Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295016	AA2.02
	Importance Rating		4.3

Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor water level

Proposed Question: SRO Question # 1

While at rated conditions, a fire in the main control room required that the control room be evacuated. The following actions were taken prior to evacuating:

- Scramming the reactor and verifying all rods inserted
- Tripping the main turbine and all reactor feed pumps
- Closing the MSIVs

Eight minutes later, conditions are:

- All SRVs have been placed in LOCAL at their ASPs
- HPCI cannot be started from the Alternate Shutdown Panels (ASPs)
- No other actions have been taken
- RPV pressure is 600 psig and lowering
- RPV level is -35 inches and lowering
- Local torus water temperature is 95 degrees and rising
- Drywell pressure is 1.0 psig and steady

Which one of the following is NOW required to be directed IAW PNPS 2.4.143, Shutdown From Outside the Control Room?

Direct that:

A. within the next 16 minutes RCIC be placed in injection mode IAW Appendix B.

B. torus cooling be placed in service IAW Appendix D and if torus water temperature reaches 120 degrees, have all SRVs placed in OPEN.

C. within the next 7 minutes inject using RHR IAW Appendix D by lowering RPV pressure below injection pressure by placing all SRVs placed in OPEN.

D. the breaker alignment of Appendix F be completed and one low pressure ECCS pump started within the next 16 minutes. Then have one SRV placed in OPEN.

Proposed Answer: A

Explanation (Optional):

A. Correct: In order to maintain RPV level above the TAF in the event of spurious SRV actuation following a fire, injection must be commenced within 24 minutes to support the engineering analysis. RCIC (or HPCI) is first used to maintain level. Since 8 minutes have already elapsed and spurious SRV actuation is indicated (RPV pressure 600 psig and lowering and torus temperature rising), RCIC injection must commence within the next 16 minutes.

B. Incorrect: Per the engineering analysis torus cooling is NOT assumed to be placed into operation until the two hour point. Plausible in that the procedure directs that if torus temperature reaches 120 degrees then the RPV should be rapidly depressurized. However the requirement to stay within normal cool down rates still applies. Opening all SRVs would exceed the cooldown rate.

C. Incorrect: Up to 16 minutes is available to start injection. Plausible in that the seven minutes is based on another time sensitive operation that must be completed within 15 minutes from the time of the transient. (15 minutes – the elapsed time of 8 minutes). This operation involves taking LOCAL control of all SRVs which has already been accomplished.

D. Incorrect: Per the 2.4.143 flowchart, injection is first attempted using RCIC. Additionally, if RPV level had lowered to -125 inches within the 16 minutes, then all SRVs are required to be opened.

Technical Reference(s): Procedure Cautions on page 9 of PNPS 2.4.143 and associated procedure discussion section. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295005	AA2.08
	Importance Rating		3.3

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Electrical distribution status

Proposed Question: SRO Question # 2

The plant is at rated conditions with all equipment operable.

Then, the Main Turbine trips and the reactor scrams. One minute later the BOP operator reports that:

- 4160 VAC bus A5 is de-energized
- 4160 VAC bus A6 is powered from the "B" EDG
- The following annunciators have alarmed:
  - C3L-A7, START UP XFMR LOCKOUT
  - C3LC-A1, A-5 LOCKOUT
  - C3L-B1, EDG A GENERATOR BKRTRIP/INOP

Per PNPS 2.4.16, Distribution Alignment Electrical Distribution Malfunctions, which one of the following is required?

A. Attempt to manually re-energize bus A5 with the Shutdown Transformer IAW PNPS 2.4.16.

B. Start the SBODG and re-energize bus A5 IAW PNPS 2.2.146, Station Blackout Diesel Generator.

C. Attempt ONE reset of the A-5 lockout IAW PNPS 1.3.11, Reset Of Lockout Relays. If the lockout resets, re-energize bus A5 with the "A" EDG.

D. Do NOT attempt to reset the A-5 lockout. Stabilize plant conditions IAW PNPS 2.4.16 and PNPS 2.4-A5, Loss of Electrical Bus A5.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: With the A5 bus lockout present PNPS 2.4.16 prohibits placing the Shutdown Transformer (SDT) on bus A5. Plausible if the candidate believes the SDT just failed to auto close as designed.

B. Incorrect: With the A5 bus lockout present PNPS 2.4.16 prohibits placing the Station Blackout Diesel on A5.

C. Incorrect: PNPS 2.4.16 directs that maintenance must be directed to investigate and clear the lockout. Additionally, PNPS 1.3.11 prohibits Operations from resetting a lockout unless Nuclear Safety is in question. With the existing plant conditions, nuclear safety is not in question.

D. Correct: With the A5 bus lockout present PNPS 2.4.16 prohibits re-energizing bus A5. PNPS 2.4-A5 then provides actions to stabilize the plant with the loss of the safety related bus.

Technical Reference(s): PNPS 2.4.16 Flowchart (Attach if not previously provided)
PNPS 1.3.11, Reset Of Lockout Relays

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295038	EA2.03
	Importance Rating		4.3

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Radiation Levels

Proposed Question: SRO Question # 3

The emergency plan is being implemented during an event involving a high off-site release. Conditions are as follows:

- The Shift Manager has just declared a Site Area Emergency
- The Digital Notification Network has failed
- You are utilizing the Backup Emergency Community Offsite Notification System (BECONS) to notify off-site agencies of the event
- There is NO evidence of a Turbine Building or unmonitored release

IAW EPIP-100, EMERGENCY CLASSIFICATION AND NOTIFICATION, which of the following radiation indications would require that you report that a release of radioactivity IS IN PROGRESS?

Note: The indications below have existed for at least 20 minutes.

Indication 1: Main Stack High Range Monitor (RI-1001-608) is reading 1 R/hr

Indication 2: Reactor Building Vent Exhaust Monitors (RM-1705-32A and B) are reading 2.5E+5 cps

Indication 3: Standby Gas Treatment System Exhaust Radiation Monitor (RIS-1705-9) is reading 1500 mR/hr

A. Indication 1 only

B. Indication 2 only

C. Indications 1 and 3

D. Indications 2 and 3

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Indication #1 is below the lowest value of this monitor (2R/Hr) associated with any EAL of Subcategory A1.

B. Correct: Per EP-IP-100: For purposes of offsite notification, any release of radioactivity is considered an emergency radioactive release in progress which: Meets any EAL of Subcategory A1, Offsite Rad Conditions OR Involves an actual or suspected Turbine Building or unmonitored release which is associated with the emergency event. Indication # 2 is above the EAL threshold for Alert AA1.1 (1E+5 cps)

C. Incorrect: Indication #1 is below the lowest value of this monitor (2R/Hr) associated with any EAL of Subcategory A1. Also indication #3 is not used for any EAL of Subcategory A1. Plausible in that Standby Gas Treatment discharges to the main stack.

D. Incorrect: Indication #3 is not used for any EAL of Subcategory A1. Plausible in that Standby Gas Treatment discharges to the main stack.

Technical Reference(s): EP-IP-100 EMERGENCY CLASSIFICATION AND NOTIFICATION, page 7 (Attach if not previously provided)

EP-IP-100.1 EMERGENCY ACTION LEVELS (EALs), Attachment 9.1

Proposed References to be provided to applicants during examination: EP-IP-100.1 EMERGENCY ACTION LEVELS, Attachment 9.1

Learning Objective: (As available)

Question Source:	Bank #	X	Updated for new Eals
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	
	55.43	4

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295018 2.4.4	
	Importance Rating		4.7

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. Partial or Total Loss of CCW

Proposed Question: SRO Question # 4

With the plant at full power and a normal configuration on the Salt Service Water (SSW) system, alarm RBCCW PUMP AREA LEAKAGE, C904L-F6, is received.

The control room crew notes the following changes in SSW and RBCCW indications on control room panel C-1:

	Before	**After**
SSW loop "A" pressure as read by PI-3828	30 psig	20 psig
SSW loop "B" pressure as read by PI-3829	30 psig	20 psig
SSW loop "A" flow as read by FI-6240	500 gpm	800gpm
SSW loop "B" flow as read by FI-6241	1750 gpm	1450 gpm
RBCCW Loop A Temp as read on TR 3835/3836	70 degrees	68 degrees and lowering
RBCCW Loop B Temp as read on TR 3835/3836	70 degrees	72 degrees and rising

Which one of the following procedures is entered in order to mitigate the effects of the above conditions and the actions required by that procedure?

A. PNPS 2.4.42, LOSS of RBCCW, Attachment 2, RBCCW Leak Outside the Drywell. Cross-connect RBCCW with the "A" loop supplying.

