Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 A3.01	
	Importance Rating	3.3	

Reactor Coolant Pump System: Ability to monitor automatic operation of the RCPs including: Seal injection flow

Proposed Question: Common 1

Given the following condition:

- Unit 2 is in MODE 1.

Which of the following indications would you expect to observe if the #1 seal on Reactor Coolant Pump 2-01 failed?

A. Seal #1 differential pressure rising.
Seal leak off flow rising.

B. Seal injection flow rising.
Seal leak off flow rising.

C. Seal #1 differential pressure rising.
Seal leak off flow lowering.

D. Seal injection flow lowering.
Seal leak off flow rising.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because seal leak off flow could be rising, however, seal #1 differential pressure would be lowering.
B. Correct. With seal injection and seal leak off flows both rising, the RCP #1 seal has failed.
C. Incorrect. Plausible because seal leak off flow could be lowering, however, seal #1 differential pressure would be lowering.
D. Incorrect. Plausible because seal leak off flow can be rising or lowering, however, seal injection flow would be rising.

Technical Reference(s) ABN-101, Section 4.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the function and operation of the following Reactor Coolant System components, flowpaths and features:

- Reactor Coolant Pump seal package (OP51.SYS.RC1.OB03)

ANALYZE the relationship between a Reactor Coolant Pump malfunction and the following:

- RCP Seals (LO21.ABN.101.OB02)

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 3, 10
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K1.05	
	Importance Rating	2.7	

Chemical and Volume Control System: Knowledge of the physical connections and/or cause-effect relationships between the CVCS and the following systems: CRDS operation in automatic mode control

Proposed Question: Common 2

Given the following conditions:

- Unit 1 is at 100% power.
- Rod Control is in AUTO.
- Latest Reactor Coolant System boron sample was 1203 ppm.
- An unsaturated mixed bed demineralizer was just placed in service without aligning Letdown to the Recycle Holdup Tank (HUT).

Which of the following is the expected plant response for the conditions listed?

Control Rods will start to...

A. withdraw as boron is absorbed by the demineralizer.

B. insert as boron is released from the demineralizer.

C. withdraw as boron is released from the demineralizer.

D. insert as boron is absorbed by the demineralizer.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because boron is being absorbed by the demineralizer, however, this will cause Control Rods to insert.
B. Incorrect. Plausible because Control Rods will insert, however, the reason is due to boron being absorbed by the demineralizer with beginning-of-cycle conditions.
C. Incorrect. Plausible if thought that a fresh mixed bed demineralizer was already borated.
D. Correct. With beginning-of-cycle conditions, a fresh mixed to bed demineralizer will absorb boron from the system and cause Control Rods to insert.

Technical Reference(s) SOP-103A, Section 3.0 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause-effect relationships between the CVCS and the following systems, components or events:

- Expected reactivity changes after valving in a new mixed-bed demineralizer that has not been pre-borated (OP51.SYS.CS1.OB9)

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 1, 5,
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005 K3.07	
	Importance Rating	3.2	

Residual Heat Removal System: Knowledge of the effect that a loss or malfunction of the RHRs will have on the following: Refueling operations

Proposed Question: Common 3

Given the following conditions:

- The Reactor Vessel Head and Upper Internals are removed.
- Core offload is in progress per RFO-102, Refueling Operation.
- One Train of Residual Heat Removal (RHR) is OPERABLE and in service.
- The other Train of RHR is AVAILABLE.

Which of the following actions is taken per RFO-102, Refueling Operations, to preclude the water above the core from boiling if a running RHR Pump trips?

A. Maintaining greater than 23 feet of water above the Reactor vessel flange.

B. Requiring both trains of RHR to be OPERABLE during Refueling.

C. Waiting to begin core offload for at least 75 hours after shutdown.

D. Starting an additional train of RHR prior to the start of Core offload.

Proposed Answer: A

Explanation:

A. Correct. Per RFO-102, the basis for requiring one RHR loop >23 feet is to prevent the water above the core from boiling in the event of a loss of RHR cooling.
B. Incorrect. Plausible because this is a requirement if level is less than 23 feet.
C. Incorrect. Plausible because this is a Spent Fuel Pool limitation.
D. Incorrect. Plausible if thought that this were required, however, it is not.

Technical Reference(s): RFO-102, Steps 4.1.8 through 4.1.11; RFO-102, Step 4.1.3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** or **STATE** how the following concepts or conditions apply to the Residual Heat Removal System:

- Loss of RHR cooling capability (OP51.SYS.RH1.OB14)

Question Source:	Bank #	CPNPP LXR	
	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	10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K5.06	
	Importance Rating	3.5	

Emergency Core Cooling System: Knowledge of the operational implications of the following concepts as they apply to ECCS: Relationship between ECCS flow and RCS pressure

Proposed Question: Common 4

Given the following conditions:

- During a Small Break Loss of Coolant Accident with normal Emergency Core Cooling System (ECCS) availability, the Reactor Coolant System (RCS) pressure stabilizes so that the ECCS flow remains restricted.
- The break is in one of the RCS Cold Legs.
- RCS inventory decreases, leading to partial core uncovery.

