MSIV Solenoids). (Hard Card Available)

NOTE

Section 3.2 addresses shutdown from the remote shutdown panel considering normal availability of plant equipment with **NO** Control Room fire.

The SRO manning the Remote Shutdown Panel (RSP) **AND** the Shift Manager Should obtain a radio prior to evacuating Control Room. Communications May be required (via radio, page, **OR** other means) between RSD room **AND** OSC. (Radios are primary means of communications)

The Operator performing functions in plant Should carry an emergency flashlight in case of a loss of offsite power **AND** a radio as the primary means of communication.

A controlled copy of this procedure is located in panel 1H22-P298, P295 **AND** P296 for NOB Operator.

Operator designations listed in this section are recommended. Shift Manager/ CRS May designate personal as desired.

WHEN shutdown is controlled from the RSP, habitability May be challenged due to limited ventilation through the adjacent switchgear rooms. Available heat stress countermeasures Should be implemented as necessary per EN-IS-108.

3.2 **IF** decision is made to evacuate Control Room by Shift Manager/Control Room Supervisor (CRS), **THEN**:

3.2.1 **DISPATCH** personnel as follows:

- Two **OR** more Reactor Operators to the Remote Shutdown Panels.
- The CRS **AND** Field Support Supervisor (FSS) to the Remote Shutdown Panels.
- Safe Shutdown Operator (Nuclear Operator A **OR** B) to the Auxiliary Building to man the Alternate Shutdown panels.
- The SM to the Outage Control Center (OCC) **OR** other suitable areas to initiate actions per 10-S-01-1, Activation of the Emergency Plan.
- Designated communicators to the Outage Control Center (OCC) **OR** other suitable area as desired by the SM.

3.2.2 **IF** time **AND** conditions permit, **THEN PERFORM** the following before leaving Control Room:

- **PLACE** Reactor Mode switch in SHUTDOWN position.
- **ENSURE** all Control Rods are fully inserted.
- **CLOSE** Inboard **AND** Outboard MSIVs.
- **IF** needed for Reactor level control **OR** MSIVs were closed, **THEN INITIATE** RCIC.
- **OBTAIN** radio, key to RSD room cabinet (Key # 7) **AND** LSS Panel (Key # 16, there are also LSS Panel keys in the RSD cabinet) keys. (SRO manning RSD panel).

3.2.3 **IF** Reactor **CANNOT** be scrammed **OR** MSIV's were **NOT** closed from the Control Room, **THEN PERFORM** the following:

a. To SCRAM Reactor, **OPEN** the following RPS breakers:

- CB2A (52-1C71102) Panel 1C71-P001 (A RPS Room 25A/189)
- CB8A (52-1C71108) Panel 1C71-P001 (A RPS Room 25A/189)
- CB2B (52-1C71202) Panel 1C71-P002 (B RPS Room 25A/148)
- CB8B (52-1C71208) Panel 1C71-P002 (B RPS Room 25A/148)

b. To close MSIVs **OPEN/CHECK OPEN** the following RPS breakers:

- CB5A (52-1C71105) Panel 1C71-P001 (A RPS Room 25A/189)
- CB7A (52-1C71107) Panel 1C71-P001 (A RPS Room 25A/189)
- CB5B (52-1C71205) Panel 1C71-P002 (B RPS Room 25A/148)
- CB7B (52-1C71207) Panel 1C71-P002 (B RPS Room 25A/148)

3.2.4 **BEFORE** manning Remote Shutdown Panels 1H22-P150/P151, **PERFORM** the following:

a. **OPEN** the following breakers in NSP64DP03 Power Panel (Div 2 Switchgear room north wall near corner with west wall) to defeat Switchgear Room CO2 systems;

- Breaker #1 C02 Panel N1P64D207 (Div 1)
- Breaker #2 C02 Panel N1P64-D208 (Div 2)
- Breaker #3 C02 Panel N1P64-D209 (Div 3)

b. **PLACE** the CO2 panel N1P64D212 lockout switch (HS-M142C) to ABORT (located on the red Fire Protection Panel next to RSD panel entrance door).

Examination Outline Cross Reference	Level	SRO
295015 Incomplete SCRAM / 1 AA2.01 Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM: AA2.01 Reactor power	Tier	1
	Group #	2
	K/A	295015 AA2.01
	Rating	4.3
	Revision	0
Revision Statement: **New Question.**		

Question: 83

An ATWS is in progress with reactor power at 3%.

Which of the following actions must be directed by the CRS?

1. Initiate ARI/RPT system
2. Start Suppression Pool Cooling
3. Initiate Standby Liquid Control system
4. Initiate and override Low Pressure ECCS system

A. 1 and 2 only

B. 1 and 3 only

C. 2 and 4 only

D. 3 and 4 only

Answer: D
Explanation: The SRO candidate must interpret reactor power, because, with an ATWS <5% the RO will perform '1', and '2' per immediate operator actions. All other actions must be directed by the CRS. If reactor power was >5% all actions would be performed by immediate operator actions. 'D' is correct.
Distracters: 'A' is wrong, these actions are perfomed per immediate operator actions in the SCRAM onep 'B' is wrong, #1 is an immediate action and #3 is correct

‘C’ is wrong, #2 is an immediate action and #4 is correct.

K/A Match

The candidate must have the ability to interpret the given information of reactor power then the SRO must decide what actions are required to be directed.

Technical References:

05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents
05-S-01-EP-2, RPV Control

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP2A, OBJ. 1

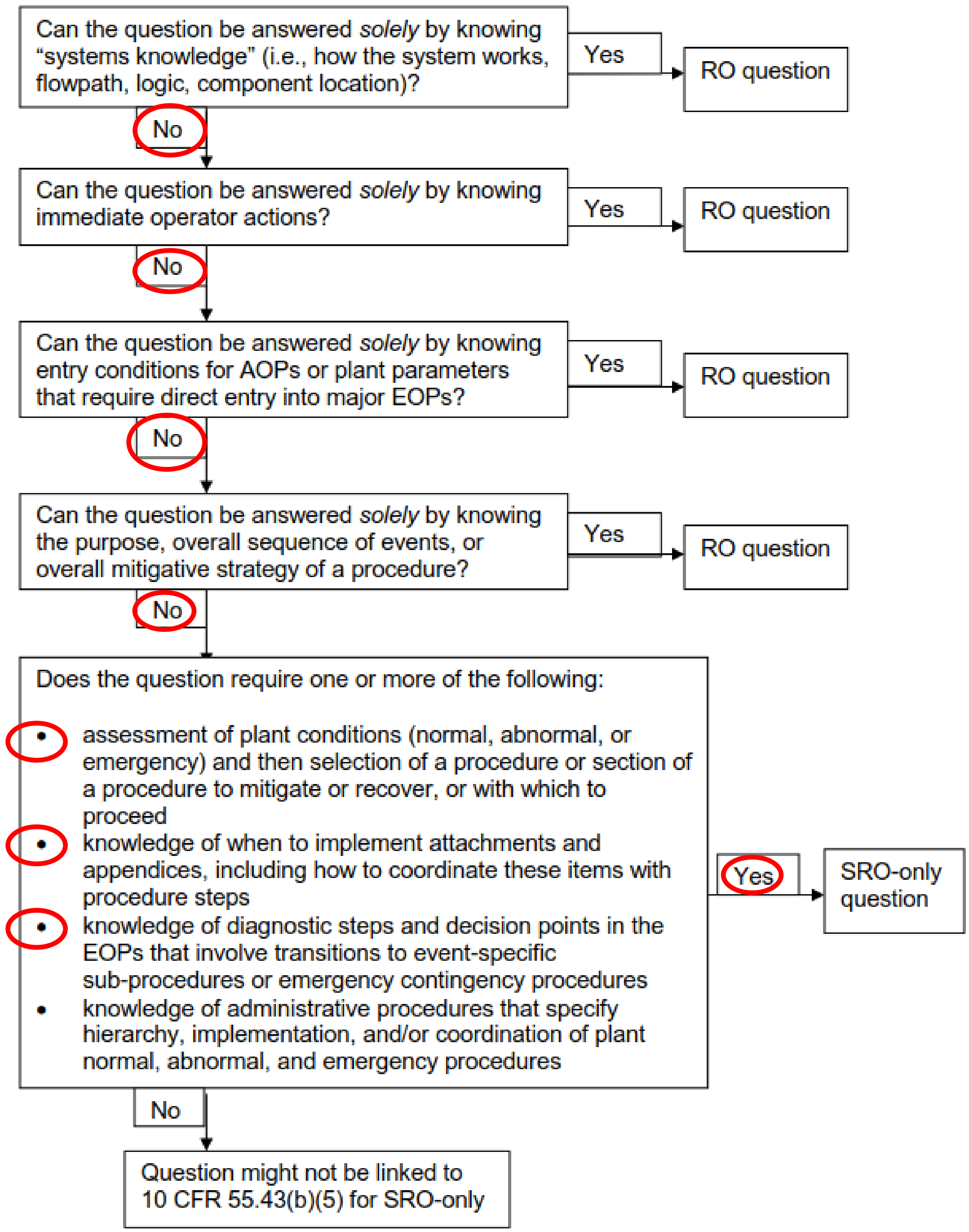
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



2.0 IMMEDIATE OPERATOR ACTIONS

2.1 **PLACE** Reactor Mode switch to **SHUTDOWN**.

2.2 **CONFIRM** all Control Rods are fully inserted.

2.3 **CONFIRM** Reactor power decreasing.

2.4 **IF** a "Failure to De-energize" ATWS occurs,
THEN ARM AND simultaneously **DEPRESS** the Manual SCRAM push buttons.

2.5 **IF** an ATWS occurs,
THEN perform the following:

2.5.1 **VERIFY** Reactor Recirc Pumps are transferred to LFMGs

2.5.2 **INITIATE** ARI/RPT

2.5.3 **INHIBIT** ADS

2.5.4 **TERMINATE AND PREVENT** HPCS

2.5.5 **IF** ATWS is greater than 5% power,
THEN PERFORM the following:

a. **INITIATE** STANDBY LIQUID CONTROL

b. **TERMINATE AND PREVENT** low pressure ECCS

c. **WHEN** directed to “Terminate and Prevent Feedwater Injection”
THEN:

1. **PUT** 1C34-LK-R600, FW LVL MASTER CONT in manual mode by depressing MAN push button.

2. **LOWER** 1C34-LK-R600, FW LVL MASTER CONT setpoint by **DEPRESSING** the OUT ⇓ pushbutton **UNTIL** OUTPUT is at -5.00 percent.

2.5.6 **MAXIMIZE** Suppression Pool Cooling.

Examination Outline Cross Reference	Level	SRO
295033 High Secondary Containment Area Radiation Levels 2.2.12 Knowledge of surveillance procedures.	Tier	1
	Group #	2
	K/A	295033 2.2.12
	Rating	4.1
	Revision	2

Revision Statement:

Rev 1, Per Ops Rep, Tech Specs now state use the Surveillance Frequency Control Program to determine surveillance frequency times, added times to stem and added 02-S-01-17 as a reference, removed reference material.

Rev 2, Validation, changed stem and gave the surveillance frequency and changed answer A to 36 hours instead of 12.

Question: 84

The plant is operating at rated power.

At 0700 on 2/1/20, while reviewing the logs, the CRS discovers the following:

The last time that a Channel Check was performed on Fuel Handling Area Ventilation radiation monitor D17-K617A was 24 hours ago.

Per the Surveillance Frequency Control Program, the surveillance frequency is 12 hours.

Per Technical Specifications, the CRS is required to:

A. Perform the surveillance no later than 1900 on 2/2/20.

B. Perform the surveillance no later than 0700 on 2/2/20.

C. Immediately declare D17-K617A INOPERABLE and enter LCO 3.3.6.2.

D. Immediately declare D17-K617A INOPERABLE and enter LCO 3.3.6.2 and perform the surveillance no later than 1900 on 2/1/20.

Answer: **B**

Explanation:

TS SR 3.0.3 allows up to 24 hours to complete a missed surveillance.

Per 02-S-01-17, 6.8.1. **IF** it is discovered that a Surveillance was **NOT** performed within its specified frequency, **THEN** per SR 3.0.3 compliance with the requirement to declare the LCO **NOT** met May be delayed, from the time of discovery, up to 24 hours **OR** up to the limit of the specified Frequency, whichever is greater. This allows time to perform the surveillance.

‘B’ is correct.

Distracters:

‘A’ is incorrect because TS SR 3.0.3 states the surveillance must be performed within the periodicity or 24 hours; whichever is longer.

‘C’ is incorrect because TS SR 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required.

‘D’ is incorrect because TS SR 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required.

K/A Match

The candidate must have knowledge of Radiation monitor surveillance procedures and Tech Specs.

Technical References:

TS 3.3.6.2 and SR 3.0.3
02-S-01-17, Control of Limiting Conditions for Operation, Rev. 134

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-TS001, OBJ. 40

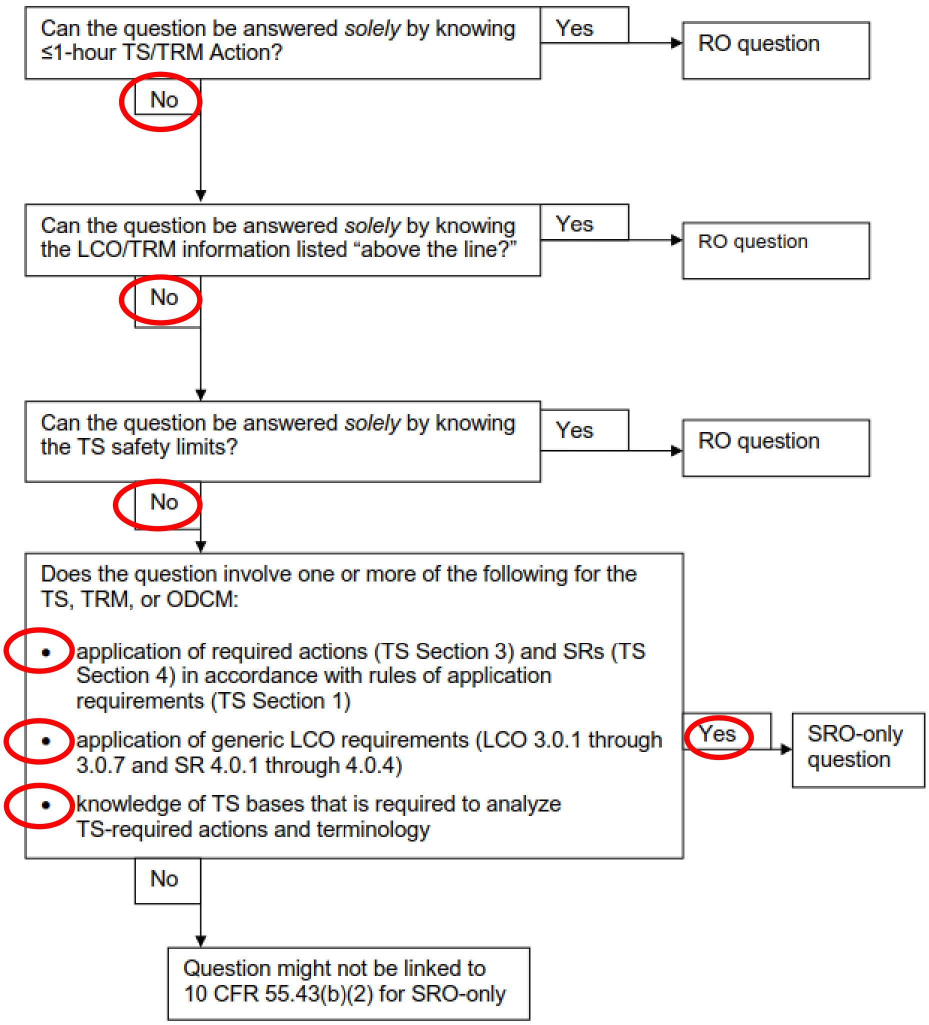
Question Source:	**Bank # 698**	2009 NRC Exam Q85
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

SRO JUSTIFICATION

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2)
(Technical Specifications)



3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

6.8 Missed Surveillance (SR 3.0.3)

6.8.1 **IF** it is discovered that a Surveillance was **NOT** performed within its specified frequency, **THEN** per SR 3.0.3 compliance with the requirement to declare the LCO **NOT** met May be delayed, from the time of discovery, up to 24 hours **OR** up to the limit of the specified Frequency, whichever is greater. This allows time to perform the surveillance.

6.8.2 A CR Must be initiated upon discovery that the Surveillance Requirement was **NOT** met.

6.8.3 SR 3.0.3 requires that a risk evaluation Shall be performed for any surveillance delayed greater than 24 hours **AND** the risk Shall be managed. The risk to be evaluated includes the effects of potential undiscovered equipment inoperability due to the missed surveillance **AND** the impact to plant operation of establishing the conditions required to perform the surveillance (which May include plant shutdown). The risk evaluation is performed as follows.

NOTE

A quantitative risk evaluation using EOOS is preferred if the affected system, structure **OR** component (SSC) is included in the plant PRA model (EOOS). **IF** the SSC is **NOT** included in EOOS, **THEN** a qualitative risk evaluation Should be performed.

a. Within 24 hours from the time of discovery perform an initial risk evaluation of potential undiscovered equipment inoperability in accordance with Reference 3.12.

(1) This initial risk evaluation informs the decision as to whether the surveillance May be delayed for greater than 24 hours **AND** the impact to normally scheduled work activities.

(2) **IF** the initial risk evaluation determines the risk impact is significant (EOOS RED), **THEN** performance of the surveillance Should **NOT** be delayed past 24 hours. Alternatively the affected equipment May be declared inoperable **AND** the appropriate LCO Conditions entered.

(3) A Potential LCOTR Should be initiated for surveillances delayed greater than 24 hours.

(4) The results of the initial risk evaluation **AND** the acceptability of delaying the surveillance by more than 24 hours is documented via either the Operability Assessment of the associated CR performed per Reference 3.9, **OR** on the Potential LCOTR initiated for the condition.

Examination Outline Cross Reference	Level	SRO
500000 High Containment Hydrogen Concentration 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	Tier	1
	Group #	2
	K/A	500000 2.1.25
	Rating	4.2
	Revision	1
Revision Statement: **Rev 1 Per NRC review (10 free view) changed wording in stem of question, all caps on MINIMUM and added the word "procedurally".**		

Question: 85

HANDOUT PROVIDED

The plant has experienced an ATWS with a steam line break in the Containment.

Current Containment pressure is 9.8 psig.

What is the MINIMUM Hydrogen concentration to procedurally require the CRS to direct Hydrogen Recombiners shutdown and Containment Spray initiated?

A. 2.9%

B. 6%

C. 7.8%

D. 9%

Answer: **C**
Explanation: Per EP-3, at 2.9% CTMT or DW hydrogen, Exit all EPs and Enter SAPs. Once in the SAPS Hydrogen leg, when Hydrogen is in the UNSAFE zone of the HDOL curve then Secure Hydrogen Recombiners and start Containment Spray. 'C' is correct.
Distracters: 'A' is incorrect This is the concentration to exit the EPs and enter the SAPs. 'B' is incorrect This is the concentration for starting the recombiners

‘D’ is incorrect This is the concentration for high limit for igniter operation.

K/A Match

The candidate has the knowledge and ability to interpret the given information and using reference materials, such as graphs to determine actions.

Technical References:

05-S-01-SAP-1, Severe Accident Procedure, Rev 10

Handouts to be provided to the Applicants during exam:

HDOL Curve

Learning Objective:

GLP-OPS-EPT19, OBJ. 2

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

HYDROGEN

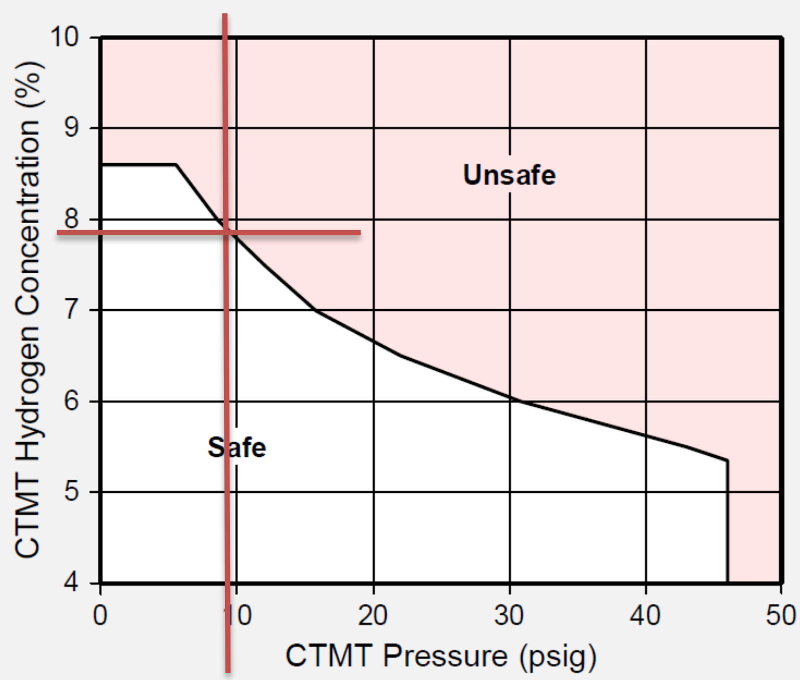
While in this section:

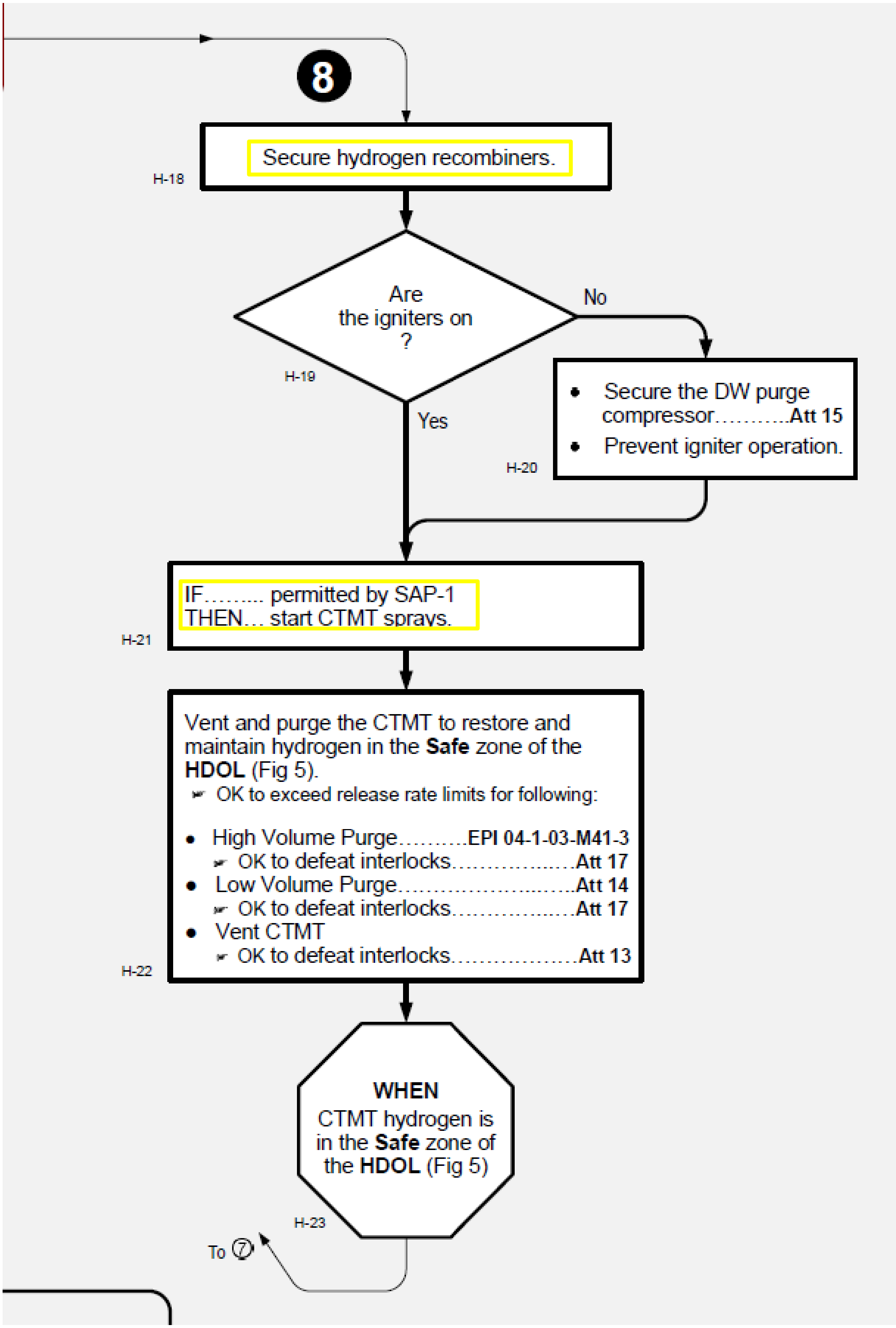
IF	THEN
The hydrogen monitoring system is unavailable	Sample the CTMT for hydrogen (CI 08-S-04-957)
Igniters are off and <u>either</u>: • <u>DW</u> hydrogen is at or above **9%** or unknown **OR** • <u>CTMT</u> hydrogen is in the **Unsafe** zone of the **HDOL** (Fig 5)	Prevent igniter operation until: • <u>DW</u> hydrogen is below **9%** **AND** • <u>CTMT</u> hydrogen is in the **Safe** zone of the **HDOL** (Fig 5).
<u>CTMT</u> hydrogen is in the **Unsafe** zone of the **HDOL** (Fig 5)	Go to ⑧

H-1

7

Fig 5
Hydrogen Deflagration Overpressure Limit (HDOL)




8
Secure hydrogen recombiners.
H-18
Are the igniters on ?
H-19
No
Yes
• Secure the DW purge compressor..........Att 15
• Prevent igniter operation.
H-20
IF........ permitted by SAP-1
THEN... start CTMT sprays.
H-21
Vent and purge the CTMT to restore and maintain hydrogen in the **Safe** zone of the **HDOL** (Fig 5).
OK to exceed release rate limits for following:
• High Volume Purge..........EPI 04-1-03-M41-3
OK to defeat interlocks.................Att 17
• Low Volume Purge.........................Att 14
OK to defeat interlocks.................Att 17
• Vent CTMT
OK to defeat interlocks.................Att 13
H-22
WHEN CTMT hydrogen is in the **Safe** zone of the **HDOL** (Fig 5)
H-23
To ⑦

Examination Outline Cross Reference	Level	SRO
211000 Standby Liquid Control System	Tier	2
	Group #	1
A2. Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A	211000 A2.03
	Rating	3.4
A2.03 A.C. power failures	Revision	0
Revision Statement:		

Question: 86

Standby Liquid Control (SLC) pump ‘B’ is tagged for Maintenance.