B. PNPS 2.4.43, LOSS OF ONE SALT SERVICE WATER LOOP. Cross connect RBCCW, "A" loop supplying and isolate the "B" SSW loop.

C. PNPS 2.4.43, LOSS OF ONE SALT SERVICE WATER LOOP. Cross connect RBCCW with the "B" loop supplying and isolate the "A" SSW loop.

D. PNPS 2.4.42, LOSS of RBCCW, section 4.3 Loss of Temperature Control Function for an RBCCW Heat Exchanger. Manually control loop temperatures by adjusting the salt service water heat exchanger outlet valves.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The leak is on SSW not RBCCW. Plausible in that the RBCCW leakage annunciator is in alarm. However this alarm will come in for either a SSW leak or a RBCCW leak.

B. Incorrect: The leak is on the "A" loop. Plausible in that flow is dropping in the "B" loop. However because the loops are cross connected, the leak is robbing flow from loop "B".

C. Correct: With flow going up in the "A" loop and lowering in the "B" loop, the leak is on the "A" side. Temperature is dropping because the leak is on the outlet of the heat exchanger. Per PNPS 2.4.43, Loss of one SSW Loop, these are the actions to be taken in the case of a pipe break.

D. Incorrect: The SSW leak needs to be isolated to prevent flooding the area. This action will not isolate the leak.

Technical Reference(s): PNPS 2.4.43, Loss of one SSW Loop
P&ID for SSW
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031	2.4.2
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions (Reactor Low Water Level)

Proposed Question: SRO Question # 5

The plant is at rated conditions. The Torus is being vented through "A" Standby Gas train using the 2 inch Torus vent valves.

While venting, a Feedline Break inside the Turbine Building results in a reactor scram. Five minutes later the following indications exist:

- All secondary containment dampers are CLOSED
- Both 2 inch Torus vent valves are OPEN
- "A" Standby Gas Treatment Inlet and Outlet dampers are OPEN
- "B" Standby Gas Treatment Inlet and Outlet dampers are CLOSED
- RCIC and HPCI have auto started and are injecting
- RPV Level is 0 inches and cannot be raised further
- Drywell pressure is 1.5 psig and slowly lowering

Then....

- A leak is reported on the discharge of the HPCI Pump
- There is 7 inches of water on the HPCI pump room floor.
- There are no abnormal Primary and Secondary Containment radiation levels for the existing plant conditions.

Which one of the following describes the required EOP actions?

A. Close the 2 inch Torus vent valves IAW PNPS 5.3.35.1, Transient Hardcard Response. Manually start "B" Standby Gas Treatment IAW PNPS 2.2.50, Standby Gas Treatment System.

B. Open the breakers for the reactor building floor and equipment sump pumps IAW EOP-04, Secondary Containment Control.
Manually start "B" Standby Gas Treatment IAW PNPS 2.2.50, Standby Gas Treatment System.

C. Open the breakers for the reactor building floor and equipment sump pumps IAW EOP-04, Secondary Containment Control.
Defeat the Secondary Containment Isolation IAW PNPS 5.3.21 Bypassing Selected

Interlocks, and restore Secondary Containment Ventilation.

D. Close the 2 inch Torus vent valves IAW PNPS 5.3.35.1, Transient Hardcard Response for Operating Crews.
Defeat the Secondary Containment Isolation IAW PNPS 5.3.21 Bypassing Selected Interlocks, and restore Secondary Containment Ventilation.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Manually starting "B" Standby Gas Treatment is not required. The "B" Train shuts down following an auto initiation after a 65 second time delay.

B. Incorrect: Opening the sump pump breakers is not required. Plausible in that the sump pumps discharge outside the secondary containment. However this action is not required unless there are indications of significant core damage. None of the symptoms of core damage as described in PNPS 5.3.35 are present. Also, manually starting "B" Standby Gas Treatment is not required. The "B" Train shuts down following an auto initiation after a 65 second time delay.

C. Incorrect: Opening the sump pump breakers is not required. Plausible in that the sump pumps discharge outside the secondary containment. However this action is not required unless there are indications of significant core damage. None of the symptoms of core damage as described in PNPS 5.3.35 are present.

D. Correct: EOP-01 entry is required because level is below + 12 inches. EOP-01 directs that isolations be verified as having gone to completion via step EOP-01, L1. The Torus Vent Valves should have closed when RPV level dropped below +12 inches. EOP-04 entry is required because of the high water level in the HPCI pump room. EOP-04 step SC-1 directs that if ALL of following exist:
- RB H&V isolates
- RB Refuel Floor exhaust radiation level is below 16 mR/hr
- A primary system is NOT discharging into secondary containment
- NO high radiation levels in the RB (as indicated by area radiation monitors or other sources)

Then, reset secondary containment isolation AND restart RB H&V (defeat high drywell pressure and low RPV water level isolation interlocks if necessary, procedure 5.3.21). None of the conditional statements are met since there is not a Primary System discharging inside the secondary containment (a leak on the HPCI pump discharge is not considered a Primary System). Restoring secondary containment ventilation would require defeating the low level isolation signal.

Technical Reference(s):	EOP-04, Secondary Containment Control PNPS 5.3.35, OPERATIONS MANAGEMENT EMERGENCY AND TRANSIENT RESPONSE	(Attach if not previously provided)

EXPECTATIONS FOR
OPERATING CREWS, page 14

5.3.35.1, Transient Hardcard
Response, Attachment 2

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295006	2.4.20
	Importance Rating		4.3

Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes. (Scram)

Proposed Question: SRO Question # 6

Following a scram and an extended loss of off-site power the following conditions exist:

- Electrical power has been restored using the Shutdown Transformer.
- No LOCA exists
- RPV water level is 20 inches and steady
- RPV pressure is 235 psig and lowering slowly
- Drywell pressure is 2.9 psig and rising slowly
- Drywell temperature is 273 degrees and rising slowly

Which one of the following actions is required to be taken in order to help preserve RPV level indication?

A. Enter EOP-17, Emergency RPV Depressurization, and depressurize the RPV.

B. Enter PNPS 2.2.19.5, RHR Modes of Operation for Transients, Attachment 5, and initiate drywell spray.

C. Enter PNPS 2.4.44, Loss of Drywell Cooling, and override the diesel load shed signal and restart the Drywell cooling fans.

D. Enter PNPS 5.4.6, Primary Containment Venting and Purging Under Emergency Conditions and commence purging the primary containment with nitrogen, overriding isolation signals as required.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: With reactor pressure currently 235 psig, drywell temperature is still well below the associated saturation temperature and RPV water level indication is currently still available as described in Caution 1 of the EOPs. If an Emergency Depressurization (ED) were performed, the saturation curve would be exceeded when pressure lowered

to ~ 60 psig. Conducting an ED at this point would increase the potential for a loss of level indication. Plausible in that EOP-03 step DT-5 does direct that an Emergency Depressurization be performed if temperature cannot be restored and maintained below 280 degrees. However, this condition has not yet occurred since the action to restore drywell cooling has not yet been attempted and temperature is still below 280 degrees.

B. Incorrect: Drywell spray cannot be used because with the high drywell temperature and low drywell pressure the drywell spray initiation limit is NOT met. Plausible in that drywell sprays are used to maintain containment parameters during LOCA conditions.

C. Correct: As discussed in PNPS 2.4.44, if Drywell cooling is lost due to a loss of AC power, Drywell pressure will exceed 2.2 psig, in turn causing a load shed signal to be generated. When power is restored with the Shutdown Transformer, Blackout Diesel, or Emergency Diesel Generators, the load shed signal will remain, preventing Drywell cooling fans from starting and thereby further aggravating the high Drywell temperature condition. If Drywell temperature continues to increase and cannot be maintained less than 280°F, then EOP-03, Primary Containment Control, will direct that EOP-17, Emergency RPV Depressurization, be entered and the plant be depressurized. This can in turn result in a loss of level indication when the reference legs reach saturated conditions. In order to prevent this chain of events, the Operator is directed to override the diesel load shed signal to the Drywell cooling fans.

D. Incorrect: There is no direction to commence a nitrogen purge to reduce drywell temperature. Plausible in that this action would introduce cold nitrogen gas to the containment.