Assuming NO operator actions, which of the following is the event that initiates the RCS pressure decrease, allowing higher ECCS flow and re-covering the core?

A. Clearing of the RCS Intermediate Leg (loop seal).

B. Hot leg level drops below mid-loop level.

C. Opening of any Main Steam Safety Valve.

D. Uncovering of the Guide Tube flow holes.

Proposed Answer: A

Explanation:

A. Correct. As outlined in the WOG E-1 Background Guideline.
B. Incorrect. Plausible if thought that this helped RCS pressure to decrease, however, core uncovery must occur for clearing of the loop seal to be accomplished.
C. Incorrect. Plausible because heat removal by the secondary system is required during a Small Break LOCA, however, this is not the event that allows the core recover with fluid.
D. Incorrect. Plausible if thought that this mechanism would help, however, it is clearing of the loop seal that allows pressure to decrease.

Technical Reference(s) WOG E-1 Background Guideline, Page 21 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the events that create pressure equilibrium during a small break loss of coolant accident. (LO21.MCO.MI2.OB02)

SUMMARIZE the normal sequence of events which would occur during a small break loss of coolant accident with normal Emergency Core Cooling Systems. (LO21.MCO.MI2.OB03)

Question Source:	Bank #	CPNPP LXR	
	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	5	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K4.01	
	Importance Rating	2.6	

Emergency Core Cooling System: Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Cooling of centrifugal pump bearings

Proposed Question: Common 5

Given the following condition:

- A Small Break Loss of Coolant Accident has occurred on Unit 1 and Safety Injection has properly actuated.

Which of the following describes how the Centrifugal Charging Pump (CCP) Bearings are cooled in this situation?

A. Component Cooling Water flow through the CCP Lube Oil Cooler.

B. Station Service Water flow through the CCP Lube Oil Cooler.

C. Pump suction from the Refueling Water Storage Tank.

D. Pump suction from the outlet of the RHR Heat Exchanger.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that the CCP lube oil cooler is cooled by CCW.
B. Correct. The CCP lube oil cooler is cooled by SSW and lube oil cools the bearings.
C. Incorrect. Plausible if thought that cool water from RWST keeps bearings cool during ECCS injection and an additional cooling source for lube oil is not required.
D. Incorrect. Plausible if thought that cool water from the outlet of the RHR Heat Exchanger keeps bearings cool during ECCS recirculation and an additional cooling source for lube oil is not required.

Technical Reference(s) OP51.SYS.SI1.LN, Page 27 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause-effect relationships between the Emergency Core Cooling System and the following systems, components or events:

- Chemical and Volume Control System (OP51.SYS.SI1.OB09)

Question Source:	Bank #	CPNPP LXR	
	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	7, 8
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 A1.01	
	Importance Rating	2.9	

Pressurizer Relief / Quench Tank System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank water level within limits

Proposed Question: Common 6

Given the following conditions on Unit 2:

- Annunciator 2-ALB-5B, Window 4.3 - PRT LVL HI / LO has just gone into alarm.
- Volume Control Tank (VCT) level is being maintained in automatic.
- The following Primary System parameters are observed:

Time:	**1000 hours**	**1100 hours**
Pressurizer Relief Tank Level:	86%	88%
Pressurizer Relief Tank Temperature:	96ºF	96ºF
Pressurizer Level:	45%	45%
Tavg:	570ºF	570ºF
Containment Temperature:	102ºF	100ºF

Which of the following caused the Pressurizer Relief Tank level increase?

A. Pressurizer PORV leakage.

B. Containment temperature change.

C. Letdown Relief Valve (inside Containment) leakage.

D. Seal Return Relief Valve (inside Containment) leakage.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the PORVs are directed to the PRT, however, one would expect PRT temperature to also rise.
B. Incorrect. Plausible because Containment temperature is changing, however, there is not a corresponding change in PRT temperature and this would be expected.
C. Incorrect. Plausible because the Letdown Relief Valve is directed to the PRT, however, given the temperature of Letdown an expected temperature increase in the PRT would occur.
D. Correct. Given the temperature of the seal return water and the corresponding increase in PRT level without an increase in temperature; this is the source of the water.

Technical Reference(s)	ALM-0052B, 2-ALB-5B, Window 4.3	Attached w/ Revision # See Comments / Reference
	OP51.SYS.RC1, Figure 17	
	OP51.SYS.RC1.LN. Page 28	

Proposed references to be provided during examination: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause-effect relationship between the Reactor Coolant System and the following systems, components or events:

- Pressurizer Relief Tank (OP51.SYS.RC1.OB11)

Question Source:	Bank #	CPNPP LXR	
	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	3, 5
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 K1.02	
	Importance Rating	3.3	

Component Cooling Water System: Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: Loads cooled by CCWS

Proposed Question: Common 7

Given the following conditions:

- Unit 1 is at full power, with a normal Component Cooling Water (CCW) System alignment.
- Subsequently, 1-HV-4701, RCP CLR CCW RET ISOL VLV, is observed with the following indication;
  - 1-HS-4701, green light LIT and red light DARK.

Which of the following describes the effect on CCW flow indications as a result of this condition?