An ATWS has occurred with reactor power at 15%.

The RO performs immediate operator actions.

Electrical bus 15AA has lost power and indicates a fault on the bus.

(1) Which of the following describes the availability of the SLC system?

(2) What action must now be taken?

A. (1) No SLC pumps are available
(2) The CRS must call for Attachment 28

B. (1) ‘A’ SLC pump is available
(2) The CRS must call for Attachment 28

C. (1) ‘A’ SLC pump is available
(2) The CRS must now call for Attachments 18, 19, and 20 only

D. (1) No SLC pumps are available
(2) The CRS must now call for Attachments 18, 19, and 20 only

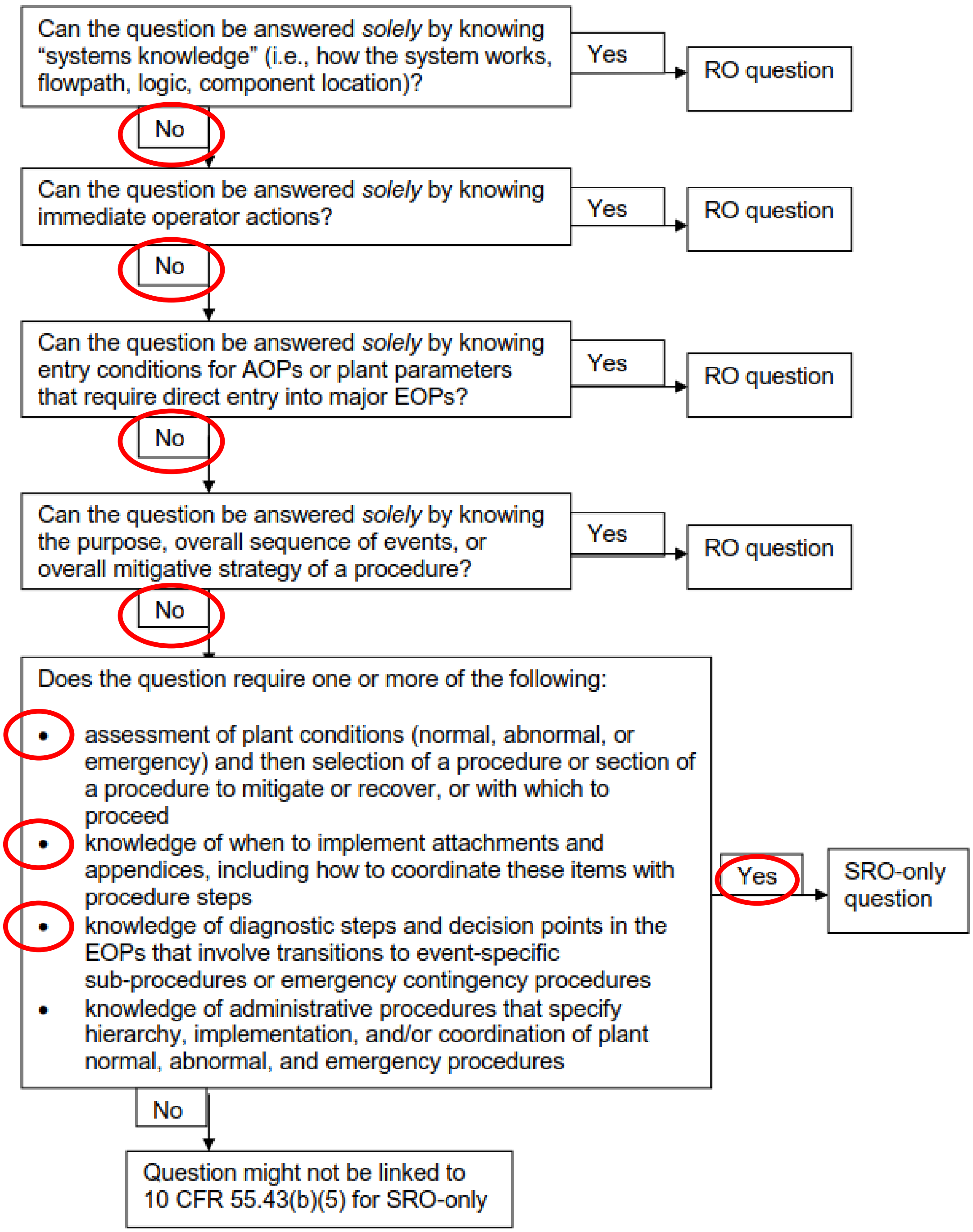
Answer: A
Explanation:

With a loss of 15AA, power is lost to SLC pump 'A' therefore no SLC pumps are available due to 'B' pump already being OOS. EP-2A calls for Attachment 28 with boron cannot be injected. 'A' is correct.
Distracters: 'B' is incorrect 'A' pump is not available. 'C' is incorrect 'A' pump is not available, these attachments will be called for but not for the loss of SLC 'D' is incorrect these attachments will be called for but not for the loss of SLC
K/A Match The candidate must predict the impact on loss of power and determine the correct procedure to use to mitigate.
Technical References: 04-1-01-1C41-1, SLC SOI, 05-1-02-EP-1, Attachment 28 02-S-01-40, EP Technical Bases
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-EP2A, OBJ. 40 GLP-OPS-C4100, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

SRO JUSTIFICATION

**Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)**



Attachment **Page 2 of 2**

Electrical Lineup Checksheet

Instruction Step 5.2.1.2, 5.5.1.2, and 5.7.1.1

COMP NO.	COMPONENT DESCRIPTION	MCC, LCC PANEL NO.	BRKR NO.	REQ'D POSITION	BRKR DEV	INITIALS 1st	INITIALS 2nd
1C41-C001A	SLC PMP A	15B21	52-152114	CLOSED			
1C41-F001A	SLC STOR TK OUTL VLV	15B21	52-152115	CLOSED			

*All Heat tracing below is retired in place

COMP NO.	COMPONENT DESCRIPTION	MCC, LCC PANEL NO.	BRKR NO.	REQ'D POSITION	BRKR DEV	INITIALS 1st	INITIALS 2nd
1H22-P110A	SLC Heat Tracing PNL	11B13	52-111316	Open			
1H22-P110B	SLC Heat Tracing PNL	12B51	52-125134	Open			
	SLC Heat Tracing PNL	1H22-P110A	Brkr #1	OFF			
	SLC Heat Tracing PNL	1H22-P110A	Brkr #4	OFF			
	SLC Heat Tracing PNL	1H22-P110A	Brkr #5	OFF			
	SLC Heat Tracing PNL	1H22-P110A	Brkr #8	OFF			
	SLC Heat Tracing PNL	1H22-P110B	Brkr #1	OFF			
	SLC Heat Tracing PNL	1H22-P110B	Brkr #4	OFF			
	SLC Heat Tracing PNL	1H22-P110B	Brkr #5	OFF			
	SLC Heat Tracing PNL	1H22-P110B	Brkr #8	OFF			

Note Exceptions: ______________________________

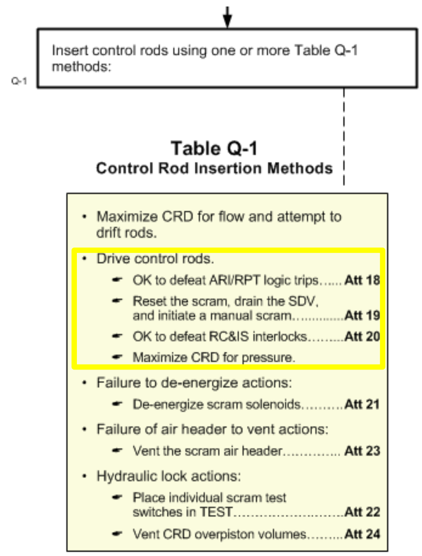
____

Performed by: ______________________________ Date ______________

Reviewed by: ______________________________ Date ______________

(Shift Supervision)

Step Q-1



Discussion

Step Q-1 identifies alternate methods of inserting control rods. These actions are performed in parallel with Steps Q-2 through Q-9, which provide directions for injecting boron into the RPV.

Shutting down the reactor using control rods is preferable to injecting boron for the following reasons:

Boron injection contaminates the primary system requiring extensive cleanup and subsequent inspection before continued plant operation is possible.

If a leak occurs below the elevation of the RPV water level being maintained, boron injection may not be successful in shutting down the reactor.

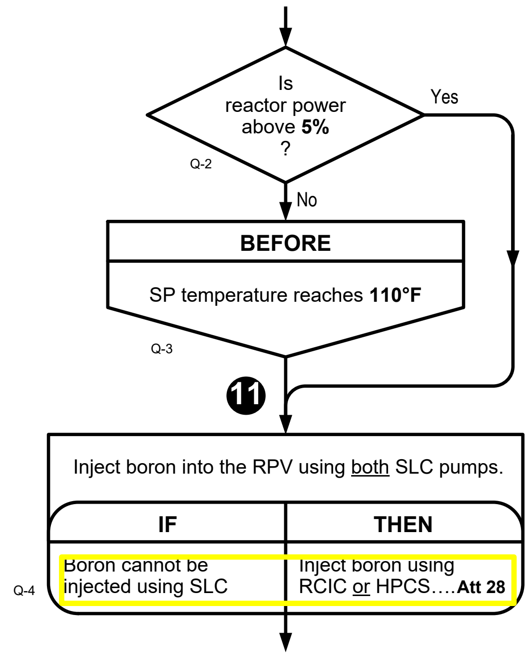
A reactor shutdown on boron is not necessarily a stable condition; if boron is subsequently diluted or displaced by a leak or an operational error, the reactor could return to criticality.

No sequence or priority is intended by the order of alternate rod insertion methods in Table Q-1; the appropriate methods and the sequence in which they are performed are dependent on event-specific conditions.

Maximize CRD for flow

Increasing CRD cooling water flow creates a higher pressure on the underside of the CRD drive piston, which can produce inward rod motion. As long as CRD hydraulics are available and the control rods are not stuck, this method can be successful even if the rods cannot be driven manually, since operation of the solenoid valves is not required. Flow is maximized by increasing the CRD flow controller to maximum and opening the pressure control valve.

Steps Q-2 through Q-4



Discussion

Steps Q-2 through Q-9 provide directions for injecting boron into the RPV. These actions are performed in parallel with Step Q-1, which identifies alternate methods of inserting control rods.

Is reactor power above 5%?

If reactor power remains above the APRM downscale setpoint (5%) following multiple attempts to scram the reactor (EP-2 Step 1, EP-2A Step 1), boron injection is initiated immediately to preclude power oscillations, avoid challenges to primary containment temperature and pressure limits, and ensure that the plant remains in a controlled state.

If reactor power is at or below 5%, thermal-hydraulic instabilities are not of significant concern since the core boiling boundary will be relatively high and the core void content will be relatively low. Some oscillations may still occur, but large-scale instabilities leading to core damage are not expected. Boron injection then need not be injected unless primary containment integrity is jeopardized.

If the MSIVs close under ATWS conditions, steam may be discharged through the SRVs, causing suppression pool temperature and primary containment pressure to increase. The requirement for boron injection when reactor power is at or below 5% is thus based upon containment temperature and pressure limits.

Examination Outline Cross Reference	Level	SRO
217000 RCIC A2. Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.01 System Initiation Signal	Tier	2
	Group #	1
	K/A	217000 A2.01
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 87

The plant is operating at rated power.

- RCIC auto initiates on an **invalid** Level 2 signal.
- Initiation logic seals in and will not reset.
- The reactor remains online.

(1) The CRS DIRECTS the RO to ___________.

(2) The CRS is directing actions from ____________.

A. (1) TRIP the RCIC Turbine
   (2) Loss of Feedwater Heating ONEP

B. (1) TRIP the RCIC Turbine
   (2) System Operating Instruction, Shutdown of the RCIC System

C. (1) Secure RCIC using SOI Hardcard
   (2) System Operating Instruction, Shutdown of the RCIC System

D. (1) Secure RCIC using SOI Hardcard
   (2) Loss of Feedwater Heating ONEP

Answer:	A
Explanation:	

With RCIC system initiation at power the SRO should enter the Loss of Feedwater Heating ONEP section ‘C’, C2 states IF RCIC initiation is NOT VALID THEN TRIP RCIC. The ONEP DOES NOT reference shutting the system using the Hardcard.

‘A’ is correct.

Distracters:

‘B’ is incorrect, but plausible due to the SOI holds all hardcards, but RCIC initiation at power is an entry into the ONEP

‘C’ is incorrect but plausible due to the SOI holds all hardcards, but RCIC initiation at power is an entry into the ONEP. If RCIC is not needed during an event the hardcard is used to shutdown but during an inadvertent initiation at power, the cold water injection downstream of the feedwater temp indicators require a more rapid shutdown

‘D’ is incorrect If RCIC is not needed during an event the hardcard is used to shutdown but during an inadvertent initiation at power, the cold water injection downstream of the feedwater temp indicators require a more rapid shutdown.

K/A Match

The candidate must have the ability to use the given information and determine what actions are to be directed and what procedure is entered.

Technical References:

05-1-02-V-5, Loss of Feedwater Heating ONEP

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

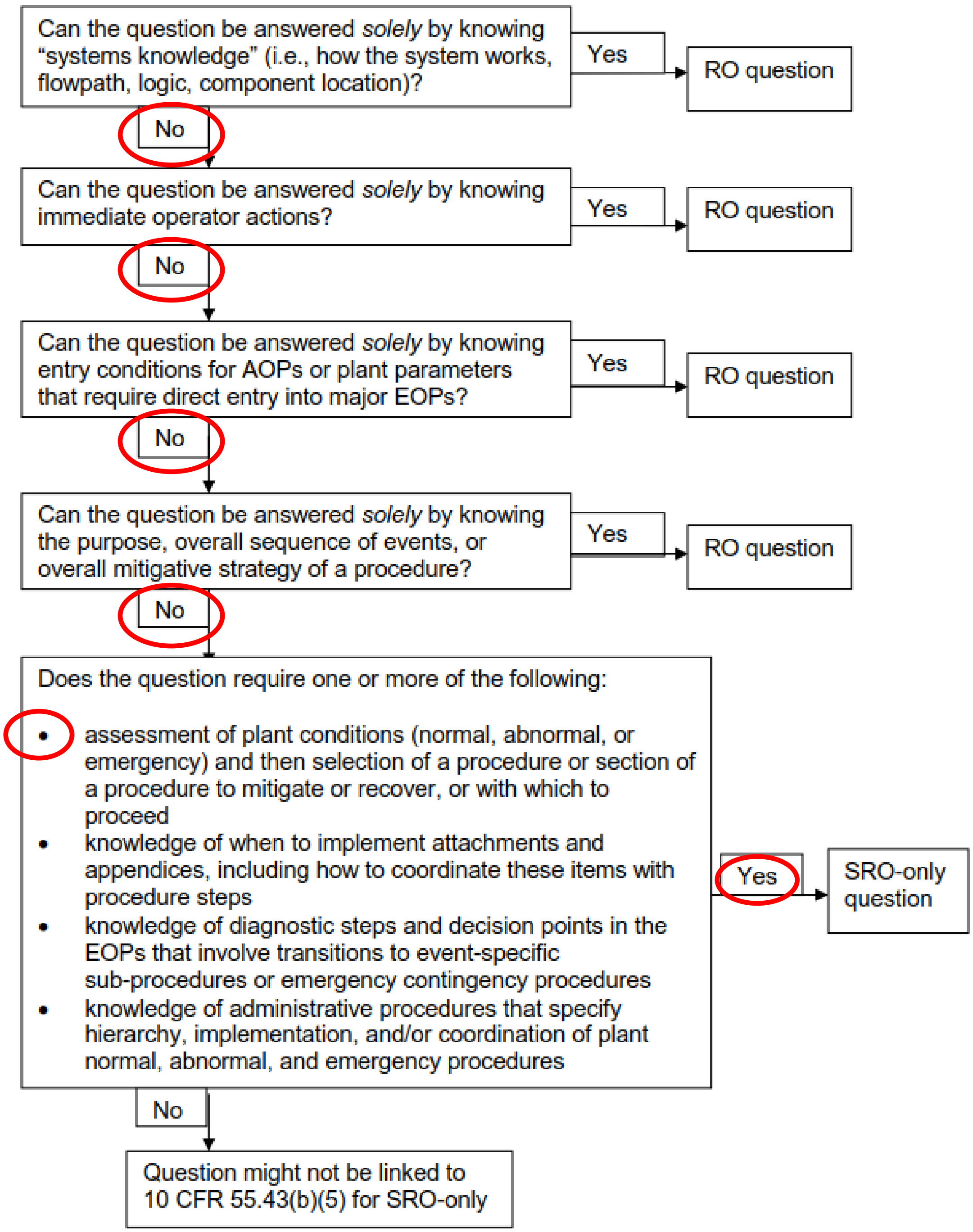
GLP-OPS-E5100 Obj 8.0 and GLP-OPS-ONEP Obj 35

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
I. RCIC Initiation in Mode 1 Entered __________ Date / Time	C1. **CHECK** by AT LEAST TWO indications that RCIC Initiation is NOT VALID. C2. **IF** RCIC initiation is NOT VALID, **THEN TRIP** RCIC. **NOTE 4** ****CAUTION 1**** C3. **IF** RCIC CANNOT be promptly SECURED, **THEN PERFORM** the following: a. **REDUCE** Core Flow to 70 Mlbm/hr, **USE** FAST Detent. b. **ENSURE** Control Rod Line < 100%, **INSERT** Control Rods, **USE** RE Instructions AND 05-1-02-III-3, Reduction In Recirculation System Flow Rate. c. **IF** Core Flow CANNOT be lowered, **THEN REDUCE** Reactor Power to < 95% APRM Power **AND ENSURE** Control Rod Line < 100%. **PERFORM** the following: 1. **INSERT** Control Rods, **USE** RE Instructions. 2. **IF** ANY Control Rod STOPS and is IMMOVABLE while traversing to its CRAM Sheet Target Position **THEN PLACE** the Reactor Mode Switch in SHUTDOWN. **NOTE 3** C4. **RECORD** occurrence of RCIC initiation on Attachment 3. **FORWARD** form to Manager, Design Engineering. **END OF SECTION**

Examination Outline Cross Reference	Level	SRO
239002 SRVs A2 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Stuck open SRV.	**Tier**	2
	Group #	1
	K/A	239002 A2.03
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 88

The plant is at 90% power following a sequence exchange.

The following occurs:

- ADS / SRV LEAK. P601-18A-G-2, and SRV/ADS VLV OPEN/DISCH LINE PRESS HI, P601-19A-A5 annunciators are received
- Generator megawatt output has lowered by approximately 75 MWe
- SRV B21-F041D has a red indication on P601 handswitch
- Suppression Pool average water temperature is currently 98°F and rising
- Suppression Pool Water Level is 18.71ft. and rising.
- Handswitches on P601 and P631 for SRV B21-F041D are taken to the CLOSE position
- I&C has been notified to pull associated fuses per ARI.

What is the <u>next</u> required action to correct, control, or mitigate the impacts of this event on the plant?

A. Lower Reactor power

B. Place the Reactor Mode Switch to SHUTDOWN

C. Place one loop of RHR in Suppression Pool Cooling

D. Place both loops of RHR in Suppression Pool Cooling

Answer:	**D**

Explanation:

With the given information the student can determine that a stuck open SRV exsist.

ONEP 05-1-02-V-21, Reactor Pressure Control Malfunctions is entered and section 3.3 refers you to the ARIs to perform actions to get the SRV closed, which are in progress from the given information.
The next action should be enter EP-3 (>95°F Suppression Pool Temp.) and start all available Suppression Pool cooling (not needed for adequate core cooling). The SRO should direct both loop of suppression pool cooling to be started per EP-3. Before SP temp reaches 110°F the SRO will enter EP-2 which requires a Mode Switch to SHUTDOWN. This is based on Tech Spec 3.6.2.1 and bases. If the actions of the ARI do not close the SRV then ONEP directs a plant shutdown.

'D' is correct.

Distracters:

'A' is incorrect – The ONEP does provide actions to lower reactor power but not in this case only if pressure control system is malfunctioning.

'B' is incorrect – IF the actions of the ARI fail and the SRO determines that Supp Pool Temp will reach 110°F then the ONEP directs a mode switch to shutdown.

'C' is incorrect – When the EPs are entered due to supp pool temp the direction is to start all available cooling systems, not just one.

K/A Match

The candidate must have the ability to predict the impact of a stuck open SRV and determine the correct procedure and action to mitigate the event.

Technical References:

05-1-02-V-21, Reactor Pressure Control Malfunctions
02-S-01-40, EP Technical Bases

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

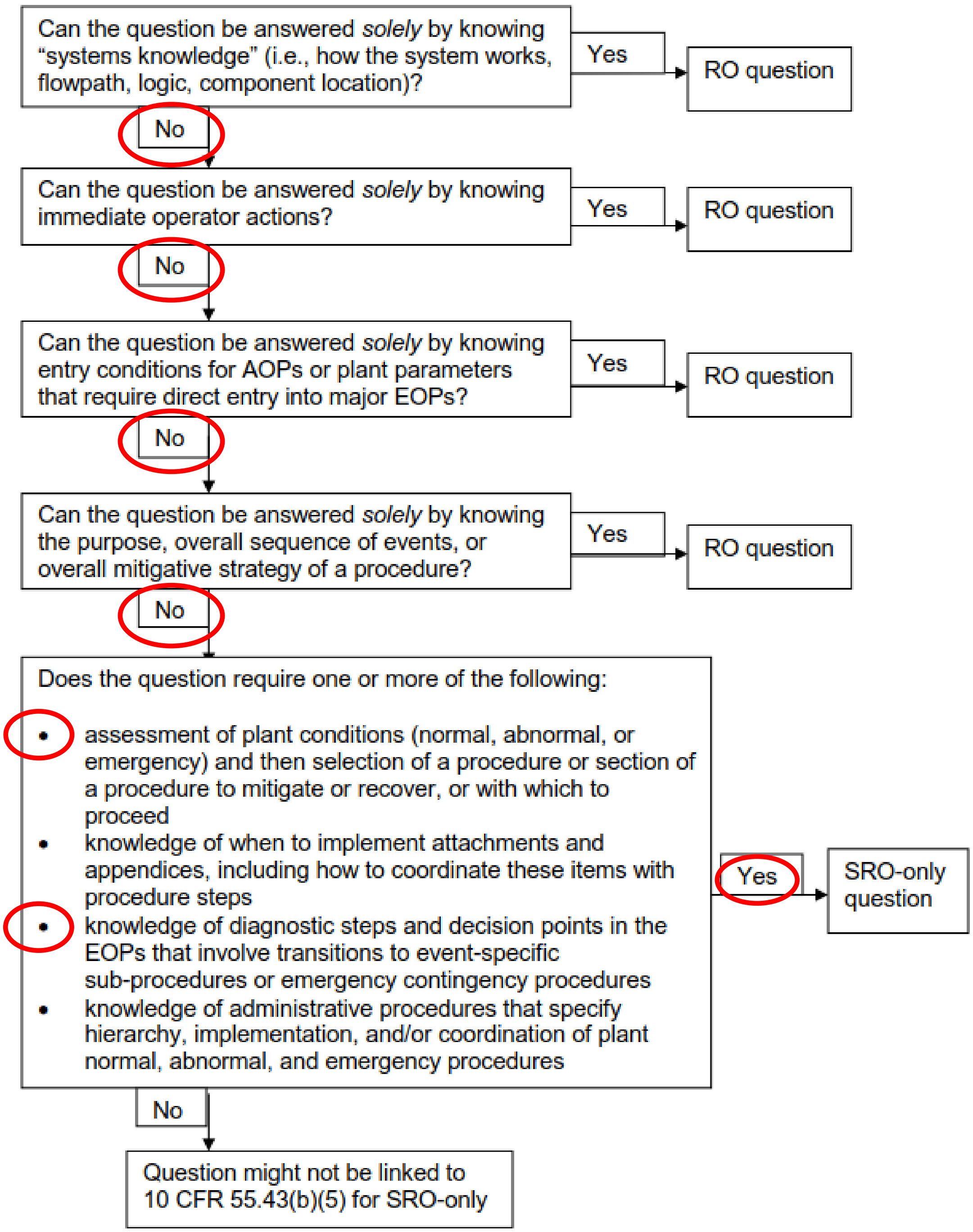
GLP-OPS-EP3, OBJ.8
GLP-OPS-ONEP, OBJ. 7

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



3.2 **IF** Reactor Scram occurs,
THEN REFER to 05-1-02-I-1, Reactor Scram ONEP

NOTE

In addition to 1H13-P601 **AND** 1H13-P631 panels, following SRVs can be operated from RSD panels 1H22-P150 **AND** 1H22-P151:

- 1B21-F051D
- 1B21-F051B
- 1B21-F047D
- 1B21-F047G
- 1B21-F051A
- 1B21-F051F

SRV position indications on 1H13-P601 panel vertical section (19B) are indicating Division II SRV solenoid valve position **AND** on 1H13-P601 panel apron section (19C) are indicating Division I pressure switch status.

3.3 **IF** one **OR** more SRVs are malfunctioning,

THEN ATTEMPT to **OPEN**/**CLOSE** SRV(s) by cycling switches on 1H13-P601, 1H13-P631, 1H22-P150, **AND**/**OR** 1H22-P151 panels as applicable.

3.3.1 **IF** SRV(s) **CANNOT** be CLOSED, **AND** Suppression Pool average temperature is approaching **OR** Could potentially exceed 110°F, **PLACE** Reactor Mode Switch in SHUTDOWN position.

3.3.2 **IF** SRV(s) **CANNOT** be OPENED, THEN **SELECT** another SRV to operate.

3.3.3 **REFER** to ARI 04-1-02-1H13-P601-19A-A5, SRV/ADS VLV OPEN/DISCH LINE PRESS HI **IF** SRV Will **NOT** CLOSE.

3.3.4 **REFER** to 05-1-02-V-5, Loss of Feedwater Heating ONEP.