Technical Reference(s): 2.4.44, pg 4
EOP-3
EOP-11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295030	EA2.04
	Importance Rating		3.7

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Drywell/ suppression chamber differential pressure: Mark-I&II

Proposed Question: SRO Question # 7

A small break LOCA has occurred. The following plant conditions exist:

- Drywell sprays are in service
- Drywell pressure is 10 psig and lowering
- Drywell to Torus Differential pressure indicator, PID-5021, on Panel C904 is pegged high at greater than 3.0 psid

Regarding the relationship between the above conditions and Torus water level:

(1) At which one of the Torus water levels below will the Drywell to Torus Differential pressure indicator, PID-5021, indicate 0.0 psid

AND

(2) What action will then be required by EOP-03, Primary Containment control?

If Torus water level.....

A. (1) Lowers to 88 inches
(2) Secure drywell sprays IAW EOP-03 Torus Level leg

B. (1) Lowers to 88 inches
(2) Emergency Depressurize the RPV IAW EOP-17

C. (1) Rises to 178 inches
(2) Secure drywell sprays IAW EOP-03 Torus Level leg

D. (1) Rises to 178 inches
(2) Emergency Depressurize the RPV IAW EOP-17

Proposed Answer: B

Explanation (Optional):

A. Incorrect: At <90 inches, the downcomers are exposed. EOP-17 torus water level leg, step 15, specifies that Emergency RPV Depressurization is required when torus level cannot be maintained above 90 inches because the pressure suppression function has been lost.

B. Correct: The water level in the downcomers allows for a differential pressure between the drywell and torus atmospheres. When level lowers to <90 inches, the downcomers are exposed and the two atmospheres will equalize. EOP-17 torus water level leg, step 15, specifies that Emergency RPV Depressurization is then required because the pressure suppression function has been lost.

C. Incorrect: The differential pressure will not equalize as level rises. It may lower when the vacuum breakers lift but will not go to 0.0 as the vacuum breakers will reclose prior to that point. Plausible in that at 180 inches there is an EOP action to secure the drywell sprays because the vacuum breakers are covered.

D. Incorrect: The differential pressure will not equalize as level rises. It may lower when the vacuum breakers lift but will not go to 0.0 as the vacuum breakers will reclose. Plausible in that the there is an action to Emergency Depressurize if level cannot be restored and maintained below 175 inches.

Technical Reference(s): EOP-03, Primary Containment control (Attach if not previously provided)

EOP-03 lesson plan, page 37

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295008	AA2.02
	Importance Rating		3.4

Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: Steam flow/feedflow mismatch

Proposed Question: SRO Question # 8

The plant is at 100% power and 60 Mlbm/hr core flow when the following indications occur.

- Alarm REACTOR WTR LEVEL HI, C905R-C7 annunciates
- RPV water level is 33 inches and rising
- Total Steam Flow as indicated on recorder PR/FR-640-27 is 8 Mlbm/hr and slowly rising
- Total Feed Flow as indicated on recorder LR/FR-640-26 is steady at 0 Mlbm/hr
- 15 seconds later alarms MG A and MG B SPEED LIMITER ON, C904RC B-5 and C904R B-4 both annunciate

Based on the above indications which of the following is correct regarding the required actions to address the:

(1) RPV water level transient

AND

(2) The Recirc Flow Transient? Assume the Recirc Flow Transient goes to completion. A Power to Flow map is provided for your use.

A. (1) Enter PNPS 2.4.49, Feedwater Malfunctions and have the Master Feed Reg Valve Controller placed in manual IAW the Subsequent Actions.
(2) Enter PNPS 2.4.165, Reactor Core Instability and direct the RPR be inserted as required. The reactor is operating in the Exclusion Region of the Power to Flow Map.

B. (1) Enter PNPS 2.4.49, Feedwater Malfunctions and dispatch operators to lock both Feed Reg Valves in the condenser compartment IAW Section 4.5.
(2) Enter PNPS 2.4.165, Reactor Core Instability and direct the RPR be inserted as required. The reactor is operating in the Exclusion Region of the Power to Flow Map.

C. (1) Enter PNPS 2.4.49, Feedwater Malfunctions and have the Master Feed Reg Valve Controller placed in manual IAW the Subsequent Actions.
   (2) Enter PNPS 2.4.20, Reactor Recirc Speed or Flow Control Malfunction, Reset the Runback, and verify Decay Ratios. The reactor is operating in the Buffer Zone of the Power to Flow Map.

D. (1) Enter PNPS 2.4.49, Feedwater Malfunctions and dispatch operators to lock both Feed Reg Valves in the condenser compartment IAW Section 4.5.
   (2) Enter PNPS 2.4.20, Reactor Recirc Speed or Flow Control Malfunction, Reset the Runback, and verify Decay Ratios. The reactor is operating in the Buffer Zone of the Power to Flow Map.

Proposed Answer: A

Explanation (Optional):

A. Correct: Feedwater Level Control (FWLC) is in 3-element when at full power. The indications provided are consistent with a failure of the Total Feedflow signal. The resultant feed flow steam flow mismatch will cause FWLC to open the feed reg valves causing level to rise. Placing the Master Controller in manual will return the controller output to its previous setting prior to the failure (the controllers are kept balanced). PNPS 2.4.49 Subsequent actions direct placing the Master controller in manual as required to control level. The loss of total feed flow signal also causes the recirc pumps to run back to minimum speed. Given the initial operating point on the power to flow map, this would place the plant in the exclusion region. PNPS 2.4.165 directs exiting the exclusion region by either inserting rods or raising core flow. Since the runback will prevent increasing core flow, the RPR must be inserted.

B. Incorrect: Following the loss of the Total Feed Flow signal the feed reg valves opened further causing level to rise. Locking the valves in this position would not terminate the level rise.

C. Incorrect: Following the runback the core is operating in the exclusion region. Plausible if the candidate believes that the FWLC failure will cause a #2 Runback (vice #1 runback). A #2 Runback will lower recirc pump speed to ~ 44% which equates to ~ 30.4 Mlbm/hr. Given the initial operating point this would result in the plant being in the Buffer Zone. A #2 runback can be reset via a pushbutton on the 904 runback. Additionally, when in the Buffer Zone, decay ratios are verified.

D. Incorrect: Locking both feed reg valves would be ineffective. Additionally, the plant is operating in the Exclusion Region

Technical Reference(s): Power to Flow Map (Attach if not previously provided)
PNPS 2.4.49
PNPS 2.4.165
ARP C904R B-4

Proposed References to be provided to applicants during examination: Power To Flow Map

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295010	2.2.12
	Importance Rating		4.1

Equipment Control: Knowledge of surveillance procedures. (High Drywell Pressure)

Proposed Question: SRO Question # 9

Surveillance procedure 8.M.2-2.10.2-3, RHR System High Drywell Pressure Auto-Initiation Trip System "A", is in progress.

You are the Control Room Supervisor when degrading grid conditions warrant terminating the surveillance and restoring RHR to its normal configuration.

IAW 8.M.2-2.10.2-3, which one of the following set of actions would satisfy the MINIMUM requirements for "backing out" of this procedure?

A. Direct that a verbal or written back out plan be developed that is independently reviewed by the Field Support Supervisor. Obtain shift manager concurrence before proceeding.

B. Conduct a pre-evolution brief with the personnel involved to determine the necessary steps for backing out of the surveillance. Obtain shift manager concurrence before proceeding.

C. Conduct a pre-evolution brief with the personnel involved. Direct the technicians to perform the surveillance in REVERSE order to ensure the system is restored without causing unintended actuations.

D. Direct the technicians to perform the surveillance in REVERSE order to ensure the system is restored without causing unintended actuations. Direct an independent restoration walk down be performed.

Proposed Answer: A

Explanation (Optional):

A. Correct: Precautions in 8.M.2-2.10.2-3, RHR System High Drywell Pressure Auto-Initiation Trip System "A" state that should it become necessary to back out of this procedure, a verbal or written plan must be developed and be independently reviewed (e.g., by another Supervisor, Lead, Operations). Appropriate schematics, P&IDs, and logic diagrams should be reviewed. Approval of the SM should be obtained before

proceeding. This is based on a PNPS event where a surveillance procedure was being performed and the crew attempted to "back-out" of the procedure due to unstable plant conditions. An inadvertent recirculation pump trip occurred when steps to restore the logic were not completed in the proper sequence. Subsequently this precaution was added to the procedure.

B. Incorrect: A verbal or written plan must be developed that is independently reviewed. A pre-evolution brief would not satisfy this requirement.

C. Incorrect: Performing the surveillance in reverse order would not necessarily reset the logics caused by the tripping of relays, removal of leads, etc., during the performance of the testing. This is the reason why the procedure specifies that the plan must consider and properly sequence the following:
- Effects of M&TE removal
- Effects of jumper removal
- Effects of removing boots or contact blocks
- Effects of re-landing leads
- Requirements to reset alarms, relays, etc.