Loss of ALL Reactor Coolant Pump…

A. Lower Bearing Lube Oil Cooler CCW Return flows.
Motor Air Cooler CCW Return flows.
Thermal Barrier Cooler CCW Return flows.

B. Upper Bearing Lube Oil Cooler CCW Return flows.
Lower Bearing Lube Oil Cooler CCW Return flows.
Motor Air Cooler CCW Return flows.

C. Upper Bearing Lube Oil Cooler CCW Return flows.
Motor Air Cooler CCW Return flows.
Thermal Barrier Cooler CCW Return flows.

D. Upper Bearing Lube Oil Cooler CCW Return flows.
Lower Bearing Lube Oil Cooler CCW Return flows.
Thermal Barrier Cooler CCW Return flows.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Lower Bearing Lube Oil Cooler and Motor Air Coolers are correct, however, the Thermal Barrier Cooler has a separate return via a common supply.
B. Correct. The common return for Upper Bearing Lube Oil Cooler, Lower Bearing Lube Oil Cooler, and Motor Air Coolers is isolated.
C. Incorrect. Plausible because Upper Bearing Lube Oil Cooler and Motor Air Coolers are correct, however, the Thermal Barrier Cooler has a separate return via a common supply.
D. Incorrect. Plausible because Upper and Lower Bearing Lube Oil Coolers are correct, however, the Thermal Barrier Cooler has a separate return via a common supply.

Technical Reference(s) OP51.SYS.CC1.LP, Figure 4 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the functions, operation and interlocks of the following Component Cooling Water System components:

- u-HV-4701 and u-HV-4708, RCP Cooler CCW Return Isolation Valves (OP51.SYS.CC1.OB02)

Question Source: Bank # CPNPP LXR
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 4, 7
55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 K2.02	
	Importance Rating	3.0	

Component Cooling Water System: Knowledge of bus power supplies to the following: CCW pump, including emergency backup

Proposed Question: Common 8

Given the following conditions:

- Unit 2 is in MODE 1.
- An XST 1 Transformer fault has just occurred.
- Component Cooling Water (CCW) Pump 2-01 was running.
- CCW Pump 2-02 is in standby.

If Safeguards Bus 2EA2 receives an 86-2 Lockout, which of the following identifies when CCW Pump 2-02 is started?

A. After the 86-2 Lockout on Bus 2EA2 is reset, due to low pressure on Train A CCW.

B. When directed by procedure, the pump is manually started.

C. At the appropriate step on the Blackout Sequencer, the pump is started.

D. Immediately after Bus 2EA2 is reenergized, due to low pressure on Train A CCW.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that the Lockout must be reset, however, the EDG will load the Bus with an 86-2 Lockout present.
B. Incorrect. Plausible if thought it would not start automatically as the pump was not previously running, however, the BOS will start the pump.
C. Correct. The Blackout Sequencer will start the CCW Pump 2-02 at Step 2.
D. Incorrect. Plausible if thought the pump would AUTO start immediately upon reenergizing the bus, however, the BOS Auto Lockout is present and prevent the pump from starting.

Technical Reference(s) OP51.SYS.ES3.LN, Pages 10 & 20 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPARE** and **CONTRAST** the function and operation of the BOS and SIS Operator Lockouts and Automatic Lockouts. (OP51.SYS.ES3.OB10)

Question Source: Bank # CPNPP LXR
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 4, 7, 8
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 A3.02	
	Importance Rating	3.6	

Pressurizer Pressure Control System: Ability to monitor automatic operation of the PZR PCS including: PZR pressure

Proposed Question: Common 9

Given the following conditions on Unit 2:

- 2-PK-455A, Pressurizer Master Pressure Controller will not control in AUTO.
- 2-PK-455A was placed in MANUAL and the demand lowered to 0%.

With NO further operator action, which of the following will be the response of the plant?

Pressurizer pressure will...

A. INCREASE until 2-PCV-456 opens. 2-PCV-455A will remain CLOSED.

B. INCREASE until 2-PCV-455A opens. 2-PCV-456 will remain CLOSED..

C. DECREASE due to all Control and Backup Heaters remaining OFF. The Spray Valves remain CLOSED.

D. DECREASE due to all Control and Backup Heaters remaining OFF. The Spray Valves will OPEN.

Proposed Answer: A

Explanation:

A. Correct. This is the controller response when placed in MANUAL with 0% demand. Heaters turn on and Spray Valves remain closed, PCV-456 will open at 2335 PSIG and PCV-455A will remain closed.
B. Incorrect. Plausible if thought that PCV-455A will open, however with PK-455A in manual at 0% demand the PZR heaters will energize and Sprays will not open and pressure will rise to 2335 PSIG but only PV-456 will open and PV-455A will remain closed.
C. Incorrect. Plausible if thought that the Control and Backup Heaters remain OFF when PK-455A is taken to 0% demand, however the Heaters will actually come on.
D. Incorrect. Plausible if thought that the Control and Backup Heaters remain OFF when PK-455A is taken to 0% demand, however the Heaters will actually come on. The Spray Valves will remain closed with PK-455A taken to 0% demand.