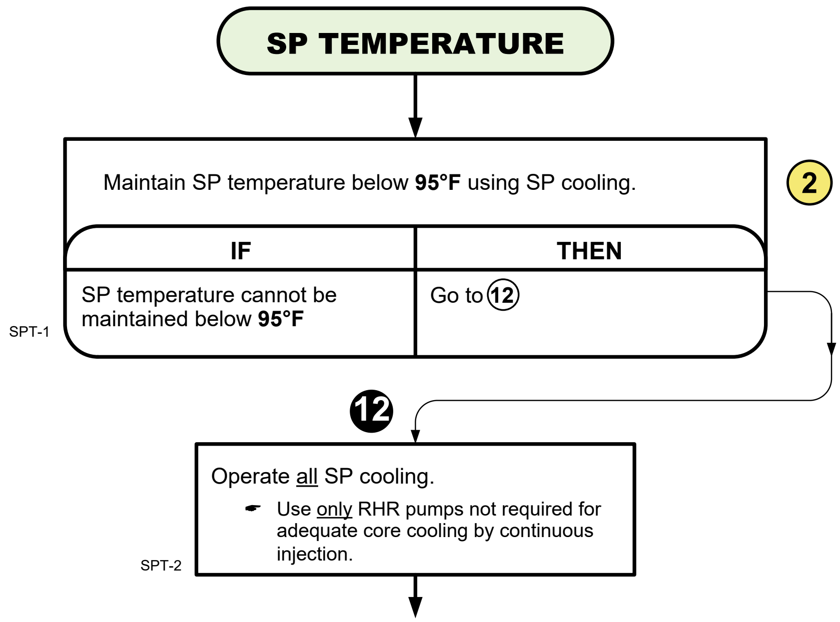
3.3.5 **REFER** to EP-3, Containment Control.

3.4 **CHECK** Main Turbine Bypass Valve positions.

3.4.1 **IF** Main Turbine Bypass Valves should be CLOSED;
THEN ENSURE Main Turbine Bypass Valves indicate CLOSED by one of following methods:

- Position indicators 1N11R602A, B, and C (1H13-P680-9D)
- Computer points N30N026A, B, and C

Steps SPT-1 and SPT-2



Discussion

Maintain SP temperature below 95°F using SP cooling

The first method used to control suppression pool temperature is that used during normal plant operations—suppression pool cooling. The first step in the SP Temperature branch thus provides a smooth transition from general plant procedures to emergency operating procedures and ensures that normal procedures are tried before more complex actions.

Operate all SP cooling

As long as suppression pool temperature remains below 95°F, the most limiting suppression pool temperature LCO, no further operator action need be taken in the SP Temperature branch. If suppression pool temperature *cannot* be maintained below 95°F, the procedure becomes more prescriptive and requires that all suppression pool cooling be used. This includes maximizing RHR flow and Standby Service Water flow through the RHR heat exchangers.

Core cooling takes precedence over suppression pool cooling in this step. Only RHR pumps not needed for core cooling may be placed in the suppression pool cooling mode. It is permissible, however, to alternate RHR pumps between modes if *continuous* injection is not required.

Examination Outline Cross Reference	Level	SRO
259002 Reactor Water Level Control System	Tier	2
	Group #	1
2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	K/A	259002 2.4.50
	Rating	4.0
	Revision	0
Revision Statement:		

Question: 89

The plant is at rated thermal power when the following alarms are received:

- RFPT A TRIP
- RX LVL 40"/32" HI/LO

The plant has the following conditions:

- Reactor power – 82%
- Core Flow - 85 mlbm/hr
- RFPT B speed – 5850 rpm
- Reactor water level is slowly lowering

(1) Given the alarms and conditions, what event has occurred?

(2) The CRS should transition to _________________ procedure that contains the actions to mitigate these conditions.

A. Complete Reactor Recirc Flow Control Valve Runback
ONEP 05-1-02-V-7, Feedwater System Malfunctions.

B. Complete Reactor Recirc Flow Control Valve Runback
ARI 04-1-02-1H13-P680-2A-A2, RFPT A TRIP.

C. Partial Reactor Recirc Flow Control Valve Runback
ARI 04-1-02-1H13-P680-2A-A2, RFPT A TRIP.

D. Partial Reactor Recirc Flow Control Valve Runback
ONEP 05-1-02-V-7, Feedwater System Malfunctions

Answer:	D

Explanation:

With the indications given the Recirc FCVs did not completely runback due to high core flow and power, Reactor power is too high for one Feedpump to handle. ONEP 05-1-02-V-7, Feedwater System Malfunctions contains the action to **ENSURE** Reactor Recirculation System Flow Control Valve runback occurs (62 Mlbm/hr). The term Ensure means if it did not occur then perform actions to make it happen.

'D' is correct.

Distracters:

'A' is incorrect – If a complete runback would have occurred then core flow would be approximately 62 Mlbm/hr.

'B' is incorrect - If a complete runback would have occurred then core flow would be approximately 62 Mlbm/hr. ARI directs to refer to the ONEP

'C' is incorrect - ARI directs to refer to the ONEP

K/A Match

The candidate uses the given alarms and parameters to verify system actions are in concurrence with the alarm response manual and transition to another procedure to mitigate the event.

Technical References:

05-1-02-V-7, Feedwater System Malfunctions ONEP, Rev. 31

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-ONEP, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		

None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
Reactor Feedwater Pump Trip Entered ____________ Date / Time	**NOTE 1** A1. **IF** Reactor Recirculation Pumps are Operating in Fast Speed, **AND** Reactor Water Level LOWERS to 32.7 in., **THEN PERFORM** the following actions: a. **ENSURE** Reactor Recirculation System Flow Control Valve runback occurs (62 Mlbm/hr). b. **ENSURE** Reactor Power is <75%, **USE** RE Instructions. ****CAUTION 1**** A2. **START** available Reactor Feed Pump as needed, **USE** SOI 04-1-01-N21-1. **END OF SECTION**

Examination Outline Cross Reference	Level	SRO
262001 A.C. Electrical Distribution 2.4.41 Knowledge of the emergency action level thresholds and classifications.	Tier	2
	Group #	1
	K/A	262001 2.4.41
	Rating	4.6
	Revision	0
Revision Statement:		

Question: 90

HANDOUT PROVIDED

From rated power a normal plant shutdown has begun with all three Emergency Diesel Generators inoperable and unavailable.

The following occurs:

- Tornado results in a loss of all offsite power
- RCIC automatically initiates and restores reactor water level to a band of -30” to +50”
- 45 minutes after the initial power loss, offsite power is restored to buses 15AA and 17AC

Bus 16AB remains de-energized.

What is the emergency classification for this event?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C
Explanation:

Per EAL SS1: Stem conditions are consistent with the SS1 declaration criteria: Loss of all offsite power and loss of all onsite AC power (i.e., no EDGs available) to Div 1, 2, & 3 ESF buses (i.e., 15AA, 16AB, 17AC) for >15 minutes.

'D' is correct.

Distracters:

Because RCIC restored level and because power was restored to at least one ESF bus (actual to two buses) within 4 hours, EAL SG1 does not apply, making 'D' plausible, but wrong.

The plausibility of distracters 'A' and 'B' are based on the SRO Applicant wrongly applying EALs SU1 or SA1 for the loss of power that occurred. SU1 is wrong because it considers a loss of power to only the Div 1 and Div 2 buses. SA1 is wrong. Although it considers a loss of power to the point of a Station Blackout (or one failure short of a Blackout), to remain at the Alert level, the condition must last no more than 15 minutes.

K/A Match

The candidate must have the knowledge of the emergency action level thresholds and classifications.

Technical References:

10-S-01-1, Section EPP 01-02

Handouts to be provided to the Applicants during exam:

EAL Emergency Classification Flowcharts

Learning Objective:

GLP-EP-EPTS6, OBJ.

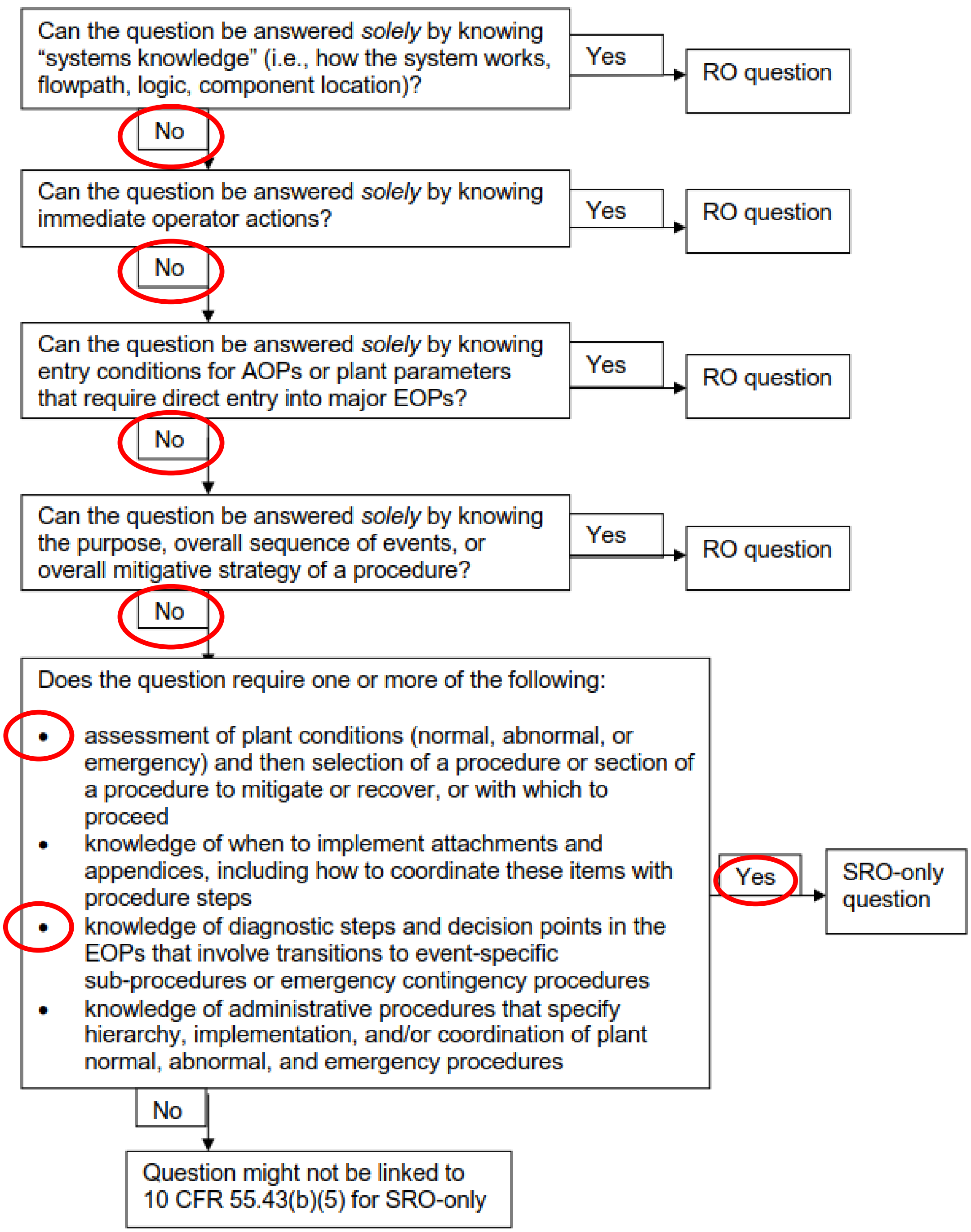
Question Source:	**Bank # 380**	2011 NRC Exam Q77
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

ESF R20 is listed as #4 on the System Importance to CDF.
EDGs are listed as #11 on the System importance to CDF.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



SS1

Initiating Condition -- SITE AREA EMERGENCY

Loss of all offsite and all onsite AC power to Div I, II & III ESF busses for $\geq$ 15 minutes.

Operating Mode Applicability:

Mode 1............. Power Operation
Mode 2............. Startup
Mode 3............. Hot Shutdown

Example Emergency Action Level(s):

Note: *The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. Loss of all offsite and all onsite AC power to Div I, II & III ESF busses for $\geq$ 15 minutes.

Basis:

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency busses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable. If this bus is the only energized bus then a Site Area Emergency per SS1 should be declared.

Escalation to General Emergency is via Fission Product Barrier Degradation (F) or IC SG1, "Prolonged loss of all offsite and all onsite AC power to Div I, II & III ESF busses.

Examination Outline Cross Reference	Level	SRO
201001 Control Rod Drive Hydraulic System A2. Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.08 Inadequate system flow	**Tier**	2
	Group #	2
	K/A	201001 A2.08
	Rating	2.8
	Revision	0
Revision Statement: **Ops review – added units in stem, changed stem to correct indication of full upscale on flow meter.**		

Question: 91

The Plant has experienced an ATWS.

The CRS calls for the Nominal Band for Reactor Water level and pressure.

Current RPV water level is -50 inches and slowly lowering.

Attachments are installed for control rod insertion and CRS has been maximized for pressure.

The RO inserting control rods reports control rod movement has stopped and CRD system flow rate is **pegged full upscale** with C11-F002A, CRD FLO CONT VLV indicating green only indication.

Which of the following describes the action to be directed by the CRS?

A. Maximize CRD for Flow

B. Restore the Aux Building

C. Call for Attachment 24, Vent CRD Overpiston Volumes, to be installed

D. Call for Attachment 22, Place Individual Scram Test Switches in TEST, to be installed

Answer:	B

Explanation:

With the given indications an Aux building/Containment isolation occurred at -41.6 inches and therefore causing instrument air to isolate to the containment. The failure of the Flow control valve is loss of air, the FCV will fail as is but drift close due to instrument air leaks. As the FCV closes control rod motion will stop. To mitigate this event the CRS must call for "Restore the Aux Building" per EP-2A and restore Air to the Aux building and Containment.

'B' is correct.

Distracters:

'A' is wrong – This action is performed during an ATWS prior to installation of Attachments 18, 19 and 20. This should cause control rods to drift in due to high cooling water pressure. This action will have no effect on the FCV due to the loss of Air.

'C' is wrong – Attachment 24 should be used when control rods can't be moved due to hydraulic block, but not in this situation, this also requires an entry into the Containment, not advised during an ATWS.

'D' is wrong - Attachment 24 should be used when control rods can't be moved due to failure to de-energize scram solenoids, but not in this situation, this also requires an entry into the Containment, not advised during an ATWS

K/A Match

The candidate must be able to predict the impacts low system flow and using procedures mitigate the event.

Technical References:

02-S-01-40, EP Technical Bases, Rev. 9
04-1-01-C11-1, CRD SOI, Rev. 155

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

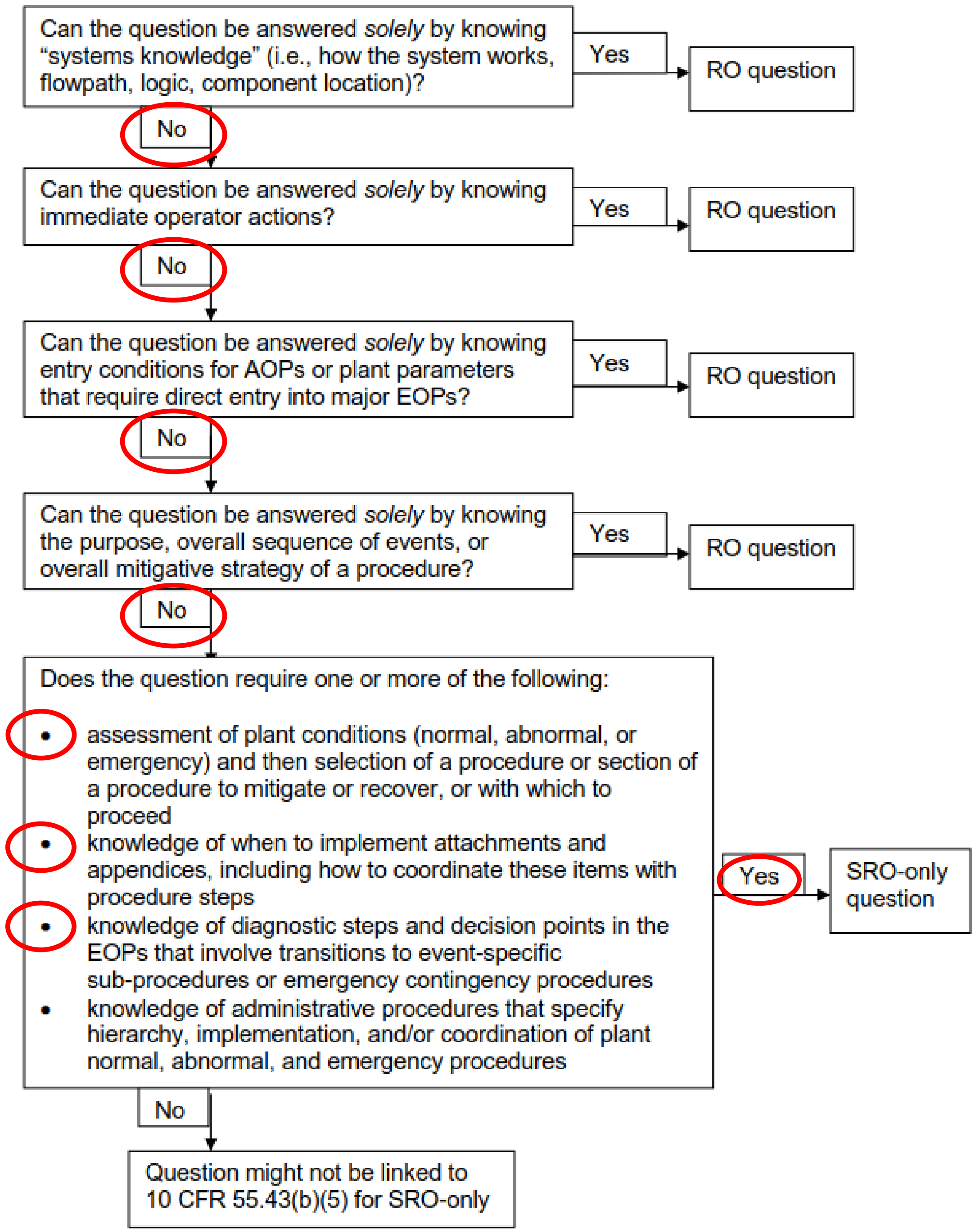
GLP-OPS-EP2A, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(6)	

Level of Difficulty:	3.0	
PRA Applicability:		
None.		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



3.0 PRECAUTIONS AND LIMITATIONS

3.1 Loss of instrument air results in a reactor scram. The Inlet **AND** Outlet Scram valves (126 **AND** 127) Will Open. The Scram Discharge Volume Vent valves (F010, F180) **AND** Drain valves (F011, F181) Will Close, **AND** the CRD Flow Control Valves FV-F002A(B) Will fail "as is" initially, But eventually close due to internal air leaks.

3.2 Loss of RPS causes Scram Pilot Solenoid valve (F417) **AND** Scram Discharge Volume Pilot Solenoid valves (F009, SV F182) to de-energize, **AND** Backup Scram Solenoid valves (F110A **AND** F110B) to energize. This condition causes a reactor scram **AND** isolation of the scram discharge volume.

3.3 Loss of RC&IS Will prevent normal movement of control rods, But Does **NOT** prevent rod insertion on a reactor scram.

3.4 In closing HCU Isolation valves, do **NOT** exceed the torque which Can be applied by manual hand operation. Excessive applied pressure May result in damage to the valve seats.

3.5 Sustained loss of cooling water to a CRD **WHEN** the reactor is at operating pressure **AND** temperature shortens the life of the CRD internal seals.

3.6 **ALWAYS** ensure CRD pumps have a minimum flow path available.

3.7 In opening HCU Isolation valves, Withdraw Riser valve (102) **AND** Scram Discharge Riser valve (112) Must be Opened before opening Insert Riser valve (101). **IF** a scram occurred with the 102 **OR** 112 valve Closed **AND** 101 valve Open, serious damage Could result to the CRD.

3.8 Cooling water flow is required to maintain CRD drive cleanliness **AND** control CRD temperature.

3.9 To prevent rod drift-in, the cooling water header pressure Should be less than 60 psid above reactor pressure. To assure normal control rod settling into a notch, the cooling water header pressure Should be less than 35 psid above reactor pressure.

3.10 Do **NOT** secure cooling water to the CRDs until reactor temperature is <250°F, unless absolutely required to perform HCU maintenance.

3.11 **WHEN** operating any of the HCU air supply valves 116XX care Should be taken **NOT** to damage the air supply tubing.

3.12 Do **NOT** insert a control rod without diagonal support in the fuel cell.

3.13 Scramming control rods Should only be done as part of testing **OR** in emergency situations.

3.14 **EITHER** isolate the Recirc Seal Purge using F026A(B) **OR** do **NOT** operate CRD pump(s), **IF EITHER** recirculation loop is completely isolated (Suction Valve B33-F023A(B) **AND** Discharge Valve B33-F067(B) Both Closed). This is to prevent pressurization of recirculation loops by CRD Hydraulic System.

Step 2

While in this procedure:

IF	THEN
All control rods are inserted to or beyond position **02**, **OR** The reactor will remain subcritical under all conditions without boron	1. Terminate boron injection. 2. Exit this procedure → Enter **EP-2**
RPV level is unknown	**FLOOD THE RPV:** Exit this procedure → Enter **EP-5A**
2 Aux Building isolates	Restore the Aux Building.

Discussion

All control rods are inserted to or beyond position 02

EP-2A is entered when one or more control rods are withdrawn past position 02 and it has not been determined that the reactor will remain shutdown under all conditions without boron. If all rods are inserted to at least position 02 or it is determined that the reactor will remain shutdown under all conditions without boron while EP-2A is in use, the condition that required entry no longer exists. Boron injection may then be terminated, if in progress, and RPV water level, RPV pressure, and reactor power controlled in accordance with EP-2, RPV Control.

Positive confirmation that the reactor will remain shutdown under all conditions is best obtained by verifying that all control rods are inserted to or beyond position 02. Position 02 is the "Maximum Subcritical Banked Withdrawal Position," defined to be the greatest banked rod position at which the reactor will remain shutdown under all conditions. The derivation of the Maximum Subcritical Banked Withdrawal Position is discussed in Section 12.0.

If one or more rods are withdrawn past position 02, other criteria may be used to determine whether the reactor will remain shutdown. Possibilities include:

Design basis shutdown margin criteria (i.e., one control rod fully withdrawn and all other control rods fully inserted).

Technical Specification requirements governing control rod position and the allowable number of inoperable control rods.

Rod pattern analyses performed by the reactor engineer.

The evaluation must address not only whether the reactor *is* shutdown, but whether the reactor will *remain* shutdown under worst-case conditions following the cooldown prescribed in the Pressure branch of EP-2A.

Discussion (continued)

The phrase “without boron” does not imply that the condition cannot be satisfied if boron has been injected, but that the negative reactivity contributed by the boron is not required. Control rod insertion alone must provide the necessary shutdown margin.

RPV level is unknown

If RPV water level is “unknown,” there is no assurance that the core is submerged. The RPV must then be flooded to ensure that the core remains adequately cooled. EP-2A is exited and control of RPV water level, RPV pressure, and reactor power transfer to EP-5A, *ATWS RPV Flooding*.

“Unknown” in this step means that the value of RPV water level cannot be determined relative to a prescribed limit or action level. (Refer to the definition of “unknown” in Section 3.0.)

Aux Building isolates

Conditions requiring entry of the EPs may result in isolation of the Auxiliary Building. The direction to “restore the Aux Building” ensures that instrument air is available to support the RPV water level and RPV pressure control actions required by EP-2A.

Step Q-1

Q-1 Insert control rods using one or more Table Q-1 methods:

Table Q-1
Control Rod Insertion Methods

- Maximize CRD for flow and attempt to drift rods.
- Drive control rods.
  - OK to defeat ARI/RPT logic trips...... **Att 18**
  - Reset the scram, drain the SDV, and initiate a manual scram............. **Att 19**
  - OK to defeat RC&IS interlocks......... **Att 20**
  - Maximize CRD for pressure.
- Failure to de-energize actions:
  - De-energize scram solenoids......... **Att 21**
- Failure of air header to vent actions:
  - Vent the scram air header............. **Att 23**
- Hydraulic lock actions:
  - Place individual scram test switches in TEST......................... **Att 22**
  - Vent CRD overpiston volumes......... **Att 24**

Discussion

Step Q-1 identifies alternate methods of inserting control rods. These actions are performed in parallel with Steps Q-2 through Q-9, which provide directions for injecting boron into the RPV.

Shutting down the reactor using control rods is preferable to injecting boron for the following reasons:

Boron injection contaminates the primary system requiring extensive cleanup and subsequent inspection before continued plant operation is possible.

If a leak occurs below the elevation of the RPV water level being maintained, boron injection may not be successful in shutting down the reactor.

A reactor shutdown on boron is not necessarily a stable condition; if boron is subsequently diluted or displaced by a leak or an operational error, the reactor could return to criticality.

No sequence or priority is intended by the order of alternate rod insertion methods in Table Q-1; the appropriate methods and the sequence in which they are performed are dependent on event-specific conditions.

Maximize CRD for flow

Increasing CRD cooling water flow creates a higher pressure on the underside of the CRD drive piston, which can produce inward rod motion. As long as CRD hydraulics are available and the control rods are not stuck, this method can be successful even if the rods cannot be driven manually, since operation of the solenoid valves is not required. Flow is maximized by increasing the CRD flow controller to maximum and opening the pressure control valve.

Examination Outline Cross Reference	Level	SRO
223001 Primary Containment System and Auxiliaries 2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	Tier	2
	Group #	2
	K/A	223001 2.4.35
	Rating	4.0
	Revision	0
Revision Statement:		

Question: 92

The plant is operating at rated conditions.

An inadvertent initiation of Division 1 ECCS occurs.

The initiating signal will not clear.

Actual Drywell Pressure is rising.

(1) Which of the following will the CRS direct to secure the ‘A’ Drywell Purge Compressors?

(2) What is the availability of the ‘A’ Drywell Purge Compressor?