D. Incorrect: Performing the surveillance in reverse order would not necessarily reset the logics caused by the tripping of relays, removal of leads, etc., during the performance of the testing. The second sentence is plausible in that a pre-requisite valve and breaker lineup is performed at the beginning of the test.

Technical Reference(s): 8.M.2-2.10.2-3, Precautions (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 3

Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295032	EA2.02
	Importance Rating		3.5

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : Equipment operability

Proposed Question: SRO Question # 10

The plant is at rated conditions. The "A" Suppression Pool Cooling and Containment Spray subsystems have been declared inoperable to allow maintenance on Torus Cooling Valve MO-1001-36A's breaker. The plant is on day 2 of the associated 7 day LCO per Tech Specs 3.5.B.1 and B.2.

Surveillance procedure 8.M.2-2.5.3, HPCI STEAM LINE HIGH TEMPERATURE, has just been completed.

- I&C reports that one HPCI Turbine Compartment Exhaust Duct temperature switch is tripping at 170 degrees F and CANNOT be adjusted any lower.
- The three other temperature switches tripped at 160 degrees.

In accordance with PNPS Technical Specifications, which one of the following statements is correct regarding continued plant operation?

A. The reactor MUST be placed in cold shutdown within 24 hours if the temperature switch is NOT repaired beforehand.

B. HPCI can be isolated and declared inoperable and continued reactor operation is now limited by the breaker maintenance.

C. A tracking LCO can be entered for the temperature switch failure. Continued reactor operation remains limited by the breaker maintenance.

D. HPCI MUST be isolated and declared inoperable AND the reactor MUST be placed in cold shutdown within 24 hours if the breaker maintenance is NOT completed beforehand.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Note 1 to table 3.2.B, states that whenever any CSCS subsystem is required by Section 3.5 to be operable, there shall be two operable trip systems. If the first

column cannot be met for one of the trip systems, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip system is made or found to be inoperable. The Minimum # of Operable Instrument Channels Per trip system for the high temperature isolation is 2 (there are only two per system). With one switch inoperable (maximum trip setting is 168 degrees) this requirement CANNOT be met. However HPCI can be declared inoperable which would result in a 14 day LCO by 3.5.C. The LCO for the breaker maintenance then becomes limiting.

B. Correct: HPCI can be isolated per note 3 of Table 3.2.B and declared inoperable which would result in a 14 day LCO by 3.5.C. The LCO for the breaker maintenance then becomes limiting.

C. Incorrect: All four temperature switches are required and an active LCO must be entered.

D. Incorrect: The breaker maintenance inops a suppression pool cooling and containment spray loop. This does not impact the HPCI LCO (as does LPCI).

Technical Reference(s): Tech Spec TABLE 3.2.B INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS (Attach if not previously provided)

Tech Spec 3.5.C

Proposed References to be provided to applicants during examination:

1. Last page ONLY of Tech Spec TABLE 3.2.B and associated note page.
2. Tech Spec 3.5.C, NO bases
3. Tech Spec 3.5.B.1, NO bases
4. Tech Spec 3.5.B.2, NO bases

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Facility operating limitations in the technical specifications and their bases.

Comments:

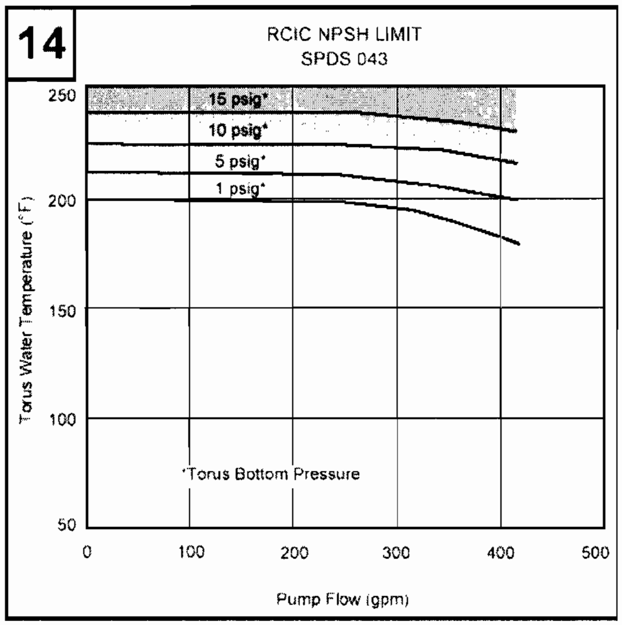
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	217000 A2.19	
	Importance Rating		3.6

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: High suppression pool temperature

Proposed Question: SRO Question # 11

Following a seismic event and associated LOCA, the following conditions exist:

- The CSTs have ruptured and are not available
- RCIC is injecting at rated flow
- RPV Level is -110 inches
- RPV pressure is 500 psig
- Drywell Sprays are in service using RHR pump "A"
- No other sources of injection are available
- Drywell Pressure is 5 psig and lowering
- Torus Bottom Pressure is 10 psig and lowering
- Torus Water Temperature is 205 degrees and rising



Given that there is inadequate cooling to maintain RCIC lube oil temperatures, which one of the following is correct regarding:

(1) The impact of plant conditions on RCIC operation

AND

(2) Any REQUIRED actions?

A. (1) NPSH limits are also currently being exceeded.
(2) Immediately secure RCIC. Enter EOP-17 and allow RHR pump "A" to inject.

B. (1) NPSH limits are also currently being exceeded.
(2) Monitor RCIC status. There is NO requirement to immediately secure RCIC.

C. (1) NPSH limits are NOT currently being exceeded.
(2) Immediately secure RCIC. Enter EOP-17 and allow RHR pump "A" to inject

D. (1) NPSH limits are NOT currently being exceeded.
(2) Monitor RCIC status. There is NO requirement to immediately secure RCIC.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: NPSH limits are not being exceeded. Plausible if the candidate uses drywell pressure vice torus bottom pressure when plotting conditions on Figure 14. Also Exceeding a caution is NOT meant to imply any specific action but merely to warn the operator that a condition may occur.

B. Incorrect: NPSH limits are not being exceeded.

C. Incorrect: Per caution 3, with a torus temperature above 140 degrees, RCIC equipment may overheat and damage may occur. Per the PNPS PSTG, Appendix A, this is the maximum allowable cooling water temperature for RCIC lube oil temperature. Additionally, exceeding a caution is NOT meant to imply any specific action but merely to warn the operator that a condition may occur. The action described is plausible if the candidate believes that exceeding a caution implies that action must be immediately taken to prevent equipment damage.

D. Correct: Per caution 3, with a torus temperature above 140 degrees, RCIC equipment may overheat and damage may occur. Per the PNPS PSTG, Appendix A, this is the maximum allowable cooling water temperature for RCIC lube oil temperature. Since the CSTs are not available, suction is aligned to the torus. With suction aligned to the torus, the cooling medium to the lube oil cooler is torus water. The NPSH limits are also NOT being exceeded. With a Torus Bottom pressure of 10 psig and a RCIC flow of 400 gpm, the Torus Water Temperature limit is ~ 220 degrees, well above the current temperature of 205 degrees. Exceeding a caution is NOT meant to imply any specific action but merely to warn the operator that a condition may occur.

Technical Reference(s):	Cautions 2 and 3 of EOP-01 EOP-11, Figure 14 PNPS PSTG Appendix A, page A-4-12 EOP-01 LP, page 28 EOP Development and Use LP, page 24 and 25	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:		(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:		Last NRC Exam:
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43 5	

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	218000	A2.02
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Large break LOCA

Proposed Question: SRO Question # 12

A large break LOCA outside the primary containment has occurred. The break cannot be isolated. The following conditions CURRENTLY exist:

- RCIC is injecting at rated flow
- HPCI and ALL low pressure ECCS are NOT available
- Operators have been dispatched to align Alternate Injection Systems IAW EOP-01, Table B
- RPV level is -7 inches and lowering at 10 inches per minute
- Drywell pressure is 1.2 psig and stable

Assuming that the above conditions do not change, which one of the following:

(1) describes the response of the Automatic Depressurization System (ADS) if NO manual action is taken?

AND

(2) what orders are required to be directed?

A. (1) all ADS valves will open ~ 6 minutes from now.
(2) Verify ADS initiates as designed. WHEN RCIC isolates and RPV Level cannot be restored and maintained above -150 inches exit all EOPs and enter SAGs.

B. (1) all ADS valves will open ~ 6 minutes from now.
(2) BEFORE ADS initiates, direct that ADS be inhibited. BEFORE RPV level lowers to -125 inches direct all SRVs be opened IAW EOP-17 Emergency RPV Depressurization.