Technical Reference(s) OP51.SYS.PP1.LN, Pages 12 & 13
OP51.SYS.PP1.LN, Figure 4

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **EXPLAIN** the Pressurizer Pressure and Level Control System design features which provide for the trips, permissives and interlocks associated with the following:

- PRZR PORVS Open Interlock in AUTO (OP51.SYS.PP1.OB07)

Question Source: Bank # CPNPP LXR
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 7
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 A1.07	
	Importance Rating	3.7	

Pressurizer Pressure Control System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR PCS controls including: RCS pressure

Proposed Question: Common 10

Given the following conditions:

- Unit 1 is in MODE 4
- Reactor Coolant System (RCS) temperature is 250ºF.
- 1-PT-405, RCS Wide Range Pressure Transmitter is out of service.
- The associated train of Low Temperature Overpressure Protection (LTOP) is removed from service.
- 1-PT-403, RCS pressure transmitter fails LOW.

Which of the following describes the consequence of the 1-PT-403 failure?

A. 1-PCV-456, PRZR PORV will open and depressurize the RCS.

B. 1-PCV-455A, PRZR PORV will open and depressurize the RCS.

C. All RCS Low Temperature Overpressure Protection via the PORVs will be lost.

D. RCS LTOP will still be provided by 1-PCV-455A, PRZR PORV.

Proposed Answer: C

Explanation:
A. Incorrect. Plausible if thought that with PT-403 failed low, PCV-456 would still function.
B. Incorrect. Plausible if thought that PCV-455A is available with PT-405 being out of service.
C. Correct. LTOP uses wide range RCS temperature, PT-405 for Train A and PT-403 for Train B. When PT-403 fails, no PORVs will automatically open.
D. Incorrect. Plausible if thought that PCV-455A is available with PT-405 being out of service.

Technical Reference(s) OP51.SYS.PP2.LN, Pages 10 & 11 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** how the Low Temperature Overpressure Protection System Main Control Board/Plant Computer controls, alarms and indications are used to predict, monitor and control changes in the system. (OP51.SYS.PP2.OB11)

Question Source:	Bank #	CPNPP LXR	
	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	7	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K3.02	
	Importance Rating	3.2	

Reactor Protection System: Knowledge of the effect that a loss or malfunction of the RPS will have on the following: Turbine generator

Proposed Question: Common 11

Given the following conditions:

- The plant is at 20% power with all systems operating normally.
- A valid Reactor Trip signal is generated and the Reactor Trip Breakers fail to open.

Which of the following identifies the impact on the Main Turbine?

A. P-4 is NOT enabled and the Main Turbine will NOT trip.

B. P-9 is NOT enabled and the Main Turbine will trip.

C. P-4 is enabled and the Main Turbine will trip.

D. P-9 is enabled and the Main Turbine will NOT trip.

Proposed Answer: A

Explanation:

A. Correct. If the Reactor Trip Breakers fail to open, P-4 is NOT enabled and the Main Turbine will NOT trip.
B. Incorrect. Plausible because P-9 is not enabled until power is greater than 50%, however, this Permissive generates a Reactor Trip if a Turbine Trip occurs.
C. Incorrect. Plausible if thought that P-4 is enabled, however, it is not enabled and the Main Turbine will not trip.
D. Incorrect. Plausible because P-9 is not enabled until power is greater than 50%, however, this Permissive generates a Reactor Trip if a Turbine Trip occurs.

Technical Reference(s): OP51.SYS.ES1.LN, Pages 67 & 68; OP51.SYS.ES1.LN, Pages 70 & 71 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **EXPLAIN** the coincidence and setpoints associated with the following Permissives and Controls:

- P-4 Reactor Trip & P-9 Turbine Trip (OP51.SYS.ES1.OB06)

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	6, 7	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K5.01	
	Importance Rating	3.3	

Reactor Protection System: Knowledge of the operational implications of the following concepts as they apply to the RPS: DNB

Proposed Question: Common 12

Which of the following Reactor Protection System Trips is NOT designed to protect against Departure from Nuclear Boiling (DNB)?

A. Pressurizer Low Pressure.

B. Overtemperature N-16.

C. Reactor Coolant Pump Undervoltage.

D. Intermediate Range High Flux.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that the Pressurizer Low Pressure Trip was only concerned with void formation.
B. Incorrect. Plausible because Overtemperature N-16 could be construed to imply a high flux type of trip which would not protect against DNB.
C. Incorrect. Plausible because the RCP Undervoltage Trip may be considered differently from the RCP Under Frequency Trip. With the latter, under frequency implies a loss of speed which converts to a loss of flow.
D. Correct. There is no credit taken in the Accident Analysis for this trip.

Technical Reference(s) OP51.SYS.ES1.LN, Pages as listed — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** or **STATE** how the following concepts or conditions apply to the Reactor Protection System:

- DNB (OP51.SYS.ES1.OB07)

Question Source:

Bank #	CPNPP LXR	
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 __7__
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 K4.03	
	Importance Rating	3.9	

Engineered Safety Features Actuation System: Knowledge of the ESFAS design feature(s) and/or interlock(s) that provide for the following: Main Steam Isolation System

Proposed Question: Common 13

Given the following conditions:

- The plant is at 75% power with power ascension in progress at 8% per hour.
- All Steam Generator pressures are slowly lowering.
- Containment temperature, pressure, and humidity are slowly rising.
- T_{AVE} is lowering.