A. (1) EOP Attachment 15, Defeating Drywell Purge Compressors Start Signals
(2) Purge compressor will start using control room handswitch

B. (1) EOP Attachment 15, Defeating Drywell Purge Compressors Start Signals
(2) Purge compressor will not start from any remote location

C. (1) Combustible Gas Control SOI, Recovery from Manual or Automatic Initiation
(2) Purge compressor will not start from any remote location

D. (1) Combustible Gas Control SOI, Recovery from Manual or Automatic Initiation
(2) Purge compressor will start using control room handswitch

Answer:	**C**
Explanation:	

An initiation signal that will not reset has a hard start signal on the Drywell Purge Compressor. The only way to stop this compressor to **locally** trip the breaker and prevent it from reclosing. This action is performed in the Combustible Gas Control System SOI, Recovery from Manual or Automatic Initiation.

'C' is correct.

Distracters:

'A' is wrong – This action is performing an EOP Attachment which can only be done when the EOP have been entered, current conditions have no EOP entry conditions. Purge compressor will not start except for **local** breaker manipulation.

'B' is wrong – This action is performing an EOP Attachment which can only be done when the EOP have been entered, current conditions have no EOP entry conditions. .

'D' is wrong - . Purge compressor will not start except for **local** breaker manipulation.

K/A Match

The candidate must have the knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Technical References:

05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Rev. 38
04-1-01-E61-1, CGCS SOI, Rev. 41

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

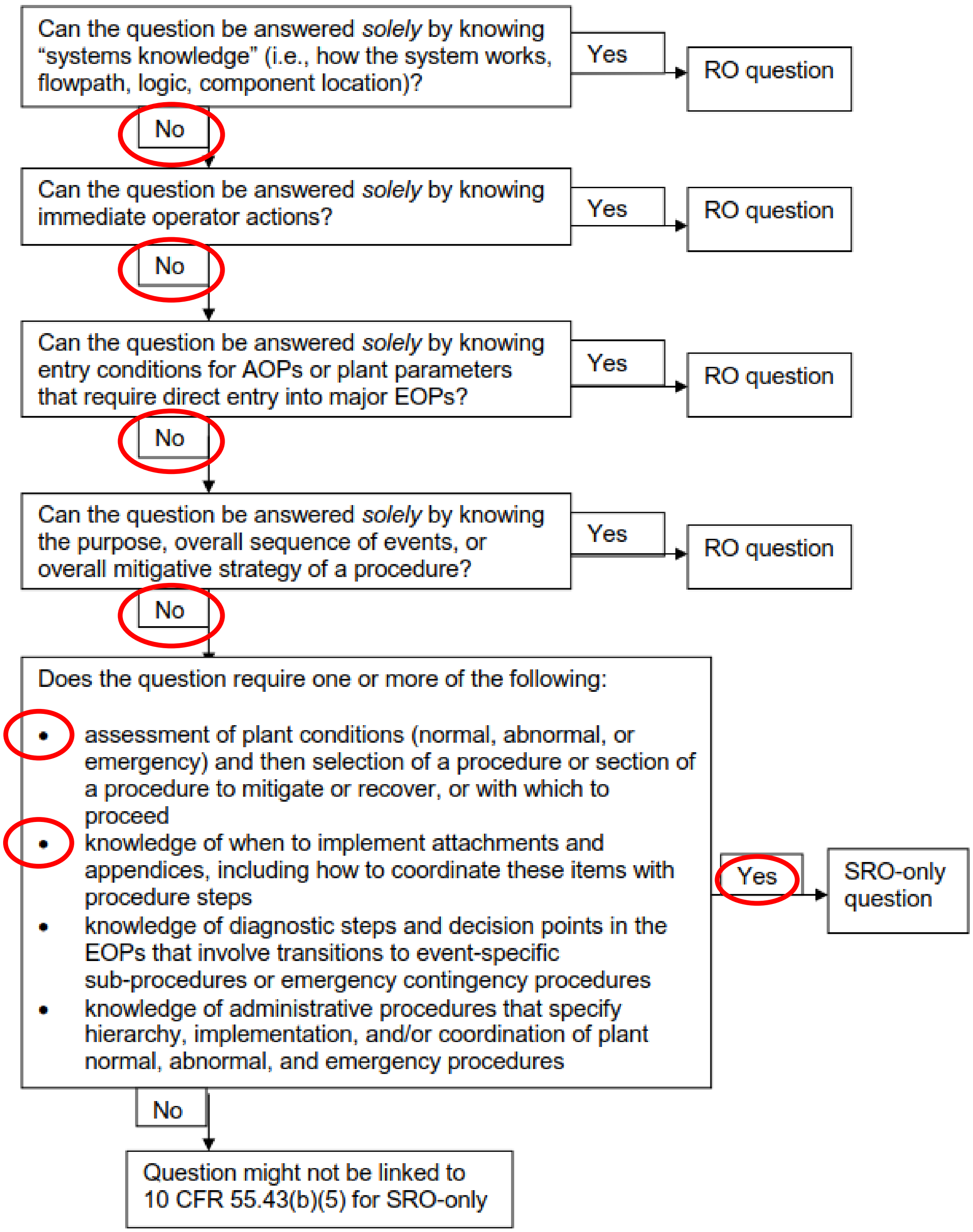
GLP-OPS-EP2, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		

None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



5.2 Recovery from Manual or Automatic Initiation

5.2.1 Prerequisites

a. Emergency Plant Manager **OR** Shift Manager has authorized shutdown of Drywell Purge Compressors

b. Drywell hydrogen concentrations are within the guidelines established in EP-3.

5.2.2 Instructions

a. **IF** possible, **STOP** D/W Purge Comp A by

(1) **DEPRESSING** the manual trip pushbutton on breaker 52-15105

AND

(2) **PULLING** lockout tab.

b. **IF** possible, **STOP** Drywell Purge Compressor B by

(1) **DEPRESSING** the manual trip pushbutton on breaker 52-16204

AND

(2) **PULLING** lockout tab.

c. **IF** required, **RESET** LSS logic per 04-1-01-R21-1.

d. **WHEN** possible, **RESET** the Combustible Gas Control A **AND** B Logic Systems by **PLACING** CGCS Div 1 **AND** 2 MAN INIT RESET keylock switches to RESET.

e. **STOP** the Drywell Purge Compressor A(B) using DRWL PURGE COMPR A(B) HS (1E61-HS-M610A(B).

(1) **VERIFY** "DRWL PURGE DIV 1(2) OPER" (1H13-P870-4A(10A)-A2) annunciator clears.

f. **CLOSE** E61 F003A(B).

g. **WHEN** desired **SHUT DOWN** H2 analyzers, H2 recombiners **AND/OR** H2 igniters per Sections 5.3, 5.4 and 5.5.

1.0 PURPOSE

1.1 Provide various attachments for support of actions directed by Emergency Procedures **AND** Severe Accident Procedures.

1.2 Provide cautions for Emergency Procedures and Severe Accident Procedures.

1.3 Provide figures for various graphs to support Emergency Procedures **AND** Severe Accident Procedures.

1.4 Revision 17 reactivated this procedure due to Emergency/Severe Accident Procedure program upgrade. The previous title of this procedure was "Level Control".

2.0 ENTRY CONDITIONS

2.1 There are **NO** entry conditions for Emergency/Severe Accident Procedure attachments, cautions, or figures. Use of these documents is directed by specific steps in Emergency Procedure **OR** Severe Accident Procedure flowcharts.

2.2 The attachments in this procedure specifies symptomatic operator actions which Will support maintaining the Reactor in a safe condition **AND** optimize plant response and margin to safety irrespective of the initiating event. Actions specified in any other procedures intended for use with this procedure Must **NOT** contradict **OR** subvert actions specified in this procedure **AND** Must **NOT** result in loss **OR** unavailability of equipment specified for use in this procedure.

3.0 OPERATOR ACTIONS

3.1 **PERFORM** individual attachments as directed by Emergency Procedure or Severe Accident Procedure flowcharts.

3.2 **REFER** to Cautions and Figures as directed by Emergency Procedure **OR** Severe Accident Procedure flowcharts.

DEFEATING DRYWELL PURGE COMPRESSORS START SIGNALS

1.0 PURPOSE

1.1 To provide instructions for defeating Drywell Purge Compressors start signals **AND** allow securing Drywell Purge Compressors with a start signal present.

2.0 INSTRUCTIONS

INITIAL / DATE

2.1 **STOP** D/W PURGE COMP A by **DEPRESSING** the manual trip pushbutton on breaker 52-15105 **AND PULLING** lockout tab. LCC 15BA1 (Area 9-119) ______ / ______

2.2 **STOP** D/W PURGE COMP B by **DEPRESSING** the manual trip pushbutton on breaker 52-16204 **AND PULLING** lockout tab. LCC 16BB2 (Area 8-139) ______ / ______

2.3 **WHEN** it is no longer necessary to defeat Drywell Purge Compressors start signals, **THEN** **AUTHORIZE** to restore system to normal per SOI 04-1-01-E61-1. ______ / ______
Shift Supv

2.4 **CLOSE** BRKR 52-15105 at LCC 15BA1 (Area 9-119). ______ / ______

______ / ______
Indep Verif

2.5 **CLOSE** BRKR 52-16204 at LCC 16BB2 (Area 8-139) ______ / ______

______ / ______
Indep Verif

2.6 **SEND** completed form to Assistant Operations Manager, Support for processing as a QA record.

3.0 REFERENCES

3.1 Schematic Diagram, Combustible Gas Control System, E-1186

Examination Outline Cross Reference	Level	SRO
259001 Reactor Feedwater System A2. Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Loss of condensate pump(s)	**Tier**	2
	Group #	2
	K/A	259001 A2.03
	Rating	3.6
	Revision	2
Revision Statement: **Rev 1 Per NRC review (10 free view) changed wording of entire question.** **Rev 2, changed wording of question #2 to clarify.**		

Question: 93

The plant is operating at 65% power for a sequence exchange.

RCIC system is INOP

The following alarm is received:

- CNDSR HTWL LVL LO, P680-2A-E9

ATC operator reports normal level in the Hotwell.

The CRS directs the crew to manually scram the reactor, enters EP-2 and Reactor Scram ONEP.

(1) What level band should the CRS direct to the crew?

(2) What procedure should be transitioned to by the CRS to restore Condensate/Fedwater for normal level control?

A. (1) +11.4 inches to +53.5 inches
(2) CNDSR HTWL LVL LO, P680-2A-E9 Alarm Response Instruction

B. (1. -30 inches to +50 inches
(2) CNDSR HTWL LVL LO, P680-2A-E9 Alarm Response Instruction

C. (1) +11.4 inches to +53.5 inches
(2) 05-1-02-V-7, Feedwater System Malfunctions ONEP

D. (1) -30 inches to +50 inches
   (2) 05-1-02-V-7, Feedwater System Malfunctions ONEP

Answer: B

Explanation:

The alarm given will cause all Condensate pumps to trip at the same time.

RPV water level will lower to the scram setpoint and both Feedpumps will trip.

Per 02-S-01-43, Transient Mitigation Strategy, step 6.6.6.d Level Control table, the Widened (Expanded) ECCS OR SRVs (non-ATWS) -30 inches to +50 inches should be used due to RCIC is INOP and ECCS (HPCS) will be used to control level.

To restore normal level control the ARI should be used First, with normal level indicated, the CRS can have the relay for low condenser hotwell level be removed and the Condensate System be restored.

'B' is correct.

Distracters:

'A' is wrong – the level band given is only used during normal situations, in this case ECCS system is being used for level control, therefore the expanded band should be used.

'C' is wrong – the level band given is only used during normal situations, in this case ECCS system is being used for level control, therefore the expanded band should be used and the actions to remove the relay is in the ARI not the ONEP.

'D' is wrong - the actions to remove the relay is in the ARI not the ONEP.

K/A Match

The candidate must have the knowledge to predict the impacts of a loss of condensate pump on the Feedwater system and determine the correct procedure to mitigate the event.

Technical References:

04-1-02-1H13-P680-2A-E9, CONDSR HTWL LVL LO, ARI Rev. 218
05-1-02-V-7, Feedwater System Malfunctions ONEP
02-S-01-43, Transient Mitigation Strategy

Handouts to be provided to the Applicants during exam:

NONE

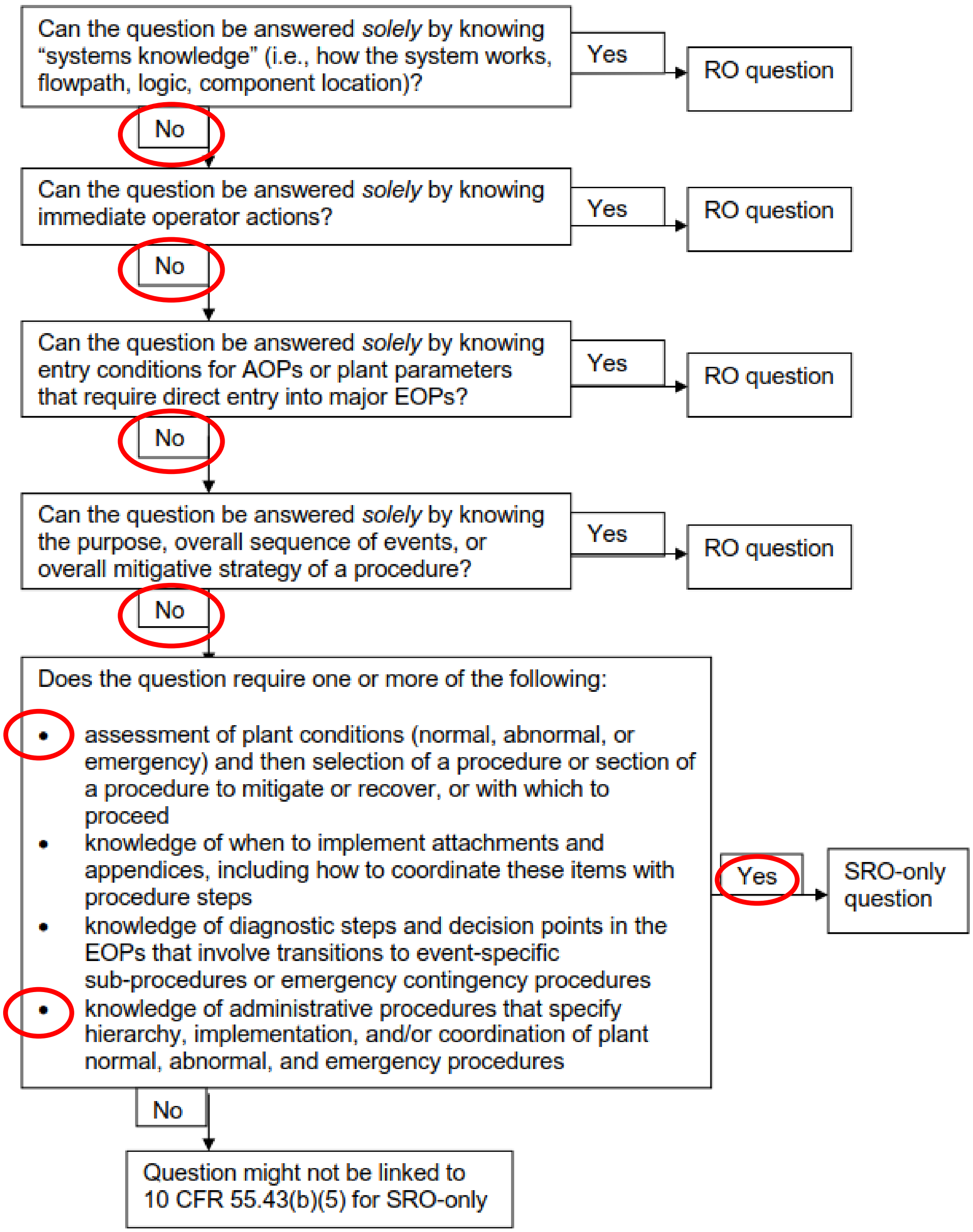
Learning Objective:

GLP-OPS-ONEP, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability: Condensate is listed as #10 on the System Importance to CDF.		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



GRAND GULF NUCLEAR STATION ALARM RESPONSE INSTRUCTION

CONDSR HTWL LVL LO

04-1-02-1H13-P680-2A-E9	
Revision: 218	Page 1 of 1

Non-Safety Related

Alarm Device 1N19-LAL-L613

1.0 POSSIBLE CAUSES

1.1 IP Condenser Hotwell Level < 1'-11" H_2O as indicated by **ANY** one (LSL-N105, LSL-N106, LSL-N107). (< 0 inch H_2O as indicated by LI-R606B.

2.0 AUTOMATIC ACTION

2.1 Condensate Pump A, B, **AND** C Trip. (**WHEN** two of three level switches trip **OR** failure of relay N19 63X-1/N105).

2.2 Condensate Booster Pumps A, B, **AND** C Trip after **ALL** three condensate pumps Trip.

3.0 IMMEDIATE OPERATOR ACTION

3.1 **REFER** to ONEP 05-1-02-V-7, Feedwater System Malfunctions.

4.0 SUBSEQUENT OPERATOR ACTION

4.1 **IF** reactor scrams, **REFER** to ONEP 05-1-02-I-1, Reactor Scram.

4.2 **IDENTIFY AND CORRECT** cause of low hotwell level as soon as possible to provide feed flow to the vessel.

4.3 **IF** desired by Shift Supervision, **REMOVE** relay N19 63X-1/N105 to defeat the Condenser Hotwell Low Level trips.
(1H13-P849 Bay C, third row of relays from the top, sixth relay from the left).

6.6.6 (Cont)

c. A target band as allowed by EP-2/2A may be given even **IF** present operation is **NOT** within the target band.

Operator **DOES NOT** have to report outside of band, but should report **IF** he anticipates target **CAN** **NOT** be attained/maintained.

This **DOES NOT** mean desired/target band has to be changed; acceptable expanded control band should be assigned **UNTIL** target band can be maintained.

d. Following are recommended nominal bands **AND** applicable control/instrument ranges for given conditions:

Level Control

Nominal level bands provided are recommended bands based on assumed plant conditions. Should these bands **NOT** be practical based on actual plant conditions,
THEN Control Room Supervision should modify bands according to steps provided in EOPs.

Condition	Nominal Level Band	Instrument Range
Normal	+11.4 inches to +53.5 inches	• P680 Narrow Range **OR** • P601 Wide Range
Widened (Expanded) ECCS **OR** SRVs (non-ATWS)	-30 inches to +50 inches	• Wide Range
ATWS with Initial Reactor Power less than or equal to 5 percent.	+11.4 inches to +53.5 inches **OR** +15 inches to +50 inches	• P680 Narrow Range **OR** • P601 Wide Range
ATWS with initial reactor power greater than 5 percent **AND** Suppression Pool less than 110F	-70 inches to -130 inches	• Wide Range **OR** • Density Compensated Fuel Zone Computer Point
ATWS with initial reactor power greater than 5 percent **AND** Suppression Pool greater than **OR** equal to 110°F	Level Which Lowered to -191 inches	• Density Compensated Fuel Zone Computer Point **OR** • Figure 1 Density Compensated Fuel Zone Nomograph

Examination Outline Cross Reference	Level	SRO
2.1.2 Knowledge of operator responsibilities during all modes of plant operation.	Tier	3
	Group #	
	K/A	2.1.2
	Rating	4.4
	Revision	2

Revision Statement:

Rev 1 Per NRC review (10 free view) changed stem to state "IAW with procedure" and changed the answers to ensure all numbers are used the same amount.

Rev 2 Per Ops Rep, changed wording of #4, changed stem to state station procedures instead of a specific one.

Rev 3 Per Ops review, added "initial" to #2 due to the acting ERO Emergency Director is not active licensed.

Question: 94

IAW with station procedures, which of the following can ONLY be a responsibility of an active licensed SRO?

1. Ensures shift manning is met
2. Assume the initial duties of the Emergency Director
3. Required to be present in the Main Control Room in Mode 4.
4. Concurs with the release and closure of outage and system work windows

A. 1, 2 and 3 only

B. 1, 3 and 4 only

C. 2, 3 and 4 only

D. 1, 2 and 4 only

Answer: **D**

Explanation:

#1 Per the procedure EN-OP-115-03, step 5.2.1 "In the case of illness or unexpected absence, SM **ENSURE** a shift member is held over and replacement personnel arranged to restore the shift complement within two hours."

#2 EN-OP-115 states the Shift Manager, "Assumes the role of Emergency Director until relieved by the Emergency Response Organization IAW Station Emergency Plan" and the CRS Assume the responsibilities and authority of the SM if the SM becomes incapacitated. Immediately notify Operations Management.

#4 EN-OP-115 states the Shift Manger Concurs with the release and closure of outage and system work windows that have an impact on the shutdown safety functions (SOER 09-01)."

'D' is correct.

Distracters:

#3 – A Senior Reactor Operator shall be present at GGNS at all times when fuel is in the reactor. A Senior Reactor Operator, normally the SM, will be present in the Control Room when the reactor is in Plant Mode 1, 2, or 3, or during emergencies. During emergencies, the SRO directing activities in the Control Room will be the Control Room Supervisor (WMC SRO prohibited).

'A' is wrong – #4 is also a SRO responsibility per EN-OP-115.

'B' is wrong – #3 is not the SROs responsibility and #1 and #2 are.

'C' is wrong - #3 is not the SROs responsibility and #1 is.

K/A Match

The candidate must have the knowledge of the responsibilities of the SRO.

Technical References:

EN-OP-115, Conduct of Operations, Rev. 26
EN-OP-115-02, Control Room Conduct and Access Controls, Rev. 6
EN-OP-115-03, Shift Turnover and Relief, Rev 004

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2.0	

PRA Applicability:
None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

5.2 WATCHSTATION RELIEF

1. In the case of illness or unexpected absence, SM **ENSURE** a shift member is held over and replacement personnel arranged to restore the shift complement within two hours.

2. In the case where a control room operator needs to be relieved during their shift, **PERFORM** the following:

   a. **OBTAIN** permission from the SM or CRS.

   b. **CONDUCT** a verbal turnover with a qualified individual including applicable items from the list below:

   - Current Plant Status (including anticipated changes in reactivity status)
   - Significant changes that occurred since the beginning of shift
   - Procedures or evolutions in progress including continuous action steps
   - Parameters being monitored and the frequency of monitoring by the person exiting the Control Room
   - Abnormal system lineups or equipment out of normal alignment
   - Defeated interlocks
   - Existing or potential plant or equipment problems
   - Issues or concerns outside of normal conditions
   - Equipment problems the relief operator needs to be aware of
   - Compensatory actions in effect that may need to be performed

   c. **NOTIFY** the SRO with Command and Control of the relief change.

   d. **DOCUMENT** the watchstander has been relieved of his duties in the Station Narrative Log .

3. Unless an emergency situation exists, off-going operators do **NOT** leave the work area until they are satisfied their relief has assumed the shift by successfully assuming the watch station using the eSOMS Station Narrative Log roster watch relief function. (Step 5.1.10 contains instructions for field turnovers conducted to support ongoing work/evolutions.)

GRAND GULF PLANT SPECIFIC ADDENDUM

Shift Manning

Each on-duty Operations shift shall be composed of at least the minimum shift composition shown in Administrative Controls Section 5.2.1 of GGNS Technical Specifications, TRM Section 7.0, including TRM Table 7.2.2-1 of GGNS Technical Requirements Manual, and Table 5.1 of GGNS Emergency Plan. Additionally the following requirements shall be met:

Except for the SM, the shift composition may be one less than the minimum requirements for a period of time not to exceed two hours to accommodate unexpected absence of on-duty shift members, including the Radiation Control/Chemistry section representative, provided immediate action is taken to restore the shift composition to within the minimum requirements. This provision is not applicable during declared emergencies and does not permit shift composition to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

If the SM becomes unavailable to perform his duties, the Control Room Supervisor (WMC SRO prohibited) assumes the SM position until relieved of duties.

The Senior Reactor Operator shown as part of the minimum on-duty Operations shift during COLD SHUTDOWN and REFUELING conditions shall not be the Senior Reactor Operator responsible for supervising core alterations.

A Senior Health Physicist shall be onsite when fuel is in the reactor(s). This position may be vacant for not more than two hours, in order to provide for unexpected absence, provided immediate action is taken to fill the position.

(4.7 Continued)

m. Perform plant inspections and observe plant personnel during the conduct of their work to identify and correct problems involving personnel performance, policies and procedures, housekeeping, material condition and personnel hazards. Reinforce management expectations.

n. Direct other departments to support plant operation as required.

o. Ensure surveillance tests are scheduled when operator distraction due to other tasks is minimized. Ensure an adequate number of personnel are available to support testing. If required, call out additional personnel or suspend activities which interfere with the test.

p. Promote teamwork and use effective communications while supervising the Control Room team and during interactions with all personnel who support plant operations. (SOER 10-2)

q. Initiate communications with offsite personnel during plant emergencies IAW Station Emergency Plan.

r. Assumes the role of Emergency Director until relieved by the Emergency Response Organization IAW Station Emergency Plan.

8. Control Room Supervisor (CRS)

a. The CRS or SRO designee (ex: Rx Management SRO) directs and monitors Control Room operators during all plant evolutions to assure equipment is operated IAW fleet standards and approved plant procedures.

- Coordinates equipment manipulations performed by reactor operators.
- Provides direct oversight of operator switch manipulations for in-service equipment in the ATC and surveillance area of the Control Room, except as prescribed for operator rounds and annunciator checks.
- Ensures no distraction exist which adversely impacts monitoring or oversight of Control Room operators conducting equipment manipulations.
- EN-OP-200 Plant Transient Response Rules establishes requirements for SRO oversight during plant transients and emergency operations.

b. Assume the responsibilities and authority of the SM if the SM becomes incapacitated. Immediately notify Operations Management.

c. Accept signature authority for those tasks delegated by the SM, except where specifically prevented by other procedures.

d. Assist in on shift training of Operations Department personnel.

e. Conduct On the Job Training (OJT) and Task Performance Evaluation (TPE).