C. (1) ADS 11 minute timer will initiate but ADS valves will not automatically open
(2) within the next 4 minutes direct that ADS be inhibited. BEFORE RPV level lowers to -125 inches direct all SRVs be opened IAW EOP-17 Emergency RPV Depressurization.

D. (1) ADS 11 minute timer will initiate but ADS valves will not automatically open.

(2) within the next 4 minutes direct that ADS be inhibited. WHEN RPV level cannot be restored and maintained above -150 inches direct all SRVs be opened IAW EOP-17 Emergency RPV Depressurization.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: ADS valves will not open because a low pressure ECCS pump running signal does not exist. The six minutes is plausible in that in 4 minutes, RPV level will lower to -46 inches at which time an ADS timer will initiate. If all conditions were met for an ADS blow down, the ADS valves would open 2 minutes later. Additionally, EOP-01 directs that ADS always be inhibited because the decision to depressurize is delegated to the SRO after evaluating plant conditions.

B. Incorrect: ADS valves will not open because a low pressure ECCS pump running signal does not exist. Additionally, an RPV depressurization is not performed until RPV level cannot be restored and maintained above -150 inches.

C. Incorrect: An RPV depressurization is not performed until RPV level cannot be restored and maintained above -150 inches.

D. Correct: Per PNPS 2.2.23, ADS, The signal for the relief valves to open and remain open is based upon simultaneous signals from all of the following:

- Drywell high pressure or expiration of the high Drywell pressure bypass timer.
- Reactor Vessel low-low water level.
- Adequate discharge pressure from at least one of the LPCI or Core Spray pumps.
- Expiration of the 2-minute ADS timer.

The initial conditions stated that other than RCIC, no other injection sources were available. Therefore the 3rd condition above will not be satisfied and ADS will not initiate.

EOP-01, step L-6, directs that before RPV level drops to -46 inches, ADS is to be inhibited. Based on the rate of level drop, this will occur in ~ 4 minutes. Additionally, steps L-15 and 16 direct that after level drops below -125 inches, and an injection system is running, that Emergency RPV Depressurization is required when RPV level cannot be restored and maintained above -150 inches.

Technical Reference(s): PNPS 2.2.23, ADS (Attach if not previously provided)

EOP-01 level leg

Proposed References to be provided to applicants during examination: None

Learning Objective:			(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	5	

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	300000	2.1.20
	Importance Rating		4.6

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: SRO Question # 13

An earthquake followed by a station blackout has resulted in the following conditions:

- A plant cooldown is in progress using SRVs
- The SRVs have been cycled over 20 times to control RPV pressure
- Annunciator C904LC-F3, AIR/N2 TO DRYWELL TROUBLE, alarms
- A Control Room operator reports that PI-4348, NITROGEN SUPPLY DRYWELL EQUIP SUPPLY PRESSURE, on Panel C7 reads 70 psig and is continuing to lower

Which one of the following is required?

A. Enter 2.2.105, Backup Nitrogen Supply System, isolate the normal supply, place the backup nitrogen system in service, and then depressurize with sustained relief valve opening.

B. Enter 5.3.8, Loss of Instrument Air, Direct that instrument air be aligned to supply drywell pneumatics and isolate the nitrogen supply, and then continue cycling SRVs as necessary to maintain the cooldown rate.

C. Enter 2.2.23, Automatic Depressurization System, and direct an operator to connect nitrogen cylinders on the Reactor Building 23 foot to recharge the SRV accumulators, and then depressurize with sustained relief valve opening.

D. Enter 2.2.70, Primary Containment Atmospheric Control System, and direct an operator to connect nitrogen cylinders on the Reactor Building 23 foot to recharge the SRV accumulators, and then continue cycling SRVs as necessary to maintain the cooldown rate.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - There no direction to isolate the normal supply to place the backup nitrogen system in service since they use the same piping. Additionally the N2 makeup lines are

not seismically qualified so they may not be available and there is no direction in 2.2.105 to then depressurize with sustained relief valves opening.

B. Incorrect – EOP-01 directs that if there is a loss of continuous N2 supply the portable N2 cylinders be aligned IAW PNPS 2.2.70. Additionally, instrument air uses the same pneumatic line as the N2 which is not seismically qualified.

C. Incorrect - there is no direction in 2.2.23, Automatic Depressurization System, to connect nitrogen cylinders to relief valve backup system. Additionally once the seismically qualified backup is placed in service, the continuous pneumatic supply has been restored and sustained SRV opening is not required.

D. Correct – A loss of drywell pneumatics is indicated by the annunciator and lowering pneumatic pressure. IAW EOP-01, the crew will need to enter 2.2.70, Sect 7.12 Backup N2 Supply for Extended SRV Operation – Emergency Operations, this will re-establish a continuous N2 supply to the SRVs and permit continuing the cycling of the SRVs for pressure control.

Technical Reference(s): EOP-1 (Table D) and 2.2.70, pg 18 and 57, 58
ARP–C904LC-F3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	215005	2.2.25
	Importance Rating		4.2

APRMs, Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question: SRO Question # 14

With the plant operating at 100% power the following events occur:

- PNPS 9.1, APRM Calibration, has just been completed
- You are informed that the following APRM gain adjustment factors (AGAF) exist:

APRM "A"	APRM "B"	APRM "C"	APRM "D"	APRM "E"	APRM "F"
.985	.999	1.02	1.01	1.00	.961

Which one of the following is (1) the required action and (2) the bases for this action?

A. (1) Declare two APRMS inoperable, no Technical Specification actions are required.
(2) Their power indications are non-conservative and could result in exceeding the fuel safety limit during an operational transient.

B. (1) Declare three APRMs inoperable, restore one RPS "A" APRM to operability within 12 hours or place RPS "A" in the tripped condition
(2) Their power indications are non-conservative and could result in exceeding the fuel safety limit during an operational transient.

C. (1) Declare two APRMS inoperable, no Technical Specification actions are required.
(2) Their power indications are non-conservative and could result in exceeding the limits established in the Core Operating Limits Report during normal operation.

D. (1) Declare three APRMs inoperable, restore one RPS "A" APRM to operability within 12 hours or place RPS "A" in the tripped condition
(2) Their power indications are non-conservative and could result in exceeding the limits established in the Core Operating Limits Report during normal operation.

Proposed Answer: A

Explanation (Optional):

A. Correct. IAW PNPS 9.1, (Steps 8 & 9) 8. OBTAIN a current Core Power and Flow Log OR REVIEW 3D Monicore graphics screen 233, "NSSS Heat Balance", AND VERIFY the AGAF of the channel being calibrated is greater than or equal to 0.87 and less than or equal to 1.00. IF NOT, THEN REPEAT Steps [7] and [8] until the AGAF meets this criteria.
9. IF unable to calibrate the APRM channel to meet the Acceptance Criteria Step 8.0[1] of this procedure, INFORM the SM so an impact on APRM operability can be evaluated.

With an AGAF of 1.02 and 1.01 APRMs C and D are reading lower than the actual power level, the APRM flow biased and high flux scram setpoint s are non conservative and the APRM must be declared inoperable. IAW with T.S. 3.1.1, with two operable tripped trip systems for each trip function the requirements for satisfying minimum operability are met and no T.S. actions are required.

With the APRM inoperable one of the abnormal operational transients analyzed may violate the fuel safety limit.

B. Incorrect: Only APRMs C and D are reading lower than the actual power level, APRM E is accurate.

C. Incorrect: The APRM scram and rod block setpoints are based on preventing the fuel from exceeding the fuel safety limits during operational transients.

D. Incorrect: Only APRMs C and D are reading lower than the actual power level, APRM E is accurate. The APRM scram and rod block setpoints are based on preventing the fuel from exceeding the fuel safety limits during operational transients.

Technical Reference(s): 9.1, Sect. 8
2.1.15, Daily Log Test #7
T.S. 3.1.1 and Bases
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43 2	

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	239002	2.1.23
	Importance Rating		4.4

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation. (SRVs)

Proposed Question: SRO Question # 15

Following a reactor scram, the following conditions exist:

- All control rods are fully inserted
- MSIVs are closed
- Reactor pressure is 600 psig and rising slowly
- RPV water level cannot be determined
- RCIC, HPCI and Reactor Feed Pumps are NOT available
- CRD and SLC are injecting
- All radiation levels are normal
- Torus water level is 48 inches and stable

Which one of the following actions is required?