Which of the following Engineered Safety Feature Actuation Signals is designed to prevent exceeding the Containment pressure limit?

A. Containment Isolation Phase A.

B. Containment Isolation Phase B.

C. Main Steam Line Isolation.

D. Safety Injection Actuation.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because all non-essential lines in or out of Containment are isolated by Phase A, however, Phase A isolation is designed to prevent radiological release from Containment.
B. Incorrect. Plausible because all CCW support for RCPs in or out of Containment is isolated by Phase B, however, Phase B isolation is designed to prevent radiological release from Containment.
C. Correct. MSLI is designed to minimize excessive cooldown and Containment pressure.
D. Incorrect. Plausible because the SI signal starts Containment Spray and initiates Phase A isolation, however, the SI signal starts pumps and positions valves to ensure the core is cooled during an accident.

Technical Reference(s) FSAR Section 7.0, I & C, Page 130; OP51.SYS.ES1.LN, Pages as listed — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** the ESF Systems and **STATE** their functions. (OP51.SYS.ES1.OB03)
LIST and **EXPLAIN** the coincidence and setpoints associated with the following ESF actuation signals:

- Steamline Isolation (OP51.SYS.ES1.OB05)

Question Source: Bank # CPNPP LXR
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 7
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	022 A2.05	
Importance Rating	3.1	

Containment Cooling System: Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Major leak in CCS

Proposed Question: Common 14

Given the following conditions:

- Unit 1 is at 50% power.
- The following Annunciators are in alarm:
  - 1-ALB-02B, Window 1.12 - CNTMT FN CLR 3 & 4 CNDS LVL HI.
  - 1-ALB-02B, Window 3.12 - CNTMT FN CLR 3 & 4 CND FILL RATE HI
  - X-ALB-11C, Window 1.11 - CH WTR SRG TK LVL HI-HI / LO-LO is alarming intermittently due to a low level.
- A Containment Ventilation Chilled Water System leak has been diagnosed.
- 1-HS-5413A, CNTMT FN CLR FN 3, and 1-HS-5417A, CNTMT FN CLR FN 4 were stopped.

Which of the following describes the impact of securing Containment Fan Coolers 3 and 4?

The Ventilation Chilled Water leak will...

A. be isolated; the Unit must be tripped and EOP-0.0A, Reactor Trip or Safety Injection, entered.

B. be isolated; continued plant operation per IPO-003A, Power Operation, is permitted.

C. NOT be isolated; enter Containment per STA-620, Containment Entry, and locally isolate the leak.

D. NOT be isolated; refer to SOP-814, Ventilation Chilled Water, and isolate leak from Ventilation Panel CV-03.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that with 2 Containment Fan Coolers out of service the unit must be tripped.
B. Incorrect. Plausible because continued plant operation is allowed, however, the leak has not been isolated.
C. Correct. A Containment entry must be made to locally isolate the leak.
D. Incorrect. Plausible because the leak is not isolated, however, leak isolation must be performed inside Containment.

Technical Reference(s): 1-ALB-02B, Windows 1.12 & 3.12; X-ALB-11C, Window 1.11 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the functions, operation and interlocks of the following Ventilation Chilled Water System components:

- Containment Air Cooling and Recirculation Units (OP51.SYS.CH2.OB03)

IDENTIFY and **DESCRIBE** the Main Control Board/Plant Computer controls, alarms and indications associated with the Ventilation Chilled Water System. (OP51.SYS.CH2.OB06)

Question Source: Bank # CPNPP LXR
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 4, 9, 10
55.43 ______

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	026 G 2.4.21	
Importance Rating	4.0	

Containment Spray System: Emergency Procedures/Plan: 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.

Proposed Question: Common 15

Given the following conditions:

- A Loss of Coolant Accident has occurred inside Containment.
- Containment pressure is 20 PSIG.
- Containment temperature is 260ºF.
- Containment sump level is 814 feet.
- Containment radiation level is 10R/hr.
- Train A Containment Spray System is in service.

Which of the following procedures should be entered?

A. A RED path condition exists.
Enter FRZ-0.1A, Response to High Containment Pressure.

B. An ORANGE path condition exists.
Enter FRZ-0.1A, Response to High Containment Pressure.

C. An ORANGE path condition exists.
Enter FRZ-0.2A, Response to Containment Flooding.

D. A YELLOW path condition exists.
Enter FRZ-0.3A, Response to High Containment Radiation.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the procedure entry is correct, however, Containment pressure would have to be greater than 50 PSIG for a red path to exist.
B. Correct. With Containment pressure at 20 PSIG, an ORANGE path condition exists and FRZ-0.1A must be entered.
C. Incorrect. Plausible because an ORANGE path condition would exist, however, containment sump level must be greater than 816 feet.
D. Incorrect. Plausible because this is the correct path and procedure for high containment radiation, however, radiation level must be greater than 20 R/hr.