GRAND GULF PLANT SPECIFIC ADDENDUM

5. A Senior Reactor Operator shall be present at GGNS at all times when fuel is in the reactor. A Senior Reactor Operator, normally the SM, will be present in the Control Room when the reactor is in Plant Mode 1, 2, or 3, or during emergencies. During emergencies, the SRO directing activities in the Control Room will be the Control Room Supervisor (WMC SRO prohibited).

   a. When fuel is in the reactor, a Reactor Operator or Senior Reactor Operator shall be present in the Control Room at all times.

   b. At least two licensed operators shall be present in the Control Room during reactor STARTUP, scheduled reactor SHUTDOWN, and during recovery from Reactor Scrams with one licensed operator monitoring important parameters such as reactor level, pressure and power during these evolutions.

   c. A non-licensed operator shall be on site when fuel is in the reactor and an additional non-licensed operator shall be on site while the unit is in Mode 1, 2 or 3.

   d. The on-shift composition shall include two Control Room Communicators who cannot be the plant response NOB and only one Communicator may be a member of the fire brigade. (One Communicator is normally covered by a Radwaste Operator.)

   e. When an individual is about to assume a new duty (RO, SRO or SM) in the Control Room, that individual should attend simulator re-qualification training with his permanently assigned shift. This may not always be feasible and alternative shift orientation (i.e. parallel watch standing) may be used as deemed appropriate by Operations Management. Careful planning should be done when changing shift assignments of active license holders so that a nucleus of experienced personnel is maintained.

7. Manager - Shift (SM)

   a. Provide effective oversight of activities supporting complex and infrequently performed plant evolutions such as plant heat up, startup, shutdown, cooldown and refueling. Provide oversight during the conduct of all plant operations. (SOER 09-1Rec 3)

   b. Maintain oversight of Control Room activities and remain in a supervisory role particularly during abnormal and emergency operations (INPO IER-L1-13-10). Avoid involvement in any single operation in times of emergency (SOER 96-01).

   c. Remain within 10 minutes of the Control Room and immediately return to the Control Room and provide oversight of activities during accident or abnormal conditions.

   d. Make conservative decisions as the senior manager on shift, with protection of the health and safety of plant personnel and the public being of highest priority (SOER 94-01).

   e. Ensure conservative actions are taken during unusual conditions that could jeopardize the reactor core when dealing with reactivity control, core cooling, or heat sink availability (SOER 94-01).

   f. In the case of illness or unexpected absence, SM should hold a shift member over or arrange for replacement personnel to restore the shift complement within two hours.

   g. During periods such as shift turnover briefings, SM should ensure one operator is continuously monitoring Control Room panels, alarms, and key parameters to ensure that attentiveness is maintained and huddling of the entire shift crew at a particular panel, to the detriment of safe operations, is avoided.

   h. Make conservative decisions when administering activities dealing with conditions hazardous to personnel (such as work in high radiation areas, confined spaces, working on high voltage electrical equipment and movement of heavy loads) (SOER 94-01).

   i. Make conservative decisions when dealing with primary system and containment integrity (SOER 94-01).

   j. Maintain overall responsibility for control of the key shutdown safety functions. Activities with the potential to challenge decay heat removal, lower reactor coolant system inventory, Spent Fuel Cooling and inventory make up, result in a loss of electrical power or affect reactivity, such as fuel or control rod movement, are overseen by the Shift Manager. (SOER 09-1 Rec 3)

   k. Concurs with the release and closure of outage and system work windows that have an impact on the shutdown safety functions (SOER 09-01).

Drives the ownership of training for their respective crew

Examination Outline Cross Reference	Level	SRO
2.1.37 Knowledge of procedures, guidelines, or limitations associated with reactivity management.	Tier	3
	Group #	
	K/A	2.1.37
	Rating	4.6
	Revision	0
Revision Statement: **NEW QUESTION.**		

Question: 95

Per EN-OP-115-14, Reactivity Management, which of the following would require dedicated SRO oversight other than the CRS with no concurrent duties?

A. Reducing power using Recirc flow to 85% for surveillance test.

B. Performing a sequence exchange after lowering power to 65%.

C. Performing 06-OP-C11-M-0001, Control Rod Operability Surveillance.

D. Raising power using Recirc flow from 80% to rated conditions at a ramp rate given by the RE.

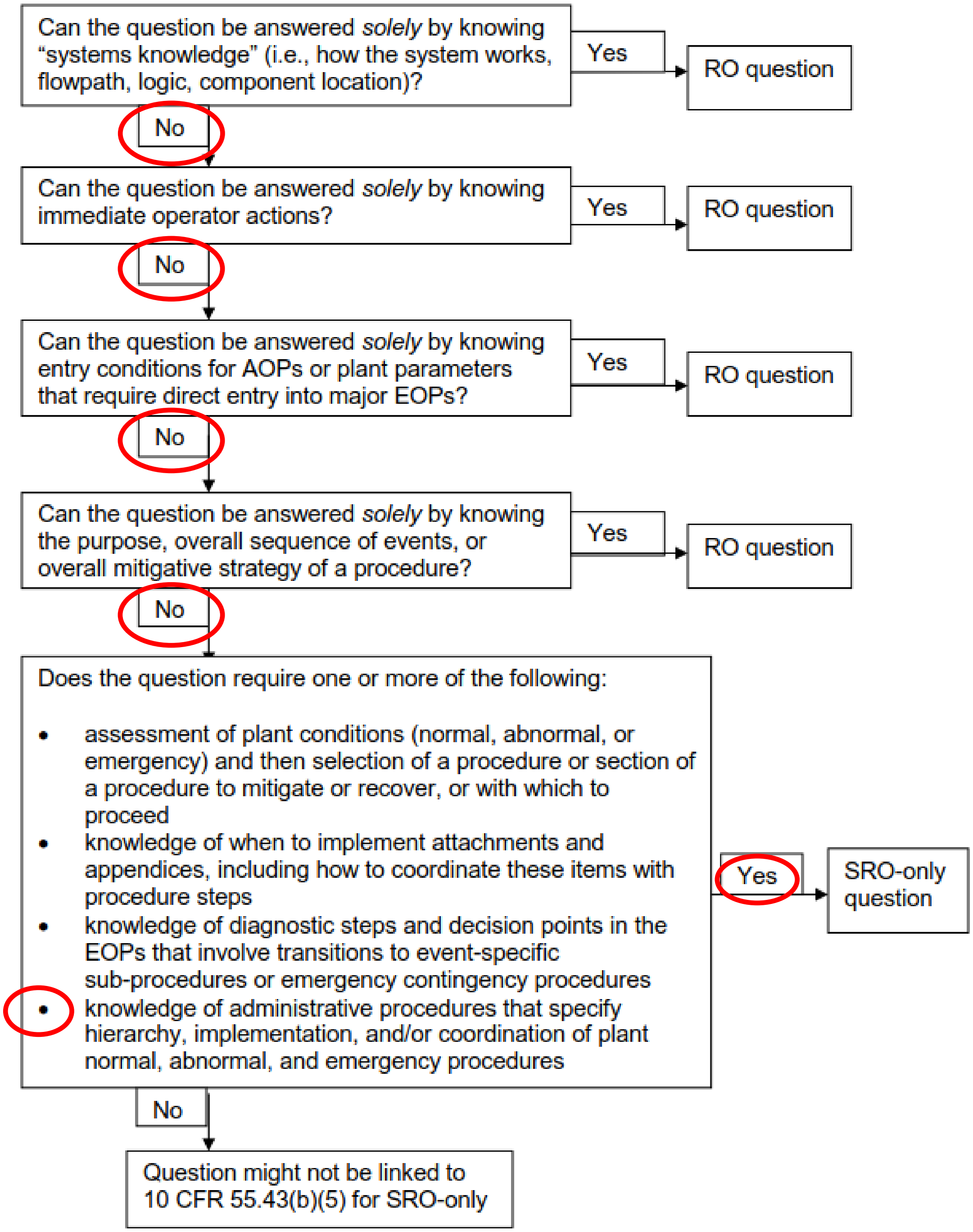
Answer: B
Explanation: Using Attachement 1 of EN-OP-115-14, Reactivity Management, the SRO must determine the risk level of the manipulation. Sequence exchange or power maneuver where power is lowered to <70% is considered a High Risk Reactivity Manipulation. The Required Minimum Controls for Reactivity Risk Level for a High Risk Reactivity call for Reactivity SRO Oversight (Dedicated SRO, other than the CRS, with no concurrent duties) 'B' is correct.
Distracters: 'A' is wrong – Procedure states This would be a Low risk manipulation. Sequences exchanges or power maneuvers where power is not lowered to less than 70% 'C' is wrong – Procedure states This would be a Normal Risk Reactivity Manipulation. Control rod exercise testing

‘D’ is wrong - Procedure states This would be a Normal Risk Reactivity Manipulation. Power ascension that is limited by the ramp rate
K/A Match The candidate must have knowledge of procedures, guidelines, or limitations associated with reactivity management.
Technical References: EN-OP-115-14, Reactivity Management, Rev 1
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-PROC, OBJ. 4.10

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



5.3 Reactivity Management Impact and Risk Level Determination

1. **IF** an activity has **NOT** been reviewed for a Reactivity Impact, **THEN INITIATE** actions to determine whether a procedure, activity, deficiency or other action has an impact to Reactivity Management.

NOTE

The following Options are available to assist in determining the Reactivity Management Risk Level:

- The work order plant impact statement may indicate the reactivity management risk level.
- The work order remarks section may indicate the reactivity management risk level.
- The procedure section being used may indicate the reactivity management risk level.

2. **IF** an activity or deficiency is determine to have a Reactivity Management Impact, **THEN** Operations SRO **DETERMINE** the Reactivity Management Risk Level Classification per Attachment 1.

3. **IF** an activity or deficiency is determined to have a Reactivity Management Impact, **THEN** Operations SRO **DETERMINE** the Minimum Required Defenses per the Required Minimum Controls For Reactivity Risk Level listed in Attachment 1.

4. **IF** an activity involves movement of control rods on a BWR, **THEN REFER TO** Attachment 2.

5.4 Low Power Operation

NOTE

Reactor operation at low power levels for extended periods of time is discouraged.

1. **PERFORM** the following additional requirements for extended low power operations:

   - **OBTAIN** Operations Manager approval for ALL low power operation.
   - **(BWR)** Station management carefully **CONSIDER** the risk of operating during off normal plant conditions such as low power operation and single loop operations.
   - Guidance should **IDENTIFY** potential problems that could be encountered such as the possibility that the core might become subcritical and predefine conditions under which operators should shut down or manually trip the reactor.

6.0 RECORDS

None

Reactivity Management Impact and Risk Level Determination

BWR REACTIVITY MANAGEMENT ACTIVITY RISK CLASSIFICATION
Normal Risk Reactivity Manipulation
• Planned adjustments to control rod position or recirculation flow where CRS direct oversight is adequate due to the short duration, low probability of mis-operation, and low number of concurrent activities. • Examples of activities that meet this requirement: • Adjustments with incremental changes of less than 5% using core flow • Control rod exercise testing • Power ascension that is limited by the ramp rate
Low Risk Reactivity Manipulation
• Planned adjustments to control rod position or recirculation flow that is greater in magnitude than Normal Risk Reactivity Manipulation, but where adequate direct oversight can be provided without a dedicated SRO. • Examples of activities that meet this requirement: • Sequences exchanges or power maneuvers where power is not lowered to less than 70% • Partially withdrawn control rod testing
High Risk Reactivity Manipulation
• Planned, non-transient changes to control rod position or recirculation flow that meets any of the following criteria: • Reactor startup [2.3.1] • Reactor shutdown • Power Suppression Testing • Sequence exchange (once the sequence exchange is complete, the power ascension rate can be evaluated and classified as a Low Risk Reactivity Manipulation) • Power maneuvers where power is lowered to less than 70%, major equipment manipulations, or complex surveillance testing. (Once the major equipment manipulation or testing is complete, the power ascension rate can be evaluated and classified as a Low Risk Reactivity Manipulation)

Reactivity Management Impact and Risk Level Determination

REQUIRED MINIMUM CONTROLS FOR REACTIVITY RISK LEVELS
Normal Risk Reactivity
• Beginning of shift Pre-job briefing • Dedicated RO with no concurrent duties (can be the ATC operator) • Dedicated SRO oversight (can be CRS/STA with no concurrent distractions) • Minimize control room activities that would distract from reactivity manipulations.
Low Risk Reactivity
• Pre-job briefing • Dedicated RO with no concurrent duties (can be the ATC operator) • Dedicated SRO oversight (can be CRS/STA with no concurrent distractions) • Minimize control room activities that would distract from reactivity manipulations. • Shift Manager or additional SRO oversight in the control room. • Restrict control room access IAW EN-OP-115-02
High Risk Reactivity
• Approved Reactivity Maneuvering Plan • Pre-job Briefing • Senior Operations Management Presence (As determined by Sr OPS MGR) • Reactor Engineering support • STA or additional SRO Oversight • Reactivity SRO oversight (Dedicated SRO, other than the CRS, with no concurrent duties) 2.3.1 • ALL unrelated parallel Control Room activities evaluated for distractions and approved by the Shift Manager and the Reactivity SRO prior to commencement. (minimal impact to control room team, no excessive alarms) • Consider Just-In-Time training • Restrict control room access IAW EN-OP-115-02

Examination Outline Cross Reference	Level	SRO
2.2.7 Knowledge of the process for conducting special or infrequent tests.	Tier	3
	Group #	
	K/A	2.2.7
	Rating	3.6
	Revision	1
Revision Statement: **Rev 1 Per NRC review (10 free view) changed the answers to ensure all numbers are used the same amount.**		

Question: 96

Consider the following:

1. Reactor startup
2. Containment entry at power
3. CORE ALTERATIONS due to Fuel movement.
4. Intentionally draining the reactor cavity level down to the RPV upper flange in MODE 5

Per EN-OP-116, Infrequently Performed Tests or Evolutions (IPTEs), which of the above is/are **required** to be controlled as IPTEs?

A. 1, 2 and 3 only.

B. 2, 3 and 4 only.

C. 1, 2 and 4 only.

D. 1, 3 and 4 only

Answer: **D**
Explanation: See EN-OP-116, Attachment 9.1 (Pre-identified IPTEs). Items 1 and 3 are contained under the section labeled All Units. Item 4 is contained under the section labeled BWR Units. 'D' is correct.
Distracters:

‘A’ is wrong - 4 is also an IPTE per the procedure.

‘B’ is wrong – 3 is not a IPTE but a normal action for GGNS and 1 & 2 are IPTEs per the procedure

‘C’ is wrong – 3 is not a IPTE but a normal action for GGNS and 4 is listed per the procedure as an IPTE.

K/A Match

The candidate must have the knowledge of what is listed as IPTEs in the procedure.

Technical References:

EN-OP-116, IPTEs

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 44.5

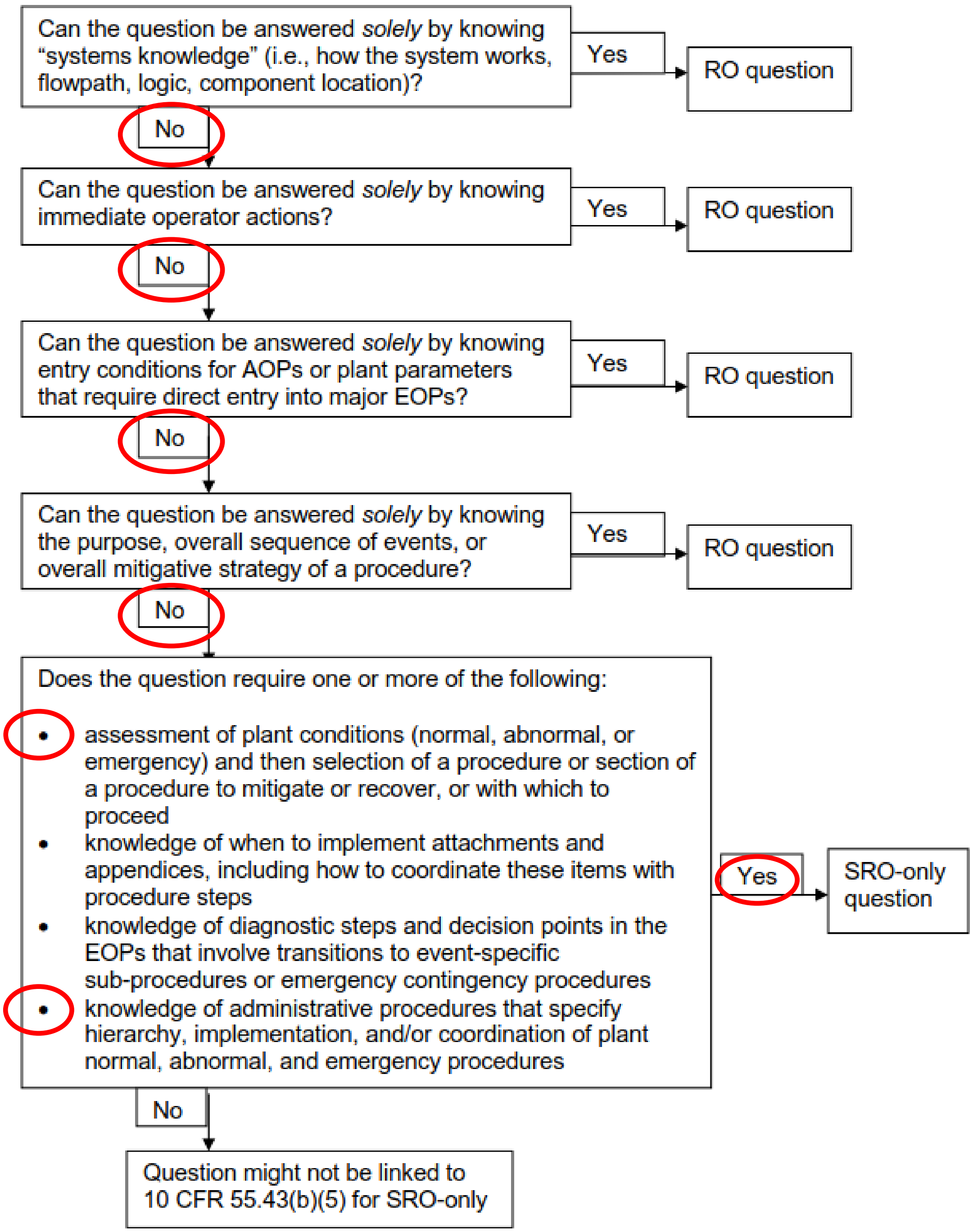
Question Source:	**Bank # 179**	Not used on a NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Attachment 2

Identified IPTEs [2.2.10]

NOTE

- These lists of identified IPTEs are not all inclusive. Activities not listed here are still being screened, as appropriate, in accordance with Section 5.1, Guidelines for Identification of IPTEs [2.2.4].
- Planned power changes greater than 30 percent are evaluated by the OM for IPTE controls.

All Units

- Radiological Diving
- Freeze seal utilized on primary coolant boundary while on-line, or which results in an Operational Potential for Draining Reactor Vessel (OPDRV) during an outage
- Low Power Physics Testing
- Reactor Start Up
- Back feeding and restoration of Main Transformers, when these evolutions are complex
- Containment Integrated Leak Rate Testing
- Reactor Head Removal
- Reactor Head Installation
- Upper Guide Structure (UGS) Removal
- Upper Guide Structure (UGS) Installation
- Incore Instrument (ICI) Cutting
- Fuel and CEA movements in core
- Any test that actually over-speeds a turbine or Emergency Diesel Generator
- Dry Fuel Storage (DFS) Activities that involve movement of a loaded canister from the Spent Fuel Pool (only mandatory for the first transfer evolution of a DFS Campaign)
- DFS Activities for transfer of a loaded canister to concrete overpacks or the transfer trailer (only mandatory for the first transfer evolution of a DFS Campaign)
- DFS unloading (mandatory for each cask unloaded)
- Integrated Emergency Diesel Generator/Engineering Safety Features Test
- Activities which could potentially impact Reduced Inventory operations

Identified IPTEs [2.2.10]

PWR Units

- Reactor Coolant System Drain Down to Lowered Inventory (For PWR Units, Lowered Inventory is Fuel in the Reactor with RCS level at or below the Reactor Vessel flange) [2.2.11]
- Incore Instrument (ICI) Withdrawal (refueling outages)

BWR Units

- Reactor Coolant System Drain Down to Lowered Inventory (For BWR Units, Lowered Inventory is Fuel in the Reactor with RCS level at or below the Reactor vessel flange and the Reactor Head detensioned) [2.2.11]
- Containment\Drywell entry at power and initial entry following shut down (does not apply to Containment entries at BWR-6 facilities)
- Control Rod Scram Time Testing (when performed less frequently than every six months)
- Leakage testing of the Reactor Coolant System

Examination Outline Cross Reference	Level	SRO
2.2.21 Knowledge of pre- and post-maintenance operability requirements.	Tier	3
	Group #	
	K/A	2.2.21
	Rating	4.1
	Revision	0
Revision Statement:		

Question: 97

HANDOUT PROVIDED

A small packing leak was discovered on P11-F130, REFUEL WTR XFER PMP SUCT FM SUPP POOL.

Maintenance has tightened the packing on the valve per a Work Order.

A maintenance leak check was performed and the leak was stopped.

What post-maintenance testing is required to be performed by Operations before the Work Order may be closed?

A. Functional stroke of valve IAW P11 SOI

B. Timed stroke of valve IAW P11 valve operability surveillance test

C. Functional stroke of valve with local visual observation IAW P11 SOI

D. Local leak rate test (LLRT) to verify Suppression Pool leakage within allowable limits IAW Engineering LLRT procedure

Answer:	**B**
Explanation:	

The candidate is expected to recognize that this is a safety–related primary containment isolation valve in the BOP condensate and refueling water transfer system.

Maintenance activities to tighten the packing of this valve represent a potential to affect the valve stroke time. A timed valve stroke per the appropriate surveillance procedure is required to demonstrate the valve meets the requirements of Tech Specs and the IST program.

‘B’ is correct.

Distracters:

‘A’ is wrong, but plausible. Although this will demonstrate functionality of the valve post–maintenance, Tech Specs and the IST program impose valve stroke time requirements that must be determined by a timed valve stroke performed IAW the applicable surveillance procedure.

‘C’ is wrong, but plausible. Although this will demonstrate functionality of the valve post–maintenance, Tech Specs and the IST program impose valve stroke time requirements that must be determined by a timed valve stroke performed IAW the applicable surveillance procedure and local observation is not required.

‘D’ is wrong, but plausible. Tightening of the valve packing does not present the potential to affect how leak–tight the valve is. While the original packing leak did present a concern for challenging limits on allowable Suppression Pool leakage into the Secondary Containment, a Maintenance leak check PMT is adequate to show that the leakage is stopped. A full LLRT of the penetration is not required.

K/A Match

The applicant must recognize the correct post maintenance test requirements

Technical References:

TS 3.6.1.3, Primary Containment Isolation Valves (PCIVs), Amendment 120
01–S–07–2, Test Control, Rev. 109, steps 2.6.1b, 5.12, 6.2.4b
EN-WM-107, Post Maintenance Testing, Attachment 3 Pg 2

Handouts to be provided to the Applicants during exam:

Learning Objective:

GLP-OPS-PROC, OBJ. 27.2

Question Source:	**Bank # 556**	**Dec 2017 NRC Exam Q97**
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		

None.

SRO Only Justification:

This question requires the applicant to first determine this valve is a PCIV within a BOP system. Then determine retest requirements.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

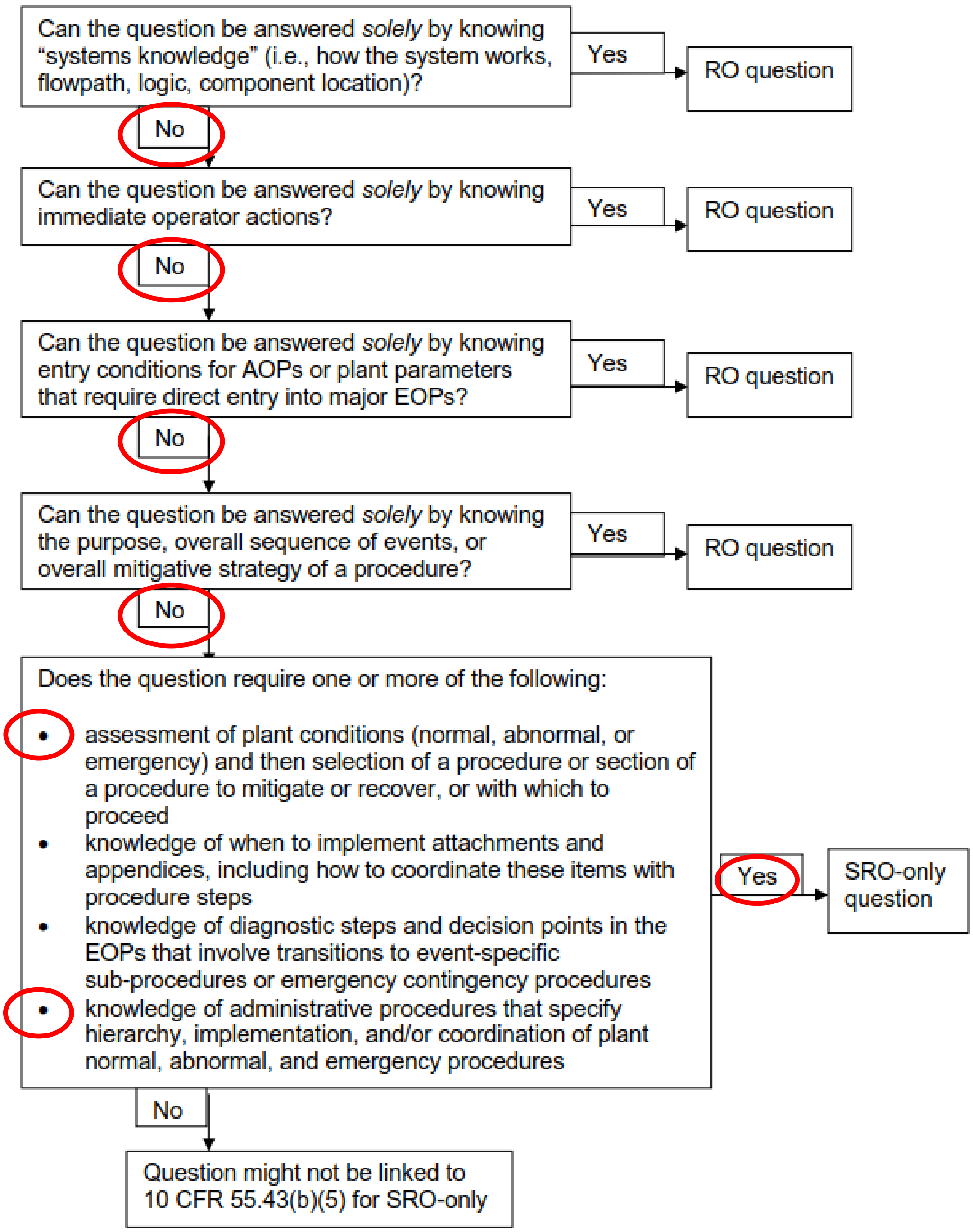


TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves#				
CTMT CLG EXH To CTMT VENT	M41-F034-(B)	35(I)	7	4
CTMT CLG EXH To CTMT VENT	M41-F035-(A)	35(O)	7	4
CTMT INST LINE ISOL VALVE	M71-F594-B	109D(O)	6A	60
CTMT INST LINE ISOL VALVE	M71-F595-A	109D(I)	6A	60
CST WTR SPLY HDR To CTMT	P11-F075-(A)	56(O)	6A	60
REFUEL WTR XFER PMP SUCT FM SUPP POOL	P11-F130-(A)	69(O) (c)	6A	60
REFUEL WTR XFER PMP SUCT FM SUPP POOL	P11-F131-(B)	69(O) (c)	6A	60
M/U WTR TREAT CTMT ISOL VLV	P21-F017-A	86(O)	6A	60
M/U WTR TREAT CTMT ISOL VLV	P21-F018-B	86(I)	6A	60
CTMT FLOOR DR SMP DISCH	P45-F061-(B)	51(I)	6A	110
CTMT FLOOR DR SMP DISCH	P45-F062-(A)	51(O)	6A	110
CTMT EQUIP DR SMP DISCH	P45-F067-(B)	50(I)	6A	110
CTMT EQUIP DR SMP DISCH	P45-F068-(A)	50(O)	6A	110
CTMT CHEM WST SMP DISCH	P45-F098-(B)	84(I)	6A	8
CTMT CHEM WST SMP DISCH	P45-F099-(A)	84(O)	6A	8
AUX BLDG EQ/FL DR PMP BK To SUPP POOL	P45-F273-A	60(O)	6A	32
AUX BLDG EQ/FL DR PMP BK To SUPP POOL	P45-F274-B	60(O)	6A	32
SVC AIR SPLY HDR To CTMT	P52-F105-(A)	41(O)	6A	60

2.4.13 Providing necessary personnel to coordinate testing activities with Scheduling, Maintenance and Operations through the Plant Scheduling Process.