A. Exit EOP-1, RPV Control, enter EOP-18, Steam Cooling and open the four SRVs.

B. Continue in EOP-1, RPV Control, and enter EOP-16, RPV Flooding and open the four SRVs.

C. Continue in EOP-1, RPV Control, and transition to EOP-18, Steam Cooling and Stabilize RPV pressure with SRVs

D. Exit EOP-1, RPV Control, enter EOP-16, RPV Flooding, and depressurize using alternate RPV depressurization systems.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. There is no direction to enter steam cooling because there is no water level indication. Exit EOP-1.

B. Incorrect. When RPV water level cannot be determined, EOP-01 is exited. RPV parameters are then controlled via EOP-16. Also, with torus water level less than 50

inches the SRVs should not be opened, alternate RPV depressurization is required. Exit EOP-1.

C. Incorrect. When RPV water level cannot be determined, EOP-01 is exited. RPV parameters are then controlled via EOP-16. Plausible in that candidate may believe that steam cooling is required due to the loss of level indication. Also plausible in that Steam Cooling directs that pressure be stabilized.

D. Correct. With a loss of RPV water level indication the SRO is required to enter EOP-16. Step F-8 of EOP-16 directs that if No SRVs can be opened, AND Feed Pumps are not available, AND RPV pressure is > 50 psig above Torus pressure than the RPV is to be depressurized using alternate depressurization systems. Given that torus water level is below 50 inches use of the SRVs is prohibited per step F-2 of EOP-16. If RPV water level cannot be determined, the actions specified in the subsequent steps cannot be performed since RPV water level and water level trend information is required for determining which actions take. Transferring control of RPV water level from the RPV Control procedure to EOP-16 "RPV Flooding," is necessary to assure continued adequate core cooling under conditions where RPV water level cannot be determined. Additionally transfer to EOP-16 is also directed from the Pressure Control flowpath to ensure the RPV is depressurized for RPV Flooding.

Technical Reference(s): EOP-1 & EOP-16
EOP-1, RPV Control Instructor Guide (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	215001	A2.08
	Importance Rating		2.9

Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to retract to shield: (Not-BWR1)

Proposed Question: SRO Question # 16

The plant was operating at 85% power with Traversing In-Core Probe (TIP) system in operation using 9.5.1, Operation of TIP Machines for Process Computer Updating, when the following occur:

- EOP-1, RPV Control has been entered due to a confirmed LOCA condition
- HPCI has automatically initiated and is maintaining RPV water level
- TIP-1 was beginning its run at the time of the transient

Based on these plant conditions what action is required if the detector associated with TIP-1 is stuck inside the core?

A. Enter 2.2.69, Traversing In-Core Probe System, attempt TIP retraction in Manual mode. If the TIP CANNOT be manually retracted the CRS must direct firing the shear valve.

B. Enter 3.M.2-5.6.13, Manual Operation of TIP System, and manually attempt to withdraw TIP 1. If the TIP CANNOT be manually retracted the CRS must direct firing the shear valve.

C. Enter 3.M.2-5.6.13, Manual Operation of TIP System, and manually attempt to withdraw TIP 1. If there is a coolant leak in the guide tube the CRS can consider firing the shear valve based on plant conditions.

D. Enter 2.2.69, Traversing In-Core Probe System, attempt TIP retraction in Manual mode of control TIP 1. If there is a coolant leak in the guide tube the CRS can consider firing the shear valve based on plant conditions.

Proposed Answer: A

Explanation (Optional):

A. Correct. Upon the receipt of a containment isolation signal (low Reactor level or high Drywell pressure), any TIP that is inserted past its in-shield limit will automatically shift to the REVERSE mode of operation and commence to withdraw to the In-Shield position and the ball valve closes. A shear valve is provided for emergency use only. If a coolant leak develops in a guide tube and for some reason the detector cable cannot, or should not, be withdrawn, or if the ball valve fails to close, then the shear valve can be detonated.
If the isolation received is verified to be an actual emergency isolation, then the TIP shear valves shall be fired in accordance with PNPS 2.2.69, Step 7.3[5].

B. Incorrect. 3.M.2-5.6.13, Manual Operation of TIP System is used to locally manually operate the TIP system. It would NOT be used during a LOCA.

C. Incorrect. 3.M.2-5.6.13, Manual Operation of TIP System is used to locally manually operate the TIP system. It would NOT be used during a LOCA. If the isolation received is verified to be an actual emergency isolation, then the TIP shear valves shall be fired in accordance with PNPS 2.2.69, Step 7.3[5].

D. Incorrect. If the isolation received is verified to be an actual emergency isolation, then the TIP shear valves shall be fired in accordance with PNPS 2.2.69, Step 7.3[5].

Technical Reference(s): 2.2.69, Sect 5.0, 5.2, 7.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	214000	2.1.7
	Importance Rating		4.7

Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (RPIS)

Proposed Question: SRO Question # 17

EOP-02, Failure to Scram was entered following an automatic scram on high drywell pressure. Current plant conditions are as follows:

- Four (4) rods are at position 02
- Eight (8) rods are at position 04
- Due to an RPIS failure, the position of one rod can NOT be determined
- CALL RODS display is MAGENTA
- Drywell Pressure is 3.5 psig
- Reactor pressure is 940 psig
- Reactor Power is on Range 2 of the fully inserted IRMs and lowering.
- NO Boron has been injected

Which one of the following is now required?

A. Exit the Power Leg of EOP-02 and enter Scram Procedure, 2.1.6
Commence a reactor plant cooldown IAW the pressure leg of EOP-02

B. Exit EOP-02 completely and enter EOP-01, RPV Control
Commence a plant cooldown IAW the pressure leg of EOP-01

C. Exit the Power Leg of EOP-02 and enter Scram Procedure, 2.1.6
Do NOT commence a plant cooldown. Stabilize pressure between 900 and 1050 psig IAW EOP-02 Pressure Leg.

D. Remain in the Power Leg of EOP-02 and insert control rods IAW Alternate Rod Insertion procedure, 5.3.23.
Do NOT commence a plant cooldown. Stabilize pressure between 900 and 1050 psig IAW EOP-02 Pressure Leg.

Proposed Answer: A

Explanation (Optional):

A. Correct: IAW the second override in the power leg of EOP-02, the power leg of EOP-02 is exited and the scram procedure entered if the reactor is shutdown and no boron has been injected. Per PNPS 5.3.35, Shutdown is defined as: As applied to the Reactor, subcritical with Reactor power below the heating range (on scale on IRM range 7 or below). Given that the IRMs are on range 2, the reactor is shutdown and the power leg is to be exited. This same term is used in the pressure leg of EOP-02, step P-6 which governs whether a plant cooldown should be commenced. Therefore a plant cooldown should also be performed since no boron has been injected. Although one rod's position cannot be determined and must be assumed to be beyond the Max Subcritical Bank Rod Position, the reactor is currently shutdown but there is no guarantee that it will remain shutdown under all conditions.

B. Incorrect: EOP-02 is completely exited via overrides in each leg if the existing control rod pattern can always assure that the reactor will remain shutdown. As defined in PNPS 5.3.35, this condition is met if ANY of the following occur:
(a) All control rods are inserted to position 04 or beyond.
(b) All control rods inserted to position 00 except one control rod may be withdrawn(at a position up to 48).
(c) Cold Shutdown boron weight has been injected into the RPV.
(d) When determined by Reactor Engineering analysis.

With the position of one rod cannot be determined, it must be assumed to be above position 04 and therefore there is no guarantee that it will remain shutdown under all conditions..

C. Incorrect: A plant cooldown is directed since the reactor is currently shutdown. Plausible if the candidate believes that a cooldown cannot be performed unless there is assurance that the reactor will not restart. However override P-7 governs this occurrence.

D. Incorrect: EOP-02 power leg should be exited as described above. Plausible in that EOP-02 does direct inserting rods via 5.3.23. However this same direction is also included in the scram procedure. Additionally a cooldown is directed as described above.

Technical Reference(s): EOP-02 (Attach if not previously provided)

PNPS 5.3.35

Proposed References to be provided to applicants during examination: EOP-02 power and pressure legs.

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

	New	X

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	201003	A2.09
	Importance Rating		3.4

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: Low reactor pressure

Proposed Question: SRO Question # 18

A reactor plant shutdown is in progress IAW PNPS 2.1.5 with the following plant conditions:

- Reactor power is 18%
- The accumulator for control rod 30-15 failed one week ago and is inoperable
- Control rod 30-15 has been declared "SLOW" in accordance with Technical Specifications
- Control rod 30-15 is at position 36
- There are NO other control rod or accumulator deficiencies
- Reactor pressure has just now lowered to less than 950 psig during the shutdown
- The rod is NOT in the current RWM step

What action is NOW required regarding control rod 30-15?