Technical Reference(s) FRZ-0.1A, CSF Status Tree Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given the Containment CSFST (and necessary parameter data or access to the data), **DETERMINE** whether entry into FRZ-0.1A/B is applicable, and, if so, **STATE** the severity of the challenge. (LO21.ERG.FZ1.OB02)

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 10
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 A4.07	
	Importance Rating	2.8	

Main and Reheat Steam System: Ability to manually operate and/or monitor in the control room: Steam dump valves

Proposed Question: Common 16

Given the following conditions:

- Reactor Coolant System Tave is 557ºF.
- 1-PK-507, Steam Dump Pressure Controller is at 50% demand in MANUAL.
- Steam Dump Mode Select Switch is in the STEAM PRESSURE Mode.
- Steam Dump Interlock Select Switch is in the NEUTRAL position.

Which of the following will be the status of the Steam Dump Cooldown Valves?

The Steam Dump Cooldown Valves are...

A. NOT open.

B. open 50%.

C. full open.

D. throttled to maintain 557ºF.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that no demand signal is present with the Steam Dump Pressure Controller in Manual.
B. Incorrect. Plausible because the Steam Dump Controller is set at 50%, however, with the conditions listed the valves will go full open.
C. Correct. With the Steam Dump Controller positioned as listed and with temperature above the Bypass Interlock setpoint, the valves will be full open.
D. Incorrect. Plausible if thought that the Bypass Interlock signal will throttle close the valves to fully closed at 553ºF.

Technical Reference(s) OP51.SYS.SD1.LN, Pages 14 & 26 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the performance and design attributes of the following Steam Dump System components, flowpaths and features:

- Mode Selector Switch and Interlock Selector Switch (OP51.SYS.SD1.OB02)

LIST the input signals to the Steam Dump System, and **DESCRIBE** how these signals are generated and utilized in controlling RCS/Secondary fluid conditions. Description should include the following:

- Number of dump valves
- Number of Banks and number of Dump valves per Bank (OP51.SYS.SD1.OB03)

Question Source: Bank # CPNPP LXR
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 7
55.43 ______

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	059 A4.03	
Importance Rating	2.9	

Main Feedwater System: Ability to manually operate and/or monitor in the control room: Feedwater control during power increase and decrease

Proposed Question: Common 17

Given the following condition:

- A power ascension is in progress on Unit 2.

Which of the following describes the operation of the Unit 2 Feedwater Preheater Bypass Valve?

Unit 2 Feedwater Preheater Bypass Valve...

A. automatically OPENS whenever the water hammer permissives are cleared and the Feedwater Isolation Valve is OPENED.

B. cannot be OPENED until the water hammer permissive is satisfied.

C. automatically OPENS when the Feedwater Isolation Valve is CLOSED due to a Feedwater Isolation Signal.

D. automatically CLOSES whenever the water hammer permissives are cleared and the Feedwater Isolation Valve is OPENED.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the water hammer permissives must be cleared, however, this condition closes the Unit 2 Feedwater Preheater Bypass Valve.
B. Incorrect. Plausible if thought that the water hammer permissives must be satisfied.
C. Incorrect. Plausible because the 1st part of the answer is correct, however, a Feedwater Isolation Signal cannot be present.
D. Correct. This is the design of the Unit 2 Feedwater Preheater Bypass Valve. Additional information is available in IPO-003B, Power Operation, Sections 5.2, 5.3, and 5.7.

Technical Reference(s) OP51.SYS.MF1.LN, Pages 51 & 72 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **EXPLAIN** the Main Feedwater System design features which provide for the trips, permissives, and interlocks associated with the following:

- Main Feedwater System water hammer interlocks (OP51.SYS.MF1.OB11)

Question Source:	Bank #	CPNPP LXR	
	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	4, 7
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	061 K5.01	
Importance Rating	3.6	

Auxiliary/Emergency Feedwater System: Knowledge of the operational implications of the following concepts as they apply to the AFW: Relationship between AFW flow and RCS heat transfer

Proposed Question: Common 18

Given the following conditions:

- Unit 1 is in MODE 1.
- The running Main Feedwater Pump just tripped on low oil pressure.
- The crew responded by tripping the Main Turbine which was loaded to 160 MWe.

Which of the following identifies the maximum Reactor power level to ensure adequate Auxiliary Feedwater flow?

A. 2%

B. 5%

C. 6%

D. 10%

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that POAH could not be achieved with Auxiliary Feedwater flow.
B. Incorrect. Plausible if thought that MODE 1 could not be achieved with Auxiliary Feedwater flow.
C. Correct. As identified in ABN-403 Caution.
D. Incorrect. Plausible because AFW must be placed in standby readiness prior to exceeding 10%.

Technical Reference(s) ABN-403, Step 2 Caution Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** the maximum reactor power equivalent that can be maintained utilizing AFW System flow Reactor Coolant System. (OP51.SYS.AF1.OB12)

Question Source:	Bank #	CPNPP LXR	
	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	5	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 A3.03	
	Importance Rating	3.9	

Auxiliary/Emergency Feedwater System: Ability to monitor automatic operation of the AFW including: AFW steam generator level control on automatic start

Proposed Question: Common 19

On Unit 2, which of the following describes the response to the AUTO start of the Motor Driven Auxiliary Feedwater Pumps (MDAFWP)?