2.4.14 Ensuring that personnel providing technical input or performing testing in accordance with the requirements of this procedure are qualified to do so in accordance with an approved training program or process.

2.5 Discipline Maintenance Superintendents - Through section personnel are responsible for performing specified maintenance, testing and inspection activities before affected equipment is returned to the Operations section.

2.5.1 Ensuring that test procedures and in-shop testing are performed properly, reviewed, and documented by qualified personnel.

2.5.2 Ensuring that testing requirements in the work package are specified prior to beginning work, consistent with the scope of work performed.

2.5.3 Coordinating with Scheduling, Engineering and Operations through the Plant Scheduling Process to schedule and perform testing activities.

2.5.4 Coordinating with Scheduling, Engineering and Operations through the Plant Scheduling Process to ensure that all delayed tests are performed as required. Delayed Tests are defined in Section 5.0.

2.6 Manager, Operations – Through appropriate Section Personnel is responsible for:

2.6.1 Verifying operability of equipment before returning the equipment to service, including:

a. Verifying all specified testing activities contained in a work package have been signed, electronically or otherwise, as complete.

b. Specifying, performing and verifying Operations testing requirements for work packages.

c. Identifying and specifying additional testing activities, if not previously specified in the work package.

2.6.2 Ensuring that testing activities that potentially affect equipment operability are authorized, performed, reviewed, and documented before restoring the equipment to Operable status.

2.6.3 Ensuring that Operations personnel performing test activities are properly qualified to do so in accordance with an approved training program or process.

2.6.4 Coordinating with Scheduling, Engineering and Maintenance through the Plant Scheduling Process to schedule and perform testing activities.

5.5 DELETE

5.6 Engineering Change Test (ECT) - A procedure used to perform post modification functional testing. ECTs demonstrate that modified or affected systems, structures, or components (SSCs) will perform satisfactorily in service and satisfy design requirements. Construction tests and inspections are excluded from this definition. ECTs are not to be used for Post-Maintenance Testing or for Special Testing, since they are generated for testing a completed plant modification related to a specific ER / EC Response.

5.7 Functional Test - A test performed in accordance with written or existing test procedures that demonstrates a structure, system or component (SSC) will perform satisfactorily in service. This test must be sufficient to confirm expected system or component results and that the maintenance or modification activity did not reduce safety of operation. Also refer to additional information in Nuclear Management Manual EN-DC-117, Post Modification Testing and Special Instructions.

5.8 In-Service Inspection Coordinator - Individual responsible for specifying the required in-service inspection requirements for ANSI/ASME components.

5.9 In-Service Testing Coordinator - Individual responsible for specifying the required in-service testing requirements for ANSI/ASME components.

5.10 Infrequently Performed Test or Evolution (IPTE) - Activities that are infrequently performed which have the potential to significantly degrade nuclear, radiological, or personnel safety and/or equipment/plant reliability. Refer to 01-S-06-58, Infrequently Performed Tests or Evolutions for additional information and requirements concerning IPTE's.

5.11 Inspection - Physical verification by means of examination, observation, and/or measurement to verify that an item meets predetermined requirements.

5.12 Maintenance Leak Check - Leak checks specified in the work package that are not performed to verify system or component operability. These leak checks are performed to ensure that maintenance activities performed to correct packing leaks, seal leaks and certain gasketed joint leaks were successful.

5.13 Deleted

6.2.2 (Cont.)

(2) Testing performed in accordance with 17-S-03-16 must be reviewed by the Valve Test Engineer or designee.

NOTE

The seat leakage test described below is not to be confused with an LLRT. These valves do not provide penetration isolation functions.

(3) The following listed valves require a Seat Leakage Test any time work activities potentially affect the leak tightness (see NOTE, above) of the valve seats for the following valves:

- P42-F105
- P42-F200A
- P42-F200B
- P42-F201A
- P42-F201B
- P44-F042
- P44-F054
- P44-F067
- P42-F205

6.2.3 Equipment Qualification (EQ) Program (NUREG 0588) Requirements

a. Guidelines for test requirements for EQ components are detailed in 07-S-01-227, Equipment Qualification Program.

6.2.4 GGNS Technical Specifications - Surveillance Program Requirements

a. Tests must be identified and performed to ensure compliance with the requirements of 01-S-06-12, GGNS Surveillance Program, when maintenance or modification work activities:

(1) Invalidate existing surveillance procedures

(2) Invalidate current surveillance test acceptance criteria

(3) Supplement current surveillance requirements

(4) Add new surveillance requirements

b. Review Work Order work scope and test activities carefully to determine if Inservice Testing (IST) required parameters, Technical Specification Acceptance Criteria or Baseline Data, such as pump vibration, flow, differential pressure (dp), or stroke times, etc., may be affected.

Examination Outline Cross Reference	Level	SRO
2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	**Tier**	3
	Group #	
	K/A	2.3.4
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 98

A Site Area Emergency is in progress.

A control room RO needs to be sent into the plant to perform a task.

Per 10-S-01-17, against what dose limits does the CRS (without additional concurrence) compare the current dose to determine if there is sufficient exposure margin?

A. 10CFR100 Reactor Site Criteria

B. GGNS Administrative Dose Limits

C. Authorized Emergency Exposure Dose Limits

D. 10CFR20 Standards for Protection Against Radiation

Answer: D
Explanation: See 10-S-01-17, sections 6.1.2 and 6.1.3. By definition, the administrative dose limits are automatically suspended and replaced by the 10CFR20 Federal dose limits at the declaration of an Alert emergency or higher. See 10-S-01-17, sections 6.2.1 and 6.7.1.b. The latter section clearly states that if the limit is likely to be exceeded (i.e., after determining if sufficient exposure margin exists), then obtain an Exposure Extension Authorization per section 6.1. Clearly then, the SRO compares the RO's current dose against the 10CFR20 Federal limits. 'D' is correct.
Distracters: 'A' is wrong – CFR 100 deals with Reactor Site Criteria.

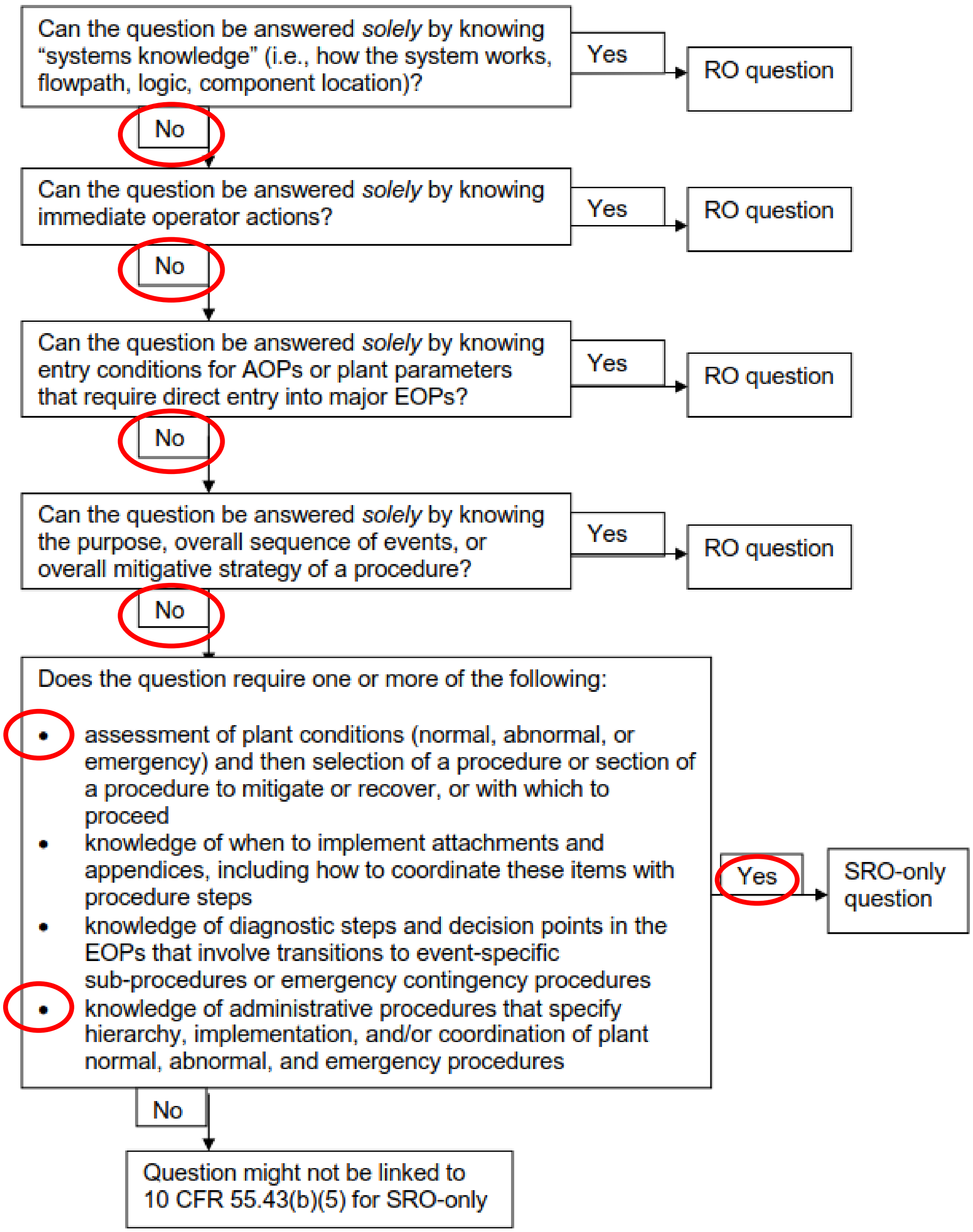
‘B’ is wrong – the administrative dose limits are automatically suspended and replaced by the 10CFR20 Federal dose limits at the declaration of an Alert emergency or higher. ‘C’ is wrong - Section 6.1 deals with the limits suggested by this answer, these limits are for protecting equipment and life.
K/A Match
Technical References: 10-S-01-17, Emergency Personnel Exposure Control
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-EPTS6, OBJ. 17

Question Source:	**Bank #**	2010 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(4)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

NOTE – This is an SRO-only question for the following reasons: 1) clearly, it relates an SRO-only responsibility in the said procedure (i.e., to determine sufficient exposure margin); 2) although the actual question being asked is a fairly elementary one (i.e., “What dose limit is to be used?”), this item of knowledge is not at all common knowledge among the non-Radiation Protection Personnel population, most especially among the RO population. Nonetheless, because this “elementary” information is associated with an SRO’s responsibility during a plant emergency situation, it is reasonable to expect an SRO Candidate to know that we suspend the usual GGNS Admin Dose Limits during a Site Area Emergency.

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



6.1 Emergency Exposure Authorization

TABLE 1

DOSE LIMITS FOR EMERGENCY WORKERS

The following table represents those extensions of administrative exposure limits for which authorization, by the Emergency Director (ED) or Emergency Plant Manager (EPM), shall be obtained prior to the exposure being received:

DOSE LIMITS (TEDE)	ACTIVITY	CONDITIONS
>5 up to 10 Rem	Protecting Valuable Property	Lower dose not practicable
>10 up to 25 Rem	Life saving or Protection of Large Populations	Lower dose not practicable
>25 Rem	Life saving or Protection of Large Populations	Only on a voluntary basis to persons fully aware of the risks involved

6.1.1 Workers performing services during emergencies should limit dose to the lens of the eye to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value.

6.1.2 The administrative limits and extension process of Reference 3.4 are automatically suspended at the declaration of an Alert, Site Area Emergency, or General Emergency.

6.1.3 Emergency Response personnel are administratively extended to the Federal Limits of 10CFR20 at the declaration of an Alert, Site Area Emergency, or General Emergency. Efforts are made to maintain personnel exposures within the limits established by 10CFR20.

NOTE

Margins are not automatically extended in the RP computer program. Manual reset of exposure margins in the computer or manually resetting EAD setpoints is required by Radiation Protection Personnel.

6.1.4 Each situation in which an individual may receive exposure in excess of Federal Limits must be evaluated with regard to the risk to the individual, protection of valuable property, or protection of other persons.

6.1.5 Those individuals requesting authorization should do so through the appropriate Lead Radiation Protection personnel (RAC or RC), if possible.

Examination Outline Cross Reference	Level	SRO
2.4.29 Knowledge of the emergency plan.	Tier	3
	Group #	
	K/A	2.4.29
	Rating	4.4
	Revision	1

Revision Statement:

Rev 1, Per Ops Rep, added alarm FP LEVEL TROUBLE, this alarm is the basis of the EP-4 entry condition.

Question: 99

HANDOUT PROVIDED

The plant is in Day 5 of a Refueling Outage.

The Reactor cavity pool is drained and preps are being made to remove the Drywell Head.

New fuel is being transferred to the Containment Pool.

The Reactor Cavity gate seal fails.

P680-4A2-A6, FP LEVEL TROUBLE alarm is received

Fuel Pool Cooling and Cleanup pumps have tripped on low Drain Tank level.

RO reports ARM D21-K622, Aux Bldg Fuel Hdlg Area, indication is rising.

(1) What is the current emergency classification?

(2) Which procedure(s) should be entered to mitigate this event?

A. (1) Unusual Event.
(2) High Radiation During Fuel Handling ONEP only

B. (1) Alert.
(2) EP-4 and High Radiation During Fuel Handling ONEP

C. (1) Unusual Event.
(2) EP-4 and High Radiation During Fuel Handling ONEP

D. (1) Alert.
(2) High Radiation During Fuel Handling ONEP only

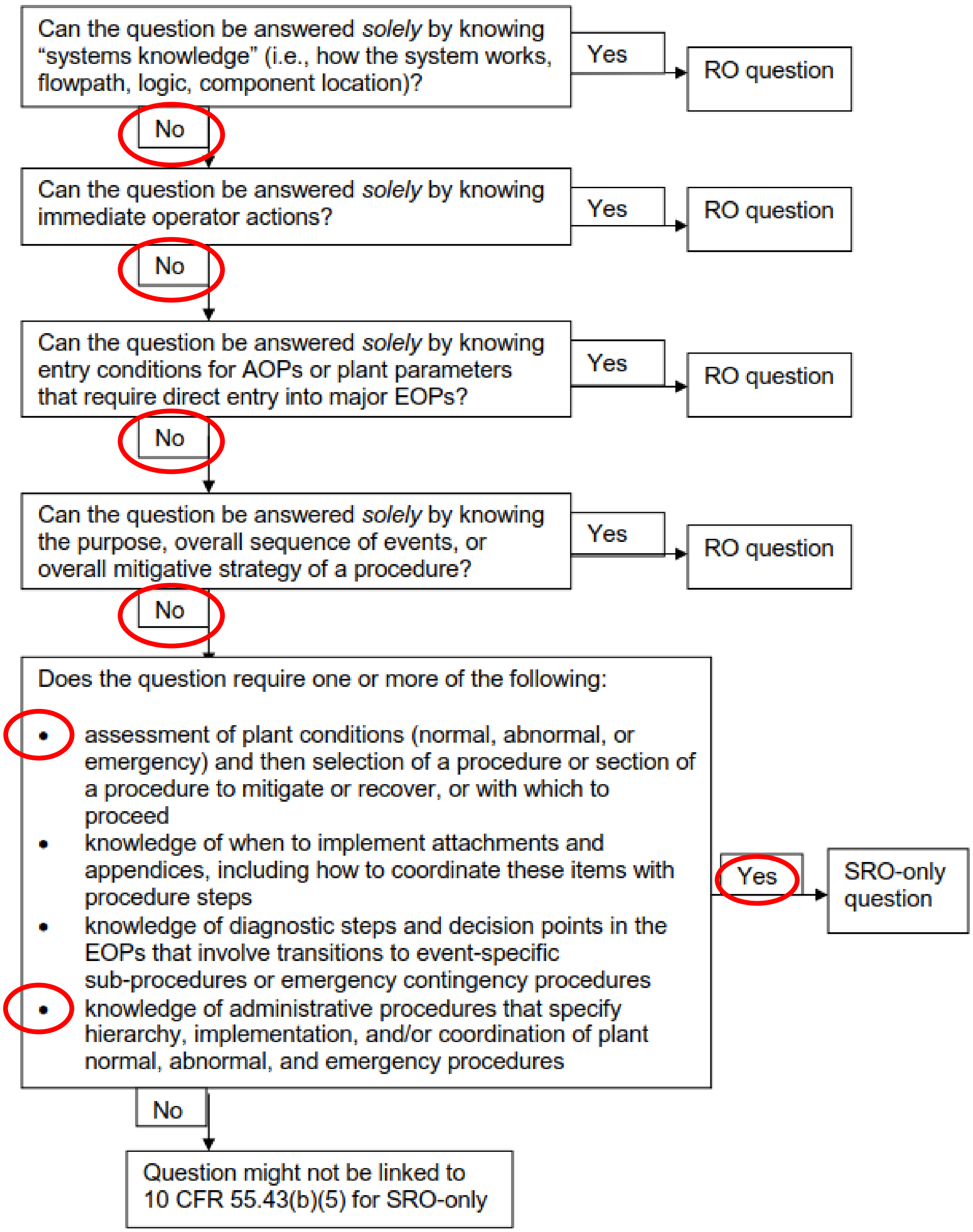
Answer: **C**
Explanation: Per EAL flow charts Modes 4 through De-Fuel UNUSUAL EVENT – AU2 UNPLANNED water level drop in reactor refueling pathways as indicated by water level drop in Upper Ctmt Pools, Aux Bldg Fuel Pools or the Fuel Transfer Canal, personnel observation or indication on area camera. **<u>AND</u>** VALID Area Radiation Monitor reading rises on any of the following: Ctmt 209 Airlock---------------------------------- **(1D21K630)** Ctmt Fuel Hdlg Area ---------------------------- **(1D21K626)** Aux Bldg Fuel Hdlg Area----------------------- **(1D21K622)** The given information is lowering pool levels and a rising radiation on D21K622, therefore an Unusual Event is required. When the Fuel Pool Cooling and Cleanup pump tripped on low drain tank level meant that the “scuppers” are uncovered due to the lowering level the scuppers are located below level of 207.49 ft. which is an entry condition for EP-4. The CRS should enter EP-4 and also enter High Rad During Fuel Handling ONEP. ‘C’ is correct
Distracters: ‘A’ is wrong – EP-4 has met an entry condition and is required to be entered. Plausible due to the candidate must have the knowledge of relative levels of the Spent Fuel Pools vs. EP-4 entry condition. ‘B’ is wrong – Only an UE is declared, the upgrade to an ALERT requires level to reach the fuel which with the given information will not happen due to the fuel is recessed within the pool and level will not reach them and a high rad alarm which has not come in yet. ‘D’ is wrong – EP-4 has met an entry condition and is required to be entered. Plausible due to the candidate must have the knowledge of relative levels of the Spent Fuel Pools vs. EP-4 entry condition. Only an UE is declared, the upgrade to an ALERT requires level to reach the fuel which with the given information will not happen due to the fuel is recessed within the pool and level will not reach them and a high rad alarm which has not come in yet.
K/A Match Candidate must have knowledge of the Emergency Play to correctly classify this event.

Technical References: 10-S-01-1, Activation of Emergency Plan EP-4 05-1-02-II-8, High Radiation During Fuel Handling
Handouts to be provided to the Applicants during exam: EAL Flow Charts
Learning Objective: GLP-OPS-EPTS6, OBJ. 1 GLP-OPS-EP-4

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

SRO JUSTIFICATION

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



AU2

Initiating Condition -- UNUSUAL EVENT

UNPLANNED rise in plant radiation

Operating Mode Applicability: **All**

Example Emergency Action Level(s): (1 or 2)

1. a. UNPLANNED water level drop in reactor refueling pathways as indicated by water level drop in Upper Ctmt Pools, Aux Bldg Fuel Pools or the Fuel Transfer Canal, personnel observation or indication on area camera.

   AND

   b. VALID Area Radiation Monitor reading rises on any of the following:

      Ctmt 209 Airlock ---------------------------- **(1D21K630)**

      Ctmt Fuel Hdlg Area ----------------------- **(1D21K626)**

      Aux Bldg Fuel Hdlg Area ----------------- **(1D21K622)**

OR

(Note: For Control Room Envelope review EAL AA3)

2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.

   NOTE: *For area radiation monitors with ranges incapable of measuring 1000 times normal* levels, classification shall be based on VALID full scale indications unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the Area Radiation Monitor indications going full scale.*

   *Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Basis:

This IC addresses rising radiation levels as a result of a water level drop above irradiated fuel or events that have resulted, or may result, in unexpected rise in radiation dose rates within plant buildings. These radiation rises represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant.

EAL #1

The refueling pathway is a site specific combination of cavities, tubes, canals and pools. While a radiation monitor could detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, a refueling bridge ARM reading may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, increased radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

For refueling events where the water level drops below the RPV flange classification would be via CU2. This event escalates to an Alert per AA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in operating modes 1-3.

EAL #2

This EAL addresses rises in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level rises that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area.

Examination Outline Cross Reference	Level	SRO
2.4.42 Knowledge of emergency response facilities.	Tier	3
	Group #	
	K/A	2.4.42
	Rating	3.8
	Revision	0
Revision Statement:		

Question: 100

After the Shift Manager is relieved, the Emergency Director reports to the ___________ facility.

What facility, after declared operational, will provide offsite notifications during an ERO emergency?

A. EOF
EOF

B. TSC
TSC

C. EOF
TSC

D. TSC
EOF

Answer: A
Explanation: Per 10-S-01-1, Activation of the Emergency Plan. 2.1.2 The Shift Manager assumes the role of Emergency Director upon initial classification of an emergency, and resumes normal Control Room duties when relieved by the EOF Emergency Director. 2.4.2 Assuming the duties of Emergency Director after the EOF is declared operational. Per EN-EP-609, Emergency Operations Facility (EOF) Operations. Offsite Communicator arrives at EOF and signs in to staffing board.

Assumes the position then obtain copies of all Emergency Notification forms that have been transmitted prior to EOF activation from the Control Room.

‘A’ is correct.

Distracters:

‘B’ is wrong – The Emergency Director responded to the TSC at one time but now, he responds to the EOF. The Emergency Plant Manager responds to the TSC

The TSC does have a Communicator but the positon only communicates with the other facilities not the state and local agencies.

‘C’ is wrong – The TSC does have a Communicator but the positon only communicates with the other facilities not the state and local agencies.

‘D’ is wrong - The Emergency Director responded to the TSC at one time but now, he responds to the EOF. The Emergency Plant Manager responds to the TSC

K/A Match

To understand the roles and responsibilities of ERO personnel the SRO must have the knowledge of all facilities within the ERO.