A. Place the reactor mode switch in SHUTDOWN immediately.

B. Declare the rod inoperable within one hour and continue with the plant shutdown. There are no additional requirements.

C. Declare the rod inoperable within one hour, fully insert the rod within 3 hours and then electrically disarm the rod within 4 hours.

D. Fully insert the control rod within 3 hours and restore compliance with BPWS within the next 8 hours by inserting all remaining rods.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: This action would be correct if CRD charging pressure was also < 940 psig as addressed in TS 3.3.D. C and D.

B. Incorrect: In addition to declaring the rod inoperable, the rod must be fully inserted and disarmed as prescribed by TS 3.3.B.1.C.

C. Correct: PNPS 2.1.5 warns that control rods with inop accumulators also become inop when pressure drops below 950 psig. TS 3.3.D.C applies in this condition (one or more accumulators inop with pressure < 950 psig). TS 3.3.D.C.1. does not apply since charging water pressure is NOT < 940 psig (normal charging water pressure is much higher). However TS 3.3.D.C.2 requires that the associated control rod be declared inop within one hour. As a result, TS 3.3.B.1.C then applies (one or more control rods inop for reasons other than being stuck). This then requires that the control rod be fully inserted within 3 hours and be disarmed within 4 hours.

D. Incorrect: The TS addressing compliance with BPWS are not applicable. Although the rod will not be in compliance, TS 3.3.H.A only addresses OPERABLE rods not in compliance.

Technical Reference(s): TS 3.3.D. C and D. (Attach if not previously provided)

TS 3.3.B.1.C

Proposed References to be provided to applicants during examination:
TS 3.3.B.1
TS 3.3.D
TS 3.3.H
NO BASES

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.19
	Importance Rating		3.8

Ability to use plant computers to evaluate system or component status.

Proposed Question: SRO Question # 19

During a reactor startup the plant is at 52% power and 45 MLB/HR. From these initial conditions the "A" Recirc Pump trips. The following indications now exist:

- Reactor power stabilizes at 45%
- Solomon indicates that the Core Decay Ratio is 0.75 and the Hot Channel decay Ratio is 0.6
- EPIC Computer Group Point display # 22 is as indicated in the table below.

Group Number: 22	Group Name: SINGLE RECIRC LOOP OPS (2.4.17)			
Point ID	Description	Status	Current Value	Engineering Units
REC052	REACTOR CORE FLOW	NML	33.0	MLB/HR
SLCORFLO	CORE FLOW SINGLE RECIRC LOOP OPS	NML	24.0	MLB/HR
REC052CP	REACTOR CORE FLOW	NML	24.0	MLB/HR
REC054	JET PUMP LOOP A FLOW	NML	4.5	MLB/HR
REC056	JET PUMP LOOP B FLOW	NML	28.5	MLB/HR
REC507	A RECIRC M-G SET FIELD BREAKER	NML	OPEN	
REC508	B RECIRC M-G SET FIELD BREAKER	NML	CLOSED	

Which one of the following actions is correct IAW PNPS 2.4.17, Recirc Pump Trip? A power to flow map is provided for your use.

A. Reverse flow exists. The single loop toggle switches for each APRM Flow Control Trip Reference card must be placed into the "SLO" position within the next 24 hrs IAW TS 3.6.F, Single Loop Operation.

B. Reverse flow exists but Core Decay and Hot Channel Decay Ratios calculations are NOT accurate. Place PBDS in service by taking the NORMAL/BYPASS toggle switch on each PBDS card to the "NORMAL" position IAW PNPS 2.2.160, PBDS.

C. Forward flow exists and Core Decay and Hot Channel Decay Ratios calculations ARE accurate. Increase recirc flow OR insert control rods to lower the decay ratios to within limits IAW PNPS 2.4.165, Reactor Core Instability.

D. Forward flow exists and the core flow inputs into the thermal limit calculations are NOT accurate. Direct Reactor Engineering to enter a substitute value for core flow into the 3D Monicore Computer within the next 24 hrs IAW TS 3.6.F, Single Loop Operation.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Using the process described in step 2 of the subsequent actions of PNPS 2.4.17 and EPIC computer points, REC052 and SLCOFLO, forward flow exists. Plausible in that if reverse flow existed, this would be a correct action.

B. Incorrect: Using the process described in step 2 of the subsequent actions of PNPS 2.4.17 and EPIC computer points, REC052 and SLCOFLO, forward flow exists. Plausible in that if reverse flow existed, and the computer was unable to calculate decay ratios accurately or was otherwise unavailable the correct action would be to place PBDS in service.

C. Incorrect: Because the process computer assumes reverse flow, the calculations for decay ratios are not accurate. Plausible in that if the calculations were accurate this would be the required action as the decay ratios are too high.

D. Correct: Using the process described in step 2 of the subsequent actions of PNPS 2.4.17 and EPIC computer points, REC052 and SLCOFLO, forward flow exists. As described in step 6 of the subsequent actions of PNPS 2.4.17 the core flow input to the EPIC computer is NOT accurate and the thermal limit and decay ratio calculations are not accurate. Per Tech Spec 3.6.F, 24 hours are available to establish the thermal limits associated with single loop. Reactor engineering must be directed to insert a substitute value for core flow as described in discussion item 4 and subsequent action 6. (b) of PNPS 2.4.17.

Technical Reference(s): PNPS 2.4.17, Subsequent actions and discussion section. (Attach if not previously provided)

TS 3.6.F

Proposed References to be provided to applicants during examination: BOTH Power to Flow maps

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:	Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.40
	Importance Rating		4.7

Equipment Control: Ability to apply technical specifications for a system.

Proposed Question: SRO Question # 20

Reactor power is 100% with all systems operable and in a normal full power configuration, when 480 VAC motor control center B-15 is lost. Operators stabilize plant conditions by performing the following actions in accordance with station procedures:

- Cross-tie RBCCW
- Close SSW Cross-Over Valve MO-3808

In accordance with the technical specifications for RBCCW and SSW, which one of the following is required and the bases for that requirement?

A. Be in cold shutdown within 24 hours because both SSW subsystems are inoperable.

B. Be in cold shutdown within 24 hours because both RBCCW subsystems are inoperable.

C. Be in cold shutdown within 24 hours because the Ultimate Heat Sink is inoperable.

D. Restore BOTH the “A” RBCCW AND “A” SSW subsystems to operable status within 72 hours due to the loss of associated cooling water pumps.

Proposed Answer: B

Explanation (Optional):

A. Incorrect response: The “A” SSW subsystem is inoperable because the “A” SSW pumps have lost power. However the “B” subsystem is operable because the SSW cross-over valve was closed.

B. Correct response: Both RBCCW subsystems are inoperable because the cross-connect valves are open (Tech Spec Bases page B3/4.5-12)

C. Incorrect response: The ultimate heat sink remains operable (Tech Spec Bases page B3/4.5-15a)

D. Incorrect response: Although both “A” RBCCW and SSW subsystems are inoperable,

the more limiting LCO for both RBCCW subsystems being inoperable is applicable.

Technical Reference(s): Tech Spec 3.5.B.3 and associated bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.11
	Importance Rating		4.3

Radiation Control: Ability to control radiation releases.

Proposed Question: SRO Question # 21

In accordance with PNPS 2.5.2.17, PROCESSING LIQUID RADIOACTIVE DISCHARGES, which of the following are required for a liquid discharge from the Pilgrim Station?

1. A valid discharge permit signed by the Shift Manager must be in the possession of the Operator prior to discharging any liquid from the Radwaste System.

2. The Shift Manager must personally supervise the lineup of valves for the tank to be discharged and check off such verification on permit.

3. Discharge flow rate recorder (FR-7214A or FR-7214B) on Panel C20 MUST be operable for the duration of the discharge.

A. 1 only

B. 1 and 2 only

C. 1 and 3 only

D. 1, 2 and 3

Proposed Answer: B

Explanation (Optional):

A. Incorrect - Requirement # 2 is also required.

B. Correct – The Shift Manager must personally supervise the lineup of valves for the tank to be discharged and check off such verification on permit. In addition, he signs off OPER-28A, OPER-28B, OPER-28C, or OPER-28D (Attachment 1, 2, 3, or 4). Additionally the SM checks off the items in the SM portion of the permit and signs to authorize the discharge.