A. Condensate Storage Tank Discharge Valves OPEN.

B. MDAFWP Flow Control Valves trip to AUTO and fully OPEN.

C. Main Feedwater Preheater Bypass Valves CLOSE.

D. Main Feedwater Isolation Valves CLOSE.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that the Condensate Storage Tank Discharge Valves would open on the AFW actuation signal, however, they close.
B. Correct. As described in SOP-304B, Auxiliary Feedwater System.
C. Incorrect. Plausible because it could be thought that the FPBVs receive a close signal to ensure AFW flow into SGs, however, the AFW line enters the Feedwater Preheater Bypass Line downstream of the FPBVs and a check valve and there is no closure signal to the FPBVs.
D. Incorrect. Plausible because it could be thought that the FIVs receive a close signal to ensure AFW flow into the SGs, however, AFW enters the Preheater Bypass Line and the closure of the FSBVs ensures AFW flow into the SGs.

Technical Reference(s)		
	SOP-304B, Step 5.2.1.F.1) Note	Attached w/ Revision # See Comments / Reference
	SOP-304B, Step 4.2	

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** and **ILLUSTRATE** the Auxiliary Feedwater System design features which provide for the automatic starts, trips, permissives and interlocks associated with the following:

- Auxiliary feedwater flow control valves (OP51.SYS.AF1.OB08)

Question Source:	Bank #	CPNPP LXR	
	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	4, 7	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 G 2.2.12	
	Importance Rating	3.7	

AC Electrical Distribution System: Equipment Control: Knowledge of surveillance procedures

Proposed Question: Common 20

Given the following conditions:

- Unit 1 is in MODE 3.
- Train A Emergency Diesel Generator 1-01 is declared INOPERABLE following surveillance testing.

Which of the following actions are required to be taken within one (1) hour of determining the Emergency Diesel Generator is INOPERABLE?

A. Perform OPT-215, Class 1E Electrical Systems Operability, to verify proper breaker alignments.

B. Verify operation of the Turbine Driven Auxiliary Feedwater Pump per OPT-206A, AFW System.

C. Initiate surveillance testing of the Train B Emergency Diesel Generator per OPT-214A, Diesel Generator Operability Test.

D. Perform OPT-414A, SI/Blackout Sequencers, to verify operability of the Blackout Sequencer.

Proposed Answer: A

Explanation:

A. Correct. This procedure meets the surveillance testing requirements of Technical Specification LCO 3.8.1, AC Sources-Operating.
B. Incorrect. Plausible because the Turbine Driven Auxiliary Feedwater Pump is considered a redundant feature, however, this only applies when 2 required Offsite Power Sources are INOPERABLE.
C. Incorrect. Plausible because surveillance testing would be required if both Emergency Diesel Generators were declared INOPERABLE, however, this action must be performed within 2 hours.
D. Incorrect. Plausible because Blackout Sequencer testing is part of this Technical Specification, however, it is the EDG not the Blackout Sequencer that was declared INOPERABLE.

Technical Reference(s) OPT-215, Section 1.0 Attached w/ Revision # See Comments / Reference

Technical Specification LCO 3.8.1

Proposed references to be provided during examination: None

Learning Objective: Given a Technical Specification or a Technical Specification situation, **DIAGNOSE** the situation and **APPLY** the LCO and SR Applicability of Section 3.0 to **DETERMINE** all corrective actions. (LO21.RLS.SL1.OB12)

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 10
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 A4.04	
	Importance Rating	2.6	

AC Electrical Distribution System: Ability to manually operate and/or monitor in the control room: Local operation of breakers

Proposed Question: Common 21

Given the following condition:

- Residual Heat Removal Pump (RHR) 1-01 is operating during a plant heat up when the control power fuses blow.

Which of the following describes how the Main Control Board RHR Pump indication and local breaker control is affected by the loss of control power?

Main Control Board RHR Pump red / green running indications will be...

A. lost.
Local OPEN / CLOSE light indication is available, and local breaker control will be lost until control power is restored.

B. lost.
Local OPEN / CLOSE mechanical indication is available, and local breaker control is possible without the control power.

C. available.
Local OPEN / CLOSE light indication is available, and local breaker control is possible without the control power.

D. available.
Local OPEN / CLOSE mechanical indication is available, and local breaker control will be lost until control power is restored.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Main Control Board red / green running indications will be lost, however, local breaker control is possible without control power.
B. Correct. With a loss of control power, Main Control Board red / green running indications will be lost. Local breaker control is still possible.
C. Incorrect. Plausible because local breaker control is possible without control power, however, local OPEN / CLOSE light indication is NOT available.
D. Incorrect. Plausible because local OPEN/CLOSE indication is available, however, Main Control Board indications are lost and local control is available.

Technical Reference(s): STA-694, Attachment 8.B; Electrical Print E1-0031, Sheet 49 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** the effect a loss of the following systems has on the Residual Heat Removal System and components:

- DC Power (OP51.SYS.RH1.OB17)

Question Source: Bank # CPNPP LXR
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam CPNPP 2009

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

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

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	063 K4.01	
Importance Rating	2.7	

DC Electrical Distribution System: Knowledge of the DC electrical system design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control

Proposed Question: Common 22

Which of the following identifies the types of trips or protective isolation circuits associated with the DC Distribution Panels?

A. Overcurrent only.

B. Ground fault only.

C. Overcurrent and undervoltage only.

D. Overvoltage, ground fault, and undervoltage.

Proposed Answer: A

Explanation:

A. Correct. This is the only trip associated with the DC Distribution Panels.
B. Incorrect. Plausible because ground fault indication does exist, however, it provides BUS TROUBLE indication only in the Control Room.
C. Incorrect. Plausible because an overcurrent trip does exist, however, an undervoltage condition provides BUS TROUBLE indication only in the Control Room.
D. Incorrect. Plausible because all three of these indications exist, however, they only provide BUS TROUBLE indication in the Control Room.

Technical Reference(s) OP51.SYS.DC1.LN, Page 32 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **EXPLAIN** the DC Electrical System design features which provide for the trips, permissives, and interlocks associated with the following:

- Protective Relay actuations (OP51.SYS.DC1.OB12)

Question Source: Bank # CPNPP LXR
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	7
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064 K4.10	
Importance Rating	3.5	

Emergency Diesel Generator System: Knowledge of the EDG system design feature(s) and/or interlock(s) which provide for the following: Automatic load sequencer: blackout

Proposed Question: Common 23

Given the following condition:

- A phase to ground fault (86-2) has opened the Preferred Power Supply Breaker to 6900 V Bus 1EA2.

Which of the following is the response of the Emergency Diesel Generator 1-02?

The Emergency Diesel Generator 1-02...

A. will emergency start but the bus remains deenergized.

B. must be manually started but the bus remains deenergized.

C. will emergency start and loads sequence normally on the Blackout Sequencer.

D. must be manually started and loads sequence normally on the Blackout Sequencer.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because an 86-1 lockout does allow EDG 1-02 to Emergency Start, however the DG supply breaker remains open.
B. Incorrect. Plausible if thought that the 86-2 lockout prevents an Emergency Start, however, EDG 1-02 will emergency start with an 86-2 lockout.
C. Correct. When the Preferred Power Supply Breaker opens, the associated Emergency Diesel Generator will Emergency Start and the Blackout Sequencer loads the bus.
D. Incorrect. Plausible because loads will sequence normally on the Blackout Sequencer, however, the DG 1-02 will Emergency Start.

Technical Reference(s): ABN-602, Step 2.3.3 Note; ABN-602, Section 2.2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPARE** and **CONTRAST** the 6.9Kv bus 86-1 and 86-2 relay operation to include the effect on the preferred, alternate and emergency power supplies. (OP51.SYS.AC2.OB11)

Question Source: Bank # CPNPP LXR
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 7
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 A3.05	
	Importance Rating	2.8	

Emergency Diesel Generator System: Ability to monitor automatic operation of the EDG system including: Operation of the governor control of frequency and voltage control during parallel operation

Proposed Question: Common 24

Given the following conditions:

- Emergency Diesel Generator (EDG) 1-01 is being paralleled to Safeguards Bus 1EA1.
- EDG Breaker 1EG1 is closed with EDG voltage (INCOMING) slightly greater than Safeguards Bus 1EA1 voltage (RUNNING).
- The Unit Supervisor has directed that zero (0) KVARs OUT be maintained following paralleling the EDG with the Safeguards Bus.

Based on the above conditions, which of the following identifies the response of the Emergency Diesel output voltage and what action should be taken?

A. 1.) EDG VAR meter will move in the positive (+) VAR (LAG-OUT) direction.
 2.) Place the EDG Voltage Control Switch in the LOWER position to adjust VAR load.

B. 1.) EDG VAR meter will move in the positive (+) VAR (LAG-OUT) direction.
 2.) Place the EDG Voltage Control Switch in the RAISE position to adjust VAR load.

C. 1.) EDG VAR meter will move in the negative (-) VAR (LEAD-IN) direction.
 2.) Place the EDG Voltage Control Switch in the RAISE position to adjust VAR load.

D. 1.) EDG VAR meter will move in the negative (-) VAR (LEAD-IN) direction.
 2.) Place the EDG Voltage Control Switch in the LOWER position to adjust VAR load.

Proposed Answer: A

Explanation:

A. Correct. With EDG voltage greater than bus voltage when the breaker is closed, a positive VAR load will be "supplied by" the Emergency Diesel Generator. The Voltage Control Switch is placed in LOWER to decrease generator terminal voltage and zero out the VAR load.
B. Incorrect. Plausible because the VAR response is correct, however, this action would only serve to increase the VAR load.
C. Incorrect. Plausible because this would be the correct action if generator voltage were lower than Safeguards Bus voltage when the breaker was closed and it was desired to zero out the VAR load.
D. Incorrect. Plausible because the action is correct, however, this VAR response would occur if generator voltage were less than Safeguards Bus voltage.

Technical Reference(s) SOP-609A, Step 5.2.X Note   Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the location (if applicable) of the following indications and controls, and **DESCRIBE** how each is interpreted or used to predict, monitor, or control changes in the Emergency Diesel Generator System:

- Voltage Regulator Raise/Lower Switch (OP51.SYS.ED1.OB04)

STATE