Technical References:

10-S-01-1, Activation of the Emergency Plan
EN-EP-609, Emergency Operations Facility (EOF) Operations
EN-EP-610, Technical Support Center (TSC) Operations

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

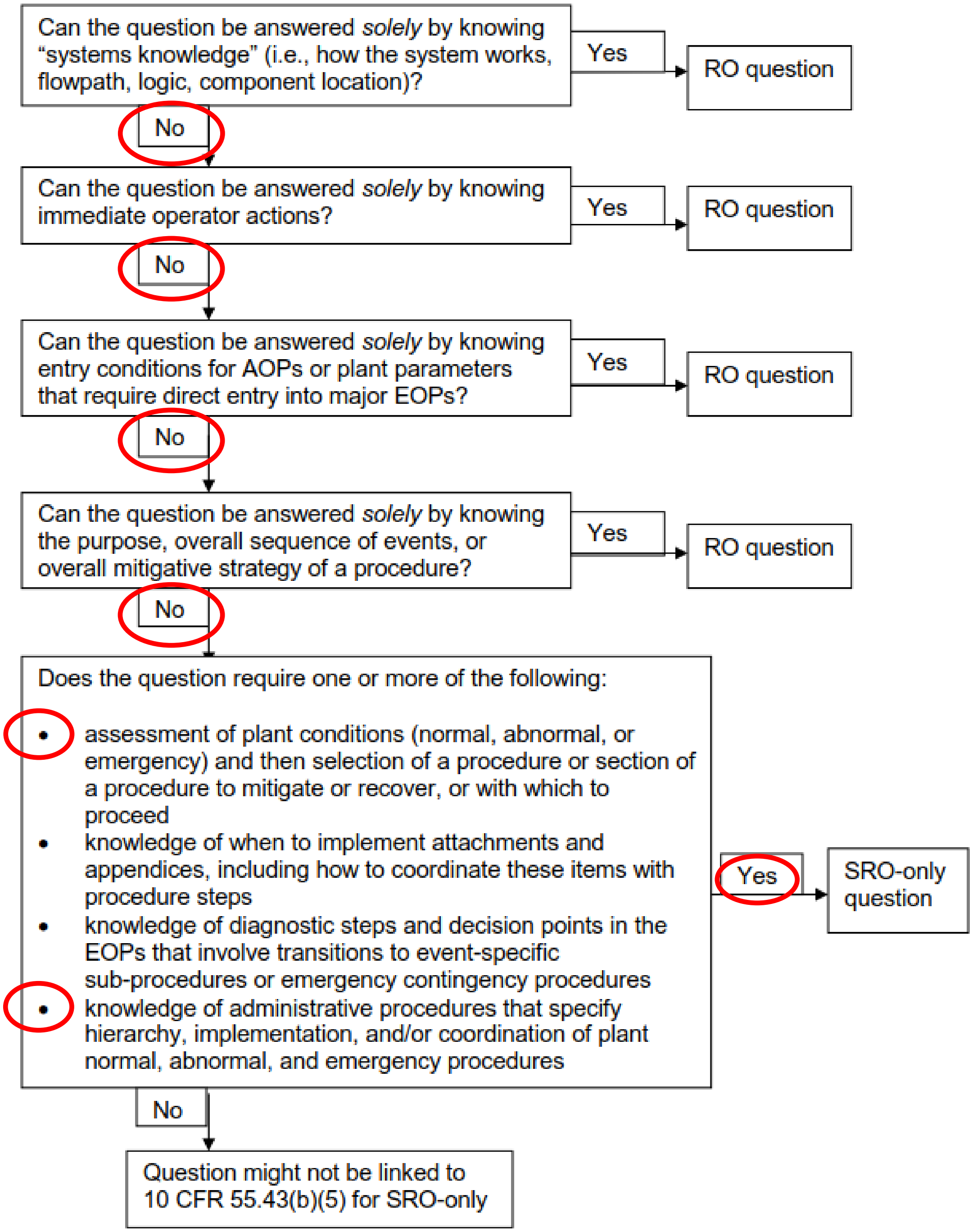
GLP-EP-EPTS26, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	

Level of Difficulty:	3.0	
PRA Applicability:		
None.		

SRO JUSTIFICATION

**Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)**



1.0 PURPOSE AND DISCUSSION

1.1 Purpose

1.1.1 This procedure provides guidance to:

a. Classify an emergency according to severity.

b. Assign responsibilities for emergency actions.

c. Establish lines of authority and communication.

d. Initiate emergency actions to safeguard the public and plant personnel.

e. Upgrade or terminate emergency classification when severity of event changes.

1.2 Discussion

1.2.1 Whenever plant conditions are identified that meet the Emergency Action Level Criteria in Attachment I or EPP 01-02 (Flowchart), this emergency plan procedure shall be implemented.

1.2.2 Emergency Plan section 6.1.2 allows suspension of normal Emergency Plan actions for Security Emergencies. This is permitted because of the potential risk to personnel safety which a security emergency may present. An armed attack against the plant is a unique security emergency that is expected to be an extremely fast moving event and present an immediate and serious threat to human life. Section 6.1.8 of this procedure contains special Emergency Plan actions for this unique event.

1.3 Changes required for implementation of 1994 TSIP were incorporated in Revision 100. For historical reference this statement should not be deleted.

2.0 RESPONSIBILITIES

2.1 Shift Manager - Is responsible for determining if emergency declaration is required.

2.1.1 If an Emergency Action Level (EAL) is reached or exceeded, the Shift Manager shall:

a. Classify the emergency and make the appropriate declaration if required.

b. Take action to ensure safe operation of plant and protection of plant personnel, the general public, and plant equipment.

c. Perform assessment actions.

d. Perform any other emergency actions as appropriate.

2.1.2 The Shift Manager assumes the role of Emergency Director upon initial classification of an emergency, and resumes normal Control Room duties when relieved by the EOF Emergency Director.

2.2 Operations Coordinator - The Operations Coordinator reports to the TSC Manager. Responsibilities include:

2.2.1 Coordinate TSC efforts in determining the nature and extent of emergencies pertaining to equipment and plant facilities in support of Control Room actions.

2.2.2 Assist the EPM in evaluating changes in event classification.

2.2.3 Ensure the Control Room, TSC, and EOF is informed of significant changes in event status.

2.2.4 Coordinate operations activities outside of the Control Room with the TSC Manager and OSC Manager.

2.3 Security Coordinator - The Security Coordinator is located in the Incident Command Post and reports to the Emergency Plant Manager. Responsibilities include:

2.3.1 Overall coordination of the offsite assistance for the security related response.

2.3.2 Designated NIMS (National Incident Management System) Liaison between the Incident Command Post and Site Organization.

2.4 EOF Emergency Director - Is responsible for:

2.4.1 Reporting to the site to assume the duties of Emergency Director upon notification of an Alert or higher classification.

2.4.2 Assuming the duties of Emergency Director after the EOF is declared operational.

2.4.3 Reporting to the site to assume duties of Emergency Director upon notification of an Unusual Event if he deems it necessary.

2.4.4 Evaluating the accident conditions and verifying that the correct emergency classification has been made.

3.0 REFERENCES

3.1 NRC Memorandum dated July 11, 1994 concerning "Branch Position on Acceptable Deviations to Appendix 1 to NUREG-0654/FEMA-REP-1".

3.2 GGNS Emergency Plan

3.3 10-S-01-6, Notification of Offsite Agencies and Plant On-Call Emergency Personnel.

3.4 10-S-01-11, Evacuation of Onsite Personnel

3.5 10-S-01-12, Radiological Assessment and Protective Action Recommendations.

3.6 10-S-01-22, Recovery

3.7 10-S-01-23, Reentry

3.8 EN-EP-609, Emergency Operations Facility (EOF) Operations

3.9 EN-EP-610 TECHNICAL SUPPORT CENTER (TSC) OPERATIONS

3.10 05-1-02-VI-4, Security Threat ONEP

1. **Termination**: The point at which the classified emergency is no longer considered to be an emergency. Unusual Event classifications are directly terminated. Alert classifications can be directly terminated or transitioned to Recovery. Site Area and General Emergency classifications must transition to Recovery for event termination.

2.0 RESPONSIBILITIES

1. The **entire staff** of the Emergency Operations Facility (EOF) is responsible for activation and operation of the facility using this procedure, their position specific checklists and applicable site-specific performance aids.

2. The **Emergency Director (ED)** is responsible for overall command and control of the emergency response, including classifications; notifications, PARs and ensuring all resources are available to mitigate emergency conditions.

3. The **EOF Manager** is responsible for overseeing operations of the EOF and assisting the ED in performance of key duties.

4. The **EOF Technical Advisor (TA)** is responsible for analyzing data from the plant and providing technical advice to the EOF Staff and tracking EALs based on plant data.

5. The **Radiological Assessment Coordinator (RAC)** is responsible for overseeing operations of the offsite dose assessment and all radiological controls for Entergy Personnel.

6. The **Offsite Communicator** is responsible for making required communications to offsite authorities.

7. The **Public Information Liaison** is responsible for providing information on emergency actions to the Joint Information Center (JIC).

8. The **Lead Offsite Liaison** is responsible for ensuring the offsite agencies, including NRC, located in the EOF and the offsite liaisons, located at the EOCs are briefed on the plant conditions.

9. The **Offsite Team Coordinator** is responsible for communicating with the offsite monitoring teams and supporting the RAC in dose assessment activities.

10. The **Administration and Logistics Coordinator** is responsible for providing support to the EOF Manager and the Emergency Response Organization. Responsibilities include managing 24-hour staffing of the emergency response facilities, managing logistics for supporting the onsite and offsite emergency response such as additional support personnel or equipment, meals, lodging, etc., and coordinating access security measures in the EOF if applicable.

11. The **IT Specialist** is responsible for assisting EOF staff in operation of facility computer and communications issues as necessary.

12. **Dose Assessor** is responsible for monitoring plant data for possible releases and performing dose projections based on any actual or potential releases.

Attachment 6 **Page 1 of 5**

Offsite Communicator

Offsite Communicator Name:_______________________________ Date:___________

1.0 INITIAL RESPONSIBILITY/ACTIVITIES

1. Initial Orientation

a. Upon arrival at the EOF:

1) **SIGN** in on EP-7-ALL, Facility Sign-In/Accountability Form

2) **SIGN** in on the EOF staffing board if applicable.

3) **OBTAIN** the Offsite Communicator binder and ID Badge

b. **REVIEW** available sources to obtain overall status of emergency situation.

c. **OBTAIN** a briefing from the Emergency Director (ED) or EOF Manager on emergency status.

2. **ASSUME** the position of Offsite Communicator

a. **<u>IF</u>** initial activation,
<u>THEN</u>:

1) **OBTAIN** copies of all Emergency Notification Forms that have been transmitted prior to EOF activation from Control Room.

2) **ENSURE** communication is available to the states and locals via primary or alternate communication methods by conducting a test of the system.

NOTE

Shift Manager retains responsibility for offsite agencies notifications until the EOF ED assumes responsibility for implementation of Emergency Plan.

3) **CONTACT** the communicator in the Control Room:

a) **DETERMINE** and **RECORD** other offsite notifications made. (NRC, Corporate).

b) **DETERMINE** and **RECORD** the next time required offsite agencies notifications are due.

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NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
SRO EXAM

TAB	PROVIDED HANDOUTS
1	OPG-37, Operating Hardcards, Operation's Section Guideline, Aux Building Area Parameters
2	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 039), HCTL Figure (Figure 1)
3	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 039), HDOL Figure (Figure 5)
4	Tech Specs 3.6.1.3, Primary Containment Isolation Valves (PCIVs)

Additional References Provided Outside of This Binder:

- Electrical Drawing E-1173-19
- Emergency Classification Flowcharts: (10-S-01-1, EPP 01-02), dated 11-21-2013
    - Modes 1 through 3 (page 1 of 2)
    - Modes 4 through De-Fuel (page 2 of 2)
- Steam Tables

TAB 1

Operating Hardcard	Aux Building Area Parameters	Date: 7/7/11

AREA	OPERATING LIMIT	MAX SAFE VALUE				
TEMPERATURE						
MSL PIPE TUNNEL TEMP	185°F(P601-18A-A3 & A4) (P601-19A-A3 & A4)	250°F E31-N604A,B,C,D,E,F	____	____	____	____
RHR EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F E31-N608A/B,N610A/B	____	____	____	____
RCIC EQUIP AREA TEMP	185°F (P601-21A-G3)	212°F (E31-N602A/B)	____	____	____	____
RADIATION LEVEL						
RHR ROOM A D21K601	10^2 MR/HR (P844-1A-D4)	8 X 10^4 MR/HR	____	____	____	____
RHR ROOM B D21K602	10^2 MR/HR (P844-1A-D4)	8 X 10^4 MR/HR	____	____	____	____
RHR HX A HATCH D21K611	10^2 MR/HR (P844-1A-C4)	8 X 10^4 MR/HR	____	____	____	____
RHR HX B HATCH D21K612	10^2 MR/HR (P844-1A-C4)	8 X 10^4 MR/HR	____	____	____	____
RCIC ROOM D21K603	10^2 MR/HR (P844-1A-D4)	8 X 10^4 MR/HR	____	____	____	____
MSL RAD MONITOR	Setpoint Log (P601-19A-D4)	8 X 10^4 MR/HR	____	____	____	____
SGTS FLTR TRN D21K613	2.5 MR/HR (P844-1A-C5)	8 X 10^2 MR/HR	____	____	____	____
WATER LEVEL						
RHR RM A	89′ 8″ (P680-8A1-A2)	93′ 6″ (P870-2A-E1)	____	____	____	____
RHR RM B	89′ 8″ (P680-8A1-B2)	93′ 6″ (P870-10A-G1)	____	____	____	____
RHR RM C	90′ 6″ (P680-8A1-C2)	93′ 6″ (P870-10A-G2)	____	____	____	____
RCIC RM	90′ 6″ (P680-8A1-C4)	93′ 6″ (P870-2A-A1)	____	____	____	____
LPCS RM	90′ 6″ (P680-8A1-A4)	93′ 6″ (P870-2A-F1)	____	____	____	____
HPCS RM	90′ 6″ (P680-8A1-B4)	93′ 6″ (P870-5A-H1)	____	____	____	____

"TWO MAX SAFE'S**"** = 2 TEMPS **or** 2 RADS **or** 2 WATER LEVEL

TAB 2

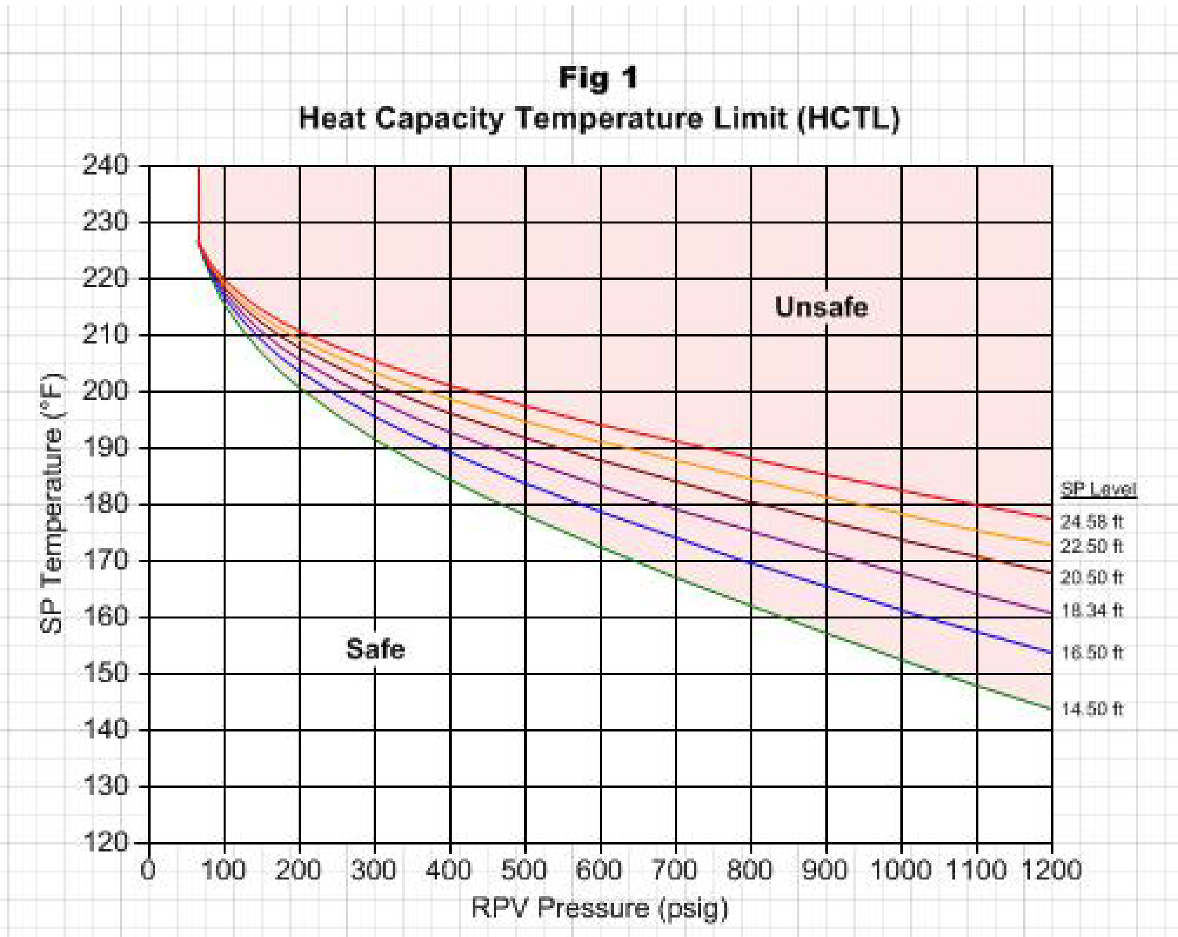
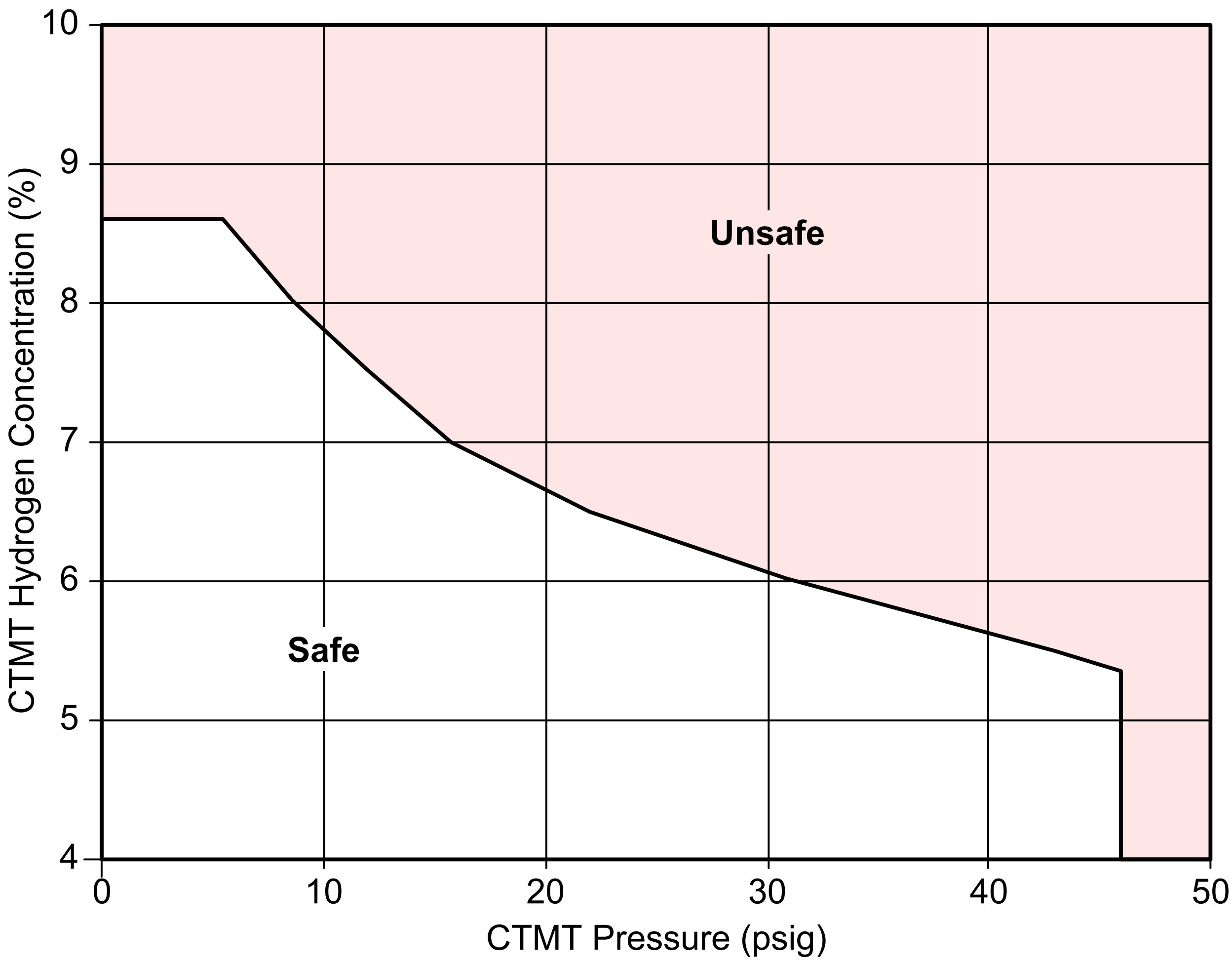


Fig 1
Heat Capacity Temperature Limit (HCTL)
Unsafe
Safe
SP Temperature (°F)
RPV Pressure (psig)
240
230
220
210
200
190
180
170
160
150
140
130
120
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
SP Level
24.58 ft
22.50 ft
20.50 ft
18.34 ft
16.50 ft
14.50 ft

TAB 3

Figure 5
Hydrogen Deflagration Overpressure Limit (HDOL)



TAB 4

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated isolation instrumentation is required to be OPERABLE per LCO 3.3.6.1 Function 2.g.

ACTIONS

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

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

(continued)

ACTION (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.	-------------NOTE------------ Relief valves are not required to be de-activated provided the relief setpoint is at least 23 psig and one of the following criteria is met: 1. the relief valve is one-inch nominal size or less, or 2. the flow path is into a closed system whose piping pressure rating exceeds the containment design pressure rating. ----------------------------	
	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours except for main steam line AND 8 hours for main steam line
	AND	
	A.2 --------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment, drywell, and steam tunnel AND (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)		Prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment, drywell, or steam tunnel
B. One or more penetration flow paths with two PCIVs inoperable except due to leakage not within limit.	------------NOTE----------- Relief valves are not required to be de-activated provided the relief setpoint is at least 23 psig and one of the following criteria is met: 1. the relief valve is one-inch nominal size or less, or 2. the flow path is into a closed system whose piping pressure rating exceeds the containment design pressure rating. ---------------------------	
	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more penetration flow paths with leakage rate not within limit except for purge valve leakage.	C.1	Restore leakage rate to within limit.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more penetration flow paths with one or more primary containment purge valves not within purge valve leakage limits.	D.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
	<u>AND</u>		
	D.2	--------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment <u>AND</u> Prior to entering MODE 2 or 3 from MODE 4 if not performed within the previous 92 days for isolation devices inside primary containment
	<u>AND</u>		
	D.3	Perform SR 3.6.1.3.5 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per 92 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours
F. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during movement of recently irradiated fuel assemblies in the primary or secondary containment.	F.1 ------------NOTE-------------- LCO 3.0.3 is not applicable. ---------------------------------- Suspend movement of recently irradiated fuel assemblies in primary and secondary containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	-------------------------------NOTES------------------------- 1. Only required to be met in MODES 1, 2, and 3. 2. Not required to be met when the 20 inch primary containment purge valves are open for pressure control, ALARA, or air quality considerations for personnel entry. 3. Not required to be met during Surveillances or special testing on the purge system that requires the valves to be open. 4. 20 inch primary containment purge valves shall not be open with the 6 inch primary containment purge or the drywell vent and purge supply and exhaust lines open. --- Verify each 20 inch primary containment purge valve is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.2	-----------------------------NOTES--------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment isolation manual valve and blind flange that is located outside primary containment, drywell, and steam tunnel and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	-------------------------NOTES------------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. -- Verify each primary containment isolation manual valve and blind flange that is located inside primary containment, drywell, or steam tunnel and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify the isolation time of each power operated, automatic PCIV, except MSIVs, is within limits.	In accordance with the INSERVICE TESTING PROGRAM

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	------------------------------NOTE------------------------------ Only required to be met in MODES 1, 2, and 3. -- Perform leakage rate testing for each primary containment purge valve with resilient seals.	In accordance with the Surveillance Frequency Control Program AND In accordance with 10 CFR 50, Appendix J, Testing Program AND ---------Note------- Not applicable to valves tested within 92 days prior to any purge valve failing to meet its acceptance criteria ------------------------- Once within 92 days, test all remaining purge valves, if any purge valve fails to meet its acceptance criteria

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.8	---NOTE--- Only required to be met in MODES 1, 2, and 3. --- Verify leakage rate through each main steam line is ≤ 100 scfh when tested at ≥ P_a, and the total leakage rate through all four main steam lines is ≤ 250 scfh when tested at ≥ P_a.	In accordance with 10 CFR 50, Appendix J, Testing Program
SR 3.6.1.3.9	---NOTE--- Only required to be met in MODES 1, 2, and 3. --- Verify combined leakage rate of 1 gpm times the total number of PCIVs through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at ≥ 1.1 P_a.	In accordance with 10 CFR 50, Appendix J, Testing Program

--------------------------------------NOTE--------------------------------------
This table provides the listing of components required by LCO 3.6.1.3

TABLE TR3.6.1.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves#				
INBD MSL DR INBD DR VLV	B21-F016-B*	19(I)	1	20*
INBD MSL DR OTBD DR VLV	B21-F019-A	19(O)	1	20
MSL "A" DRWL INBD ISOL	B21-F022A	5(I)	1	5
MSL "B" DRWL INBD ISOL	B21-F022B	6(I)	1	5
MSL "C" DRWL INBD ISOL	B21-F022C	7(I)	1	5
MSL "D" DRWL INBD ISOL	B21-F022D	8(I)	1	5
MSL A CTMT OTBD ISOL	B21-F028A	5(O)	1	5
MSL B CTMT OTBD ISOL	B21-F028B	6(O)	1	5
MSL C CTMT OTBD ISOL	B21-F028C	7(O)	1	5
MSL D CTMT OTBD ISOL	B21-F028D	8(O)	1	5
OTBD MSL A DR VLV	B21-F067A-A	5(O)	1	9
OTBD MSL B DR VLV	B21-F067B-A	6(O)	1	9
OTBD MSL C DR VLV	B21-F067C-A	7(O)	1	9
OTBD MSL D DR VLV	B21-F067D-A	8(O)	1	9
DRWL INST Line Isol Valve	D23-F591-B	109A(I)	6A	60
DRWL INST Line Isol Valve	D23-F592-A	109A(O)	6A	60
DRWL INST Line Isol Valve	D23-F593-B	109B(I)	6A	60
DRWL INST Line Isol Valve	D23-F594-A	109B(O)	6A	60
RHR SHUTDN CLG OTBD SUCT VLV	E12-F008-A	14(O)	3	40
RHR SHUTDN CLG INBD SUCT VLV	E12-F009-B	14(I)	3	40

Note*: Opening time for valve B21F016 must not be less than 15 seconds

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves#				
RHR HX "A" DISCH To S/P	E12-F011A-A	23(O)(b)	5	60
RHR HX "B" DISCH To S/P	E12-F011B-B	67(O)(b)	5	60
RHR "C" Test RTN To S/P	E12-F021-B	24(O)(b)	5	144
RHR "A" Test RTN To S/P	E12-F024A-A	23(O)(b)	5	144
RHR "B" Test RTN To S/P	E12-F024B-B	67(O)(b)	5	144
CTMT SPR "A" SPARGER INL VLV	E12-F028A-A	20(I)	5	90
CTMT SPR "B" SPARGER INL VLV	E12-F028B-B	21(I)	5	90
RHR "A" To CTMT Pool	E12-F037A-A	20(I)	3	74
RHR "B" To CTMT Pool	E12-F037B-B	21(I)	3	74
LPCS Test RTN To S/P	E21-F012-A	32(O)(b)	5	144
HPCS Test RTN To Supp Pool	E22-F023-C	27(O)(b)	6B	75
RCIC PMP SUCT FM S/P	E51-F031-A	28(O)(b)	4	60
RCIC STM SPLY DRWL INBD ISOL	E51-F063-B	17(I)	4	60
RCIC STM SPLY DRWL OTBD ISOL	E51-F064-A	17(O)	4	60
RCIC STM Line Warmup VLV	E51-F076-B	17(I)	4	60
RCIC TURB EXH TO SP	E51-F068-A	29(O)(c)	9	120
RCIC TURB EXH OTBD VAC BRKR	E51-F077-A	29(O)(c)	9	60
RCIC TURB EXH INBD VAC BRKR	E51-F078-B	75(O)	9	60
PURGE SPLY CTMT OTBD ISO VLV	E61-F009-(A)	65(O)	7	4
PURGE SPLY CTMT INBD ISOL VLV	E61-F010-(B)	65(I)	7	4

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP (a)	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves#				
PURGE EXH CTMT INBD ISOL VLV	E61-F056-(B)	66(I)	7	4
PURGE EXH CTMT OTBD ISOL VLV	E61-F057-(A)	66(O)	7	4
RWCU PMP SUCT DRWL INBD ISOL	G33-F001-B	87(I)	8	35
RWCU PMP SUCT CTMT OTBD ISOL	G33-F004-A	87(O)	8	35
RWCU BLWDN CTMT INBD ISOL	G33-F028-B	43(I)	8	35
RWCU BLWDN CTMT OTBD ISOL	G33-F034-A	43(O)	8	35
RWCU RTN CTMT OTBD ISOL	G33-F039-A	83(O)	8	35
RWCU RTN CTMT INBD ISOL	G33-F040-B	83(I)	8	35
RWCU PMP DISCH CTMT INBD ISOL	G33-F053-B	88(I)	8	35
RWCU PMP DISCH CTMT OTBD ISOL	G33-F054-A	88(O)	8	35
RWCU HX RTN To RWCU PMPS	G33-F252-B	87(I)	8	35
RWCU BKW RCV TK XFER To RADWST	G36-F101-(A)	49(O)	6A	60
RWCU BKW RCV TK XFER To RADWST	G36-F106-(B)	49(I)	6A	60
FPCC RTN VLV CTMT OTBD ISOL	G41-F028-A	57(O)	6A	60
CTMT FP DR CTMT OTBD ISOL	G41-F029-A	58(O)	6A	60
CTMT FP DR CTMT INBD ISOL	G41-F044-B	58(I)	6A	60
DRWL/CTMT SPLY AIR FM PURGE FANS	M41-F011-(A)	34(O)	7	4
DRWL/CTMT SPLY AIR FM PURGE FANS	M41-F012-(B)	34(I)	7	4

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves#				
CTMT CLG EXH To CTMT VENT	M41-F034-(B)	35(I)	7	4
CTMT CLG EXH To CTMT VENT	M41-F035-(A)	35(O)	7	4
CTMT INST LINE ISOL VALVE	M71-F594-B	109D(O)	6A	60
CTMT INST LINE ISOL VALVE	M71-F595-A	109D(I)	6A	60
CST WTR SPLY HDR To CTMT	P11-F075-(A)	56(O)	6A	60
REFUEL WTR XFER PMP SUCT FM SUPP POOL	P11-F130-(A)	69(O) (c)	6A	60
REFUEL WTR XFER PMP SUCT FM SUPP POOL	P11-F131-(B)	69(O) (c)	6A	60
M/U WTR TREAT CTMT ISOL VLV	P21-F017-A	86(0)	6A	60
M/U WTR TREAT CTMT ISOL VLV	P21-F018-B	86(I)	6A	60
CTMT FLOOR DR SMP DISCH	P45-F061-(B)	51(I)	6A	110
CTMT FLOOR DR SMP DISCH	P45-F062-(A)	51(O)	6A	110
CTMT EQUIP DR SMP DISCH	P45-F067-(B)	50(I)	6A	110
CTMT EQUIP DR SMP DISCH	P45-F068-(A)	50(O)	6A	110
CTMT CHEM WST SMP DISCH	P45-F098-(B)	84(I)	6A	8
CTMT CHEM WST SMP DISCH	P45-F099-(A)	84(O)	6A	8
AUX BLDG EQ/FL DR PMP BK To SUPP POOL	P45-F273-A	60(O)	6A	32
AUX BLDG EQ/FL DR PMP BK To SUPP POOL	P45-F274-B	60(O)	6A	32
SVC AIR SPLY HDR To CTMT	P52-F105-(A)	41(O)	6A	60

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP (a)	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves#				
INSTR AIR SPLY HDR To CTMT	P53-F001-(A)	42(O)	6A	60
INSTR AIR SPLY To ADS RCVRS	P53-F003-A	70(O)	6A	60
SPCU RTN FM CNDS PC FLTR	P60-F009-A	85(O)	6A	60
SPCU RTN FM CNDS PC FLTR	P60-F010-B	85(O)	6A	60
PCW RTN FM SMPL WTR CLRS/CTMT CLRS	P71-F148-(A)	39(O)	6A	60
PCW RTN FM SMPL WTR CLRS/CTMT CLRS	P71-F149-(B)	39(I)	6A	60
PCW SPLY To SMPL WTR CLRS/CTMT CLRS	P71-F150-(A)	38(O)	6A	60
DWCW RTN HDR FM CTMT	P72-F121-A	37(O)	6A	60
DWCW RTN HDR FM CTMT	P72-F122-A	36(O)	6A	60
DWCW RTN HDR FM CTMT	P72-F123-B	36(I)	6A	60

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
2. Manual Isolation Valves#		
FW INL SHUTOFF VLV	B21-F065A-A	9(O)(d)
FW INL SHUTOFF VLV	B21-F065B-A	10(O)(d)
JP POST-ACC SMPL CTMT OTBD ISOL	B33-F125-A	81(O)
JP POST-ACC SMPL CTMT INBD ISOL	B33-F126-B	81(I)
RW POST-ACC SMPL CTMT OTBD ISOL	B33-F127-A	47(O)
RW POST-ACC SMPL CTMT INBD ISOL	B33-F128-B	47(I)
DRIVE WTR FLTR OUTL VLV	C11-F083-A	33(O)
RHR PUMP "A" SUCT FM S/P	E12-F004A-A	11(O)(b)
RHR PUMP "B" SUCT FM S/P	E12-F004B-B	12(O)(b)
RHR PUMP "C" SUCT FM S/P	E12-F004C-B	13(O)(b)
RHR "A" SYS SHUTOFF VLV	E12-F027A-A	20(O)
RHR "B" SYS SHUTOFF VLV	E12-F027B-B	21(O)
RHR "A" INJ SHUTOFF VLV	E12-F042A-A	20(I)
RHR "B" INJ SHUTOFF VLV	E12-F042B-B	21(I)
RHR "C" INJ SHUTOFF VLV	E12-F042C-B	22(O)
MIN FLO To SUPP POOL	E12-F064A-A	23(O)(b)
RHR "B" MIN FLO TO S/P	E12-F064B-B	67((O)(b)
RHR "C" MIN FLO TO S/P	E12-F064C-B	24(O)(b)
RHR HX "A" OTBD VENT VLV	E12-F073A-A	77(O)(b)
RHR HX "B" OTBD VENT VLV	E12-F073B-B	48(O)(b)
RHR SMPL RTN ISOL VLV	E12-F346-B	71B(O)(c)
LPCS PMP SUCT FM S/P	E21-F001-A	30(O)(b)
LPCS INJ SHUTOFF VLV	E21-F005-A	31(O)
LPCS MIN FLO TO S/P	E21-F011-A	32(O)(b)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
2. Manual Isolation Valves#		
HPCS INJ SHUTOFF VLV	E22-F004-C	26(O)
HPCS MIN FLO TO SUPP POOL	E22-F012-C	27(O) (b)
HPCS PMP SUCT FM SUPP POOL	E22-F015-C	25(O) (b)
S POOL INST LINE ISOL VALVE	E30-F591A-A	116(O) (b)
S POOL INST LINE ISOL VALVE	E30-F591B-B	120(O) (b)
S POOL INST LINE ISOL VALVE	E30-F592A-A	114(O) (b)
S POOL INST LINE ISOL VALVE	E30-F592B-B	118(O) (b)
S POOL INST LINE ISOL VALVE	E30-F593A-A	113(O) (b) (c)
S POOL INST LINE ISOL VALVE	E30-F593B-B	117(O) (b) (c)
S POOL INST LINE ISOL VALVE	E30-F594A-A	115(O) (b) (c)
S POOL INST LINE ISOL VALVE	E30-F594B-B	119(O) (b) (c)
INBOARD VALVES TEST	E32-F001A-A	5(O)
INBOARD VALVES TEST	E32-F001E-A	6(O)
INBOARD VALVES TEST	E32-F001J-A	7(O)
INBOARD VALVES TEST	E32-F001N-A	8(O)
RCIC MIN FLO TO S/P	E51-F019-A	46(O) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F595A-(A)	107D(O) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F595B-(B)	107D(I) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F595C-(A)	106A(O) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F595D-(B)	106A(I) (b)
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F596A-(A)	108A(O) (b)
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F596B-(B)	108A(I) (b)
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F596C-(A)	105A(O) (b)
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F596D-(B)	105A(I) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F597A-(A)	107B(O) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F597B-(B)	107B(I) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F597C-(A)	106B(O) (b)
DRWL H_2 ANAL INST LINE ISOL VLV	E61-F597D-(B)	106B(I) (b)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
2. Manual Isolation Valves#		
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F598A-(A)	107B(O)(b)
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F598B-(B)	107B(I)(b)
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F598C-(A)	106E(O)(b)
CTMT H_2 ANAL INST LINE ISOL VLV	E61-F598D-(B)	106E(I)(b)
FLEX N_2 SUPPLY OB CTMT ISOL VLV	M41-F103	105D (O)
FLEX N_2 SUPPLY IB CTMT ISOL VLV	M41-F101	105D (I)
D/W INST LINE ISOL VLV	M71-F591A-A	101F(O)(b)
D/W INST LINE ISOL VLV	M71-F591B-B	102D(O)(b)
CTMT INST LINE ISOL VLV	M71-F592A-A	103D(O)(b)
CTMT INST LINE ISOL VLV	M71-F592B-B	104D(O)(b)
D/W INST LINE ISOL VLV	M71-F593-A	101C(O)(b)
SSW INL TO DRWL PURGE COMPR A (LOOP A)	P41-F159A-A	89(O)(b)(c)
SSW INL TO DRWL PURGE COMPR B (LOOP B)	P41-F159B-B	92(O)(b)(c)
SSW OTBD OUTL FM DRWL PURGE COMPR A (LOOP A)	P41-F160A-A	90(O)(b)(c)
SSW OTBD OUTL FM DRWL PURGE COMPR B (LOOP B)	P41-F160B-B	91(O)(b)(c)
SSW INBD OUTL FM DRWL PURGE COMPR A	P41-F168A-A	90(I)(b)(c)
SSW INBD OUTL FM DRWL PURGE COMPR B	P41-F168B-B	91(I)(b)(c)
CCW SPLY HDR TO CTMT	P42-F066-A	44(O)
CCW RTN HDR FM CTMT	P42-F067-A	45(O)
CCW RTN HDR FM CTMT	P42-F068-B	45(I)
AUX BLDG SLC WASTE STATION LINE	P48-F009	111C(I)
AUX BLDG SLC WASTE STATION LINE	P48-F010	111C(O)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

		SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
3.	Other Isolation Valves#		
	Check Valve	B21-F010A	9(I)(b)
	Check Valve	B21-F010B	10(I)(b)
	Check Valve	B21-F032A	9(O)(d)
	Check Valve	B21-F032B	10(O)(d)
	Inside Cont. CRD Header Stop Check Valve	C11-F122	33(I)
	Penetration Outboard Isolation	C41-F150	61(O)
	Penetration Inboard Isolation	C41-F151	61(I)
	RELIEF VLV	E12-F005	76B(O)
	Jockey Pump Relief VLV	E12-F017A	11(O)(b)
	Jockey Pump Relief VLV	E12-F017B	12(O)(b)
	Jockey Pump Relief VLV	E12-F017C	13(O)(b)
	Relief VLV	E12-F025A	20(I)
	Relief VLV	E12-F025B	21(I)
	RHR PUMP "C" Relief Valve DISCH HDR	E12-F025C	71B(O)(b)
	RHR "C" Testable CHK VLV	E12-F041C-B	22(I)
	"A" Loop LPCI Fill From CRWST	E12-F044A	20(I)
	Fill & Flush Line To "B" Spray HDR	E12-F044B	21(I)
	RELIEF VLV	E12-F055A	77(O)(b)
	RELIEF VLV	E12-F055B	48(O)(b)
	RWCU Relief Valve	G33-F267	87(O)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
3. Other Isolation Valves#		
"A" Spray HDR Vent	E12-F107A	20(I)(b)
"B" Spray HDR Vent	E12-F107B	21(I)(b)
"B" LOOP TEST LINE DRN	E12-F212	67(O)(b)(e)
"B" LOOP TEST LINE DRN	E12-F213	67(O)(b)(e)
"A" RHR MIN FLOW LINE DRN To DRW	E12-F227	23(O)(b)(e)
"A" RHR MIN FLOW LINE DRN To DRW	E12-F228	23(O)(b)(e)
"C" Loop LPCI Line DRN	E12-F234	22(O)(b)
"B" RHR MIN FLOW DRN To DRW	E12-F249	67(O)(b)(e)
"B" RHR MIN FLOW DRN To DRW	E12-F250	67(O)(b)(e)
"A" LOOP TEST LINE DRN	E12-F259	23(O)(b)(e)
"A" LOOP TEST LINE DRN	E12-F260	23(O)(b)(e)
"A" LOOP TEST LINE DRN	E12-F261	23(O)(b)(e)
"A" LOOP TEST LINE DRN	E12-F262	23(O)(b)(e)
"B" LOOP TEST LINE DRN To DRW	E12-F276	67(O)(b)(e)
"B" LOOP TEST LINE DRN To DRW	E12-F277	67(O)(b)(e)
"C" RHR MIN FLOW LINE DRN	E12-F280	24(O)(b)(d)
"C" RHR MIN FLOW LINE DRN	E12-F281	24(O)(b)(d)
RHR JKY PMP "A" DISCH BLOCK VLV	E12-F290A-A	23(O)(b)
RHR JKY PMP "B" DISCH BLOCK VLV	E12-F290B-B	67(O)(b)
Shutdown Cooling Suction Stop Check	E12-F308	14(I)
DRAIN	E12-F334	67(O)(b)(c)
DRAIN	E12-F335	67(0)(b)(c)
DRAIN	E12-F338	23(O)(b)(c)
DRAIN	E12-F339	23(O)(b)(c)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
3. Other Isolation Valves#		
PASS RETURN To SUPP POOL CHECK	E12-F406	71B(I) (c)
LPCS TESTABLE CHK VLV	E21-F006-(A)	31(I)
LPCS RELIEF VALVE VENT HEADER	E21-F018	71A(O)(b)
JOCKEY PMP SUCT RELIEF	E21-F031	30(O)(b)
LPCS INJECTION HDR DRYWELL VENT	E21-F200	31(I) (b)
F006 UPSTREAM DRAIN	E21-F207	31(I) (b)
MIN FLOW HDR DRAIN To DRW	E21-F217	32(O) (b)(d)
MIN FLOW HDR DRAIN To DRW	E21-F218	32(O) (b)(d)
HPCS TESTABLE CHECK	E22-F005-(C)	26(I)
HPCS SUCT HDR PRESS RELIEF VLV	E22-F014	25(O)(b)
HPCS DISCH HDR PRESS RELIEF VLV	E22-F035	27(O)(b)
HPCS INJECTION LINE VENT SHUTOFF	E22-F201	26(I) (b)
HPCS INJECTION LINE DRN SHUTOFF	E22-F218	26(I) (b)
TEST LINE To SUPP POOL DRN ISOL	E22-F301	27(O) (b)(e)
TEST LINE To SUPP POOL DRN SHUTOFF	E22-F302	27(O) (b)(e)
RCIC TURB EXH CHK VLV	E51-F040	29(O) (c)
MIN FLOW LINE To S/P DRAIN	E51-F251	46(O) (b)(c)
MIN FLOW LINE To S/P DRAIN	E51-F252	46(O) (c)
FUEL TRANSFER TUBE	F11-E015	4(I)(c)
CHECK VLV	G41-F040	57(I)
RWST SUPPLY/RETURN To CTMT POOLS	G41-F053	54(O)
DRAIN/FILL LINE FROM REFUEL WATER TK ISOLATION	G41-F201	54(I)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
3. Other Isolation Valves#		
CTMT OTBD ISOL	M61-F014	110A(O) (b)
CTMT INBD ISOL	M61-F015	110A(I) (b)
CTMT OTBD ISOL	M61-F016	110F(O) (b)
CTMT INBD ISOL	M61-F017	110F(I) (b)
CTMT OTBD ISOL	M61-F018	110C(O) (b)
CTMT INBD ISOL	M61-F019	110C(I) (b)
CHECK VLV	P11-F004	56(I)
CHECK VLV	P41-F169A	89(I) (b) (c)
CHECK VLV	P41-F169B	92(I) (b) (c)
CHECK VLV	P42-F035	44(I) (b)
CHECK VLV	P52-F122	41(I)
CHECK VLV	P53-F002	42(I)
CHECK VLV	P53-F006	70(I)
CHECK VLV	P71-F151	38(I)
CHECK VLV	P72-F165	37(I)
Safety Relief Valve	G33-F263	88(I)
Safety Relief Valve	G33-F264	43(O)
Safety Relief Valve	P21-F390	86(I)
BLIND FLANGES		
Cont. Leak Rate Sys.	NA	40(I)(O)
Cont. Leak Rate Sys.	NA	82(I)(O)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
4. Test Connections		
Test Line ISOL	B21-F025A	5(O) (b)
Test Line ISOL	B21-F025B	6(O) (b)
Test Line ISOL	B21-F025C	7(O) (b)
Test Line ISOL	B21-F025D	8(O) (b)
FW Line Test	B21-F030A	9(O) (b)
FW Line Test	B21-F030B	10(O) (b)
FW Line Test	B21-F063A	9(O) (b)
FW Line Test	B21-F063B	10(O) (b)
CRD Outside Cont. Penetration T/C	C11-F128	33(O) (b)
Penetration T/C	C41-F152	61(O) (b)
Upstream T/C F008	E12-F002	14(O) (b)
"C" Loop LPCI Line T/C	E12-F056C	22(O) (b)
RO-D003 T/C	E12-F304	24(O) (b) (c)
RO-D003 T/C	E12-F311	24(O) (b) (c)
Vent	E12-F321	67(O) (b) (c)
Drain	E12-F322	23(O) (b) (c)
Drain	E12-F331	67(O) (b) (c)
Drain	E12-F336	23(O) (b) (c)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
4. Test Connections		
Drain	E12-F348	23(O) (b) (c)
Drain	E12-F349	23(O) (b) (c)
Drain	E12-F350	67(O) (b) (c)
Vent	E12-F351	67(O) (b) (c)
T/C Isolation	E12-F408	71B(O) (b) (c)
T/C Isolation	E12-F409	71B(I) (b) (c)
T/C VLV FOR F024B	E12-F430B	67(O) (b) (c)
T/C VLV FOR F024B	E12-F432B	67(O) (b) (c)
T/C VLV FOR F064B	E12-F434B	67(O) (b) (c)
T/C VLV FOR F064B	E12-F436B	67(O) (b) (c)
T/C VLV FOR F004C	E12-F439C	13(O) (b) (c)
T/C VLV FOR F004C	E12-F440C	13(O) (b) (c)
T/C VLV FOR F004C	E12-F441C	13(O) (b) (c)
T/C VLV FOR F004C	E12-F442C	13(O) (b) (c)
T/C VLV FOR F008	E12-F445	14(O) (b)
T/C VLV FOR F008	E12-F446	14(O) (b)
LPCS Injection HDR T/C	E21-F013	31(O) (b)
Minflow/Test Return Flow T/C	E21-F221	32(O) (b) (c)
Minflow/Test Return Flow T/C	E21-F222	32(O) (b) (c)
HPCS PMP DISCHARGE TO RPV	E22-F021	26(O) (b)
TEST LINE To S/P T/C	E22-F303	27(O) (b) (c)
TEST LINE To S/P	E22-F304	27(O) (b) (c)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

		SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
4.	Test Connections		
	SUPP POOL SUCT VLV EQUALIZATION	E22-F800	25(O) (b)
	SUPP POOL SUCT VLV EQUALIZATION	E22-F801	25(O) (b)
	T/C UPSTREAM MOV-F064 ISOLATION	E51-F072	17(O) (b)
	T/C UPSTREAM MOV-F068 ISOLATION	E51-F212	29(O) (b) (c)
	DRAIN FOR F068	E51-F257	29(O) (b) (c)
	DRAIN FOR F077	E51-F258	29(O) (b) (c)
	T/C VLV FOR F031	E51-F269	28(O) (b) (c)
	T/C VLV FOR F031	E51-F270	28(O) (b) (c)
	T/C VLV FOR F031	E51-F272	28(O) (b) (c)
	T/C VLV FOR F031	E51-F273	28(O) (b) (c)
	TEST CONNECTION	E61-F017	65(O) (b)
	TEST LINE ISOLATION	G33-F002	87(O) (b)
	TEST LINE ISOLATION	G33-F055	83(O) (b)
	TEST LINE ISOLATION	G33-F075	83(I) (b)
	TEST ISOLATION DOWNSTREAM F054	G33-F061	88(O) (b)
	TEST LINE ISOLATION	G33-F077	88(I) (b)
	TEST LINE ISOLATION	G33-F070	43(O) (b)
	DRAIN OFF 8" SUPPLY To POOLS	G41-F340	57(I) (b)
	LOWER CTMT AIRLOCK TEST CONNECTION	M23-FX011	3(I) (f)
	LOWER CTMT AIRLOCK TEST CONNECTION	M23-FX012	3(O) (f)
	LOWER CTMT AIRLOCK TEST CONNECTION	M23-FX013	3(I) (f)
	LOWER CTMT AIRLOCK TEST CONNECTION	M23-FX014	3(O) (f)
	UPPER CTMT AIRLOCK TEST CONNECTION	M23-FX015	2(I) (f)
	UPPER CTMT AIRLOCK TEST CONNECTION	M23-FX016	2(O) (f)
	UPPER CTMT AIRLOCK TEST CONNECTION	M23-FX017	2(I) (f)
	UPPER CTMT AIRLOCK TEST CONNECTION	M23-FX018	2(O) (f)

TABLE TR3.6.1.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

	SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
4. Test Connections		
AUX SEAL VALVE FOIL OUTLET DUCT TEST ISOL	M41-F042	34(O)(b)
AUX ISOL VLV F035 INLET DUCT TEST ISOL	M41-F051	35(O)(b)
EXH FILTER TRAIN E61-F057 INLET DUCT DRN	M41-F054	66(O)(b)
FLEX N2 SUPPLY TEST CONNECTION VLV	M41-F105	105D (O)
T/C ISOL	M61-F009	40(I)(b)
T/C ISOL	M61-F010	82(I)(b)
CTMT CNDS XFER HDR T/C	P11-F095	56(O)(b)
TEST CONN	P11-F132	69(O)(b)(c)
TEST CONN	P11-F425	69(O)(b)(c)
"A" DRYWELL PURGE CPRSR CLR SUPPLY TEST CONN (LOOP "A")	P41-F163A	89(O)(b)(c)
"B" DRYWELL PURGE CPRSR CLR INL TEST CONN (LOOP "B")	P41-F163B	92(O)(b)(c)
CCW SPLY OTBD ISOL T/C	P42-F161	44(O)(b)
CCW RTN LINE T/C SHUTOFF	P42-F162	45(I)(b)
AUX BLDG FLR DR XFER TK PUMP BACK To SUPP POOL T/C	P45-F275	60(O)(b)
AUX BLDG FLR DR XFER TK PUMP BACK To SUPP POOL T/C	P45-F290	60(O)(b)
CTMT & DRYWELL SERV AIR HDR T/C	P52-F258	41(O)(b)
TEST CONNECTION	P53-F036	42(O)(b)
PRESSURE POINT	P53FX003	42(O)(b)
ADS MAKEUP ISOLATION	P53-F043	70(O)(b)
PRESSURE INDICATION	P53FX004	70(O)(b)

TABLE TR3.6.1.3-1 (Continued)

		SYSTEM AND VALVE NUMBER	PENETRATION NUMBER
4.	Test Connections		
	SUPPR POOL CLEANUP RETURN HDR TEST CONN	P60-F011	85(O)(b)
	F010 T/C	P60-F034	85(O)(b)
	AUX BLDG SEC LOOP SPLY OB T/C	P71-F232	38(O)(b)
	PRESSURE POINT	P71FX359	38(O)(b)
	AUX BLDG SEC LOOP OB T/C	P71-F246	39(O)(b)
	PRESSURE POINT	P71FX358	29(O)(b)
	TEST CONNECTION	P72-F167	37(O)(b)

(a) See the Bases for Technical Specification 3.3.6.1 for isolation signal(s) that operates each valve group.

(b) Type C testing is not required.

(c) Hydrostatically tested with water to 1.10 P_a, 13.31 psig, when applicable.

(d) Hydrostatically tested by pressurizing system to 1.10 P_a, 13.31 psig, when applicable.

(e) Hydrostatically tested during system functional tests, when applicable.

(f) These valves have a design function as primary containment isolation when the inner airlock door is nonfunctional or during performance of airlock barrel testing or pnuematic tubing testing or at any time the inner airlock door/bulkhead is breached.

(g) Valve gagged closed.

\# The "-A, -B, -C, -(A), -(B), -(C)" designators on the valve numbers indicate associated electrical divisions.