C. Incorrect - Requirement # 3 is not required. If the discharge flow rate recorder (FR-7214A or FR-7214B) on Panel C20 fails, effluent releases must be secured however the

SM may permit the release to be re-started provided tank levels are recorded every five (5) minutes. Additionally, requirement # 2 is also required.

D. Incorrect - Requirement # 3 is not required as discussed above.

Technical Reference(s): 2.5.2.17, Sect. 7.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.50
	Importance Rating		4.0

Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: SRO Question # 22

Control rod 18-23 is being moved from position 22 to 28 as part of a rod pattern exchange. During the withdrawal the following occurs:

- C905L- B4, ROD WITHDRAWAL BLOCK, alarms stopping the control rod at position 26
- C905L- D3, RBM HI/INOP, alarms
- RBM A indicates 112%
- RBM B indicates 115%
- All LPRM inputs are determined to be operable
- Reactor Power is 75%
- A Limiting Control Rod Pattern exists

In accordance with C905L ARPs and Technical Specification Table 3.2.C-1, which one of the following is required?

A. Notify Reactor Engineering and I&C the rod pattern may require adjustment, then direct control rod pattern adjustments as required.

B. Direct the insertion of the withdrawn control rod to its original position and verify the alarms clear, no further actions are required.

C. Direct the operator to de-select control rod 18-23 and then re-select 18-23 and continue with the control rod withdrawals, withdrawing 18-23 to position 28.

D. Declare one RBM channel inoperable and restore the inoperable RBM channel to operable status within 24 hours or place one rod block monitor channel in the tripped condition

Proposed Answer: A

Explanation (Optional):

A. Correct. The trip setpoint for the RBMs at 75% power is 115% (COLR) or 114% by

2.2.68. RBM B has reached this setpoint and therefore the alarm is valid. IAW T.S and the ARP for C905L, D3, For Reactor Power < 90% and MCPR < 1.72 - notify Reactor Engineering and I&C the rod pattern may require adjustment.

B. Incorrect. There is no direction to insert the control rod and further action is required.

C. Incorrect. This would null the RBM and possibly allow the control rod to be withdrawn to position 28, this would be against the procedure that requires stopping rod movement and informing RE.

D. Incorrect. Both channels of the RBM are operating as designed.

Technical Reference(s): TS Table 3.2.C-1 and notes 2 and 5
ARP for C905L, D3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.27
	Importance Rating		3.9

Emergency Procedures / Plan: Knowledge of "fire in the plant" procedures.

Proposed Question: SRO Question # 23

The plant is operating at 100% power when the following events occur:

- A fire has been reported in the area of the Standby Liquid Systems Tanks and Pumps, Reactor Building El. 91'3.
- Currently no conditions threaten safe plant operation
- 5.5.1, General Fire Procedure has been entered

Which of the following additional procedures are entered AND what actions are required?

1. 5.5.2, Special Fire Procedure
2. 2.4.143.1, Shutdown With a Fire in Reactor Building East (Fire Area 1.9)
3. 2.4.143.2, Shutdown With a Fire in Reactor Building West (Fire Area 1.10)

Enter procedure(s)

A. 2 only AND lower reactor power by running back recirculation pumps to minimum speed.

B. 3 only AND lower reactor power by running back recirculation pumps to minimum speed.

C. 1 and 2 AND lower reactor power by reducing core flow to as close to, but less than, 43 Mlb/hr total core flow.

D. 1 and 3 AND lower reactor power by reducing core flow to as close to, but less than, 43 Mlb/hr total core flow.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - 5.5.2, Special Fire Procedure is entered by the fire brigade to fight the fire. Recirculation pumps would only be run back to minimum speed if Reactor power is below the 60% load line.

B. Incorrect - 5.5.2, Special Fire Procedure is entered by the fire brigade to fight the fire. Recirculation pumps would only be run back to minimum speed if Reactor power is below the 60% load line.

C. Correct – Because the area of the Standby Liquid Systems Tanks and Pumps is in the east side of the reactor building the crew must enter 2.4.143.1. This procedure requires reducing reactor power by reducing core flow in accordance with PNPS 2.1.14, "Station Power Changes", Section 7.11. Under these conditions Section 7.11 directs lowering reactor power by reducing core flow to as close to, but less than, 43 Mlb/hr total core flow.

D. Incorrect - Because the area of the Standby Liquid Systems Tanks and Pumps is in the east side of the reactor building the crew must enter 2.4.143.1.

Technical Reference(s): 2.4.143.1
5.5.2
2.1.14
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			3
	K/A #	G3	2.3.14	
	Importance Rating			3.8

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: SRO Question # 24

A General Emergency has been declared following a major loss of coolant accident. The following plant conditions exist:

- Injection from all low pressure ECCS is required to maintain adequate core cooling
- The primary containment is about to be vented to maintain torus bottom pressure below the PCPL.
- The duration of the venting is unknown.
- Drywell CHRMs are reading 3000 R/Hr
- Torus CHRMs are reading 650 R/hr
- Wind direction is from 90°

Based on the above conditions and utilizing Attachment 1 of EP-IP 400, PROTECTIVE ACTION RECOMMENDATIONS, the required Protective Action Recommendation is:

Evacuate sub-area(s):

A. 12. Shelter all other sub-areas.

B. 1, 3, and 12. Shelter all other sub-areas.

C. 1, 2, 3, 4 and 12. Shelter all other sub-areas.

D. 1, 2, 3, 4, 6, 7, 8, 11 and 12. Shelter all other sub-areas.

Proposed Answer: D

.

Explanation (Optional):

A. Incorrect: This would have been the PAR if the question regarding the duration of the release /vent was answered incorrectly

B. Incorrect: This would be the PAR if any of the questions asked by step P-6, 7 or 8 were answered incorrectly and only the 2 mile ring and 5 miles downwind were to be

evacuated.

C. Incorrect: This would have been the PAR if the candidate does not understand that the PAR is based on the direction the wind is coming from and not the direction it is heading.

D. Correct: Based on the information given, substantial core damage has occurred (Torus CHRM reading), a significant release of reactor coolant has occurred (all ECCS pumps required) and containment failure is imminent (containment about to be vented). These three conditions taken together require that the 5 mile ring and 10 miles downwind be evacuated. Since the wind is from 90 degrees, this comprises 1, 2, 3, 4, 6, 7, 8, 11 and 12 subareas.

Technical Reference(s): EP-IP 400, PROTECTIVE ACTION RECOMMENDATIONS, Attachment 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EP-IP 400, Attachment 1, Sheets 2 and 3 ONLY

Learning Objective: (As available)

Question Source: Bank # LOR Bank # 181

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.32
	Importance Rating		4.0

Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: SRO Question # 25

Given the following conditions:

- The plant was shutdown to perform maintenance on the Group VI isolation circuitry.
- Shutdown cooling is in service with reactor water temperature at 140°F.
- MO-1201-2, RWCU Pump Suction Inboard Isolation Valve, and MO-1201-5, RWCU Pump Suction Outboard Isolation Valve, are open with their auto closure capabilities disabled.
- Both drywell personnel airlock doors are open with interlocks defeated.

Then ...

- A loss of shutdown cooling occurs and the RPV begins to heat up.
- One drywell personnel airlock door is then closed and sealed.
- The operator then closes the MO-1201-2 valve using the C904 control switch
- Primary Coolant temperature is now 214 degrees.

In accordance with PNPS Technical Specifications, Primary Containment Integrity....

A. was re-established when the one airlock door was closed and sealed.

B. was re-established when the one airlock door was closed and sealed AND MO-1201-2 was manually closed.

C. has NOT been established because NO valves in the RWCU suction line have been deactivated in the closed position.

D. has NOT been established because BOTH airlock doors have not been closed and sealed.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Primary Containment Integrity has NOT been re-established. MO-1201-5, RWCU Pump Suction Outboard Isolation is a motor operated PCIS isolation valve. Primary containment Integrity requires that all automatic primary containment isolation valves be operable except as specified in TS 3.7.A.2.b. TS 3.7.A.2.b.requires that at least one containment isolation valve in each line having an inoperable valve be deactivated in the isolated condition.

B. Incorrect: Primary Containment Integrity has NOT been re-established because PCIS must have one RWCU valve deactivated in the closed position.

C. Correct: Primary Containment Integrity has not yet been established. Primary Containment Integrity requires one personnel airlock to be closed and one valve in the line with inop PCIS valves to have one valve deactivated in the closed position.

D. Incorrect: Primary Containment Integrity requires only one personnel airlock to be closed; Tech Specs permits one airlock door to be open in this condition.

Technical Reference(s): Tech Spec 3.7.A.2.a
Tech Spec 3.7.A.2.b (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2002 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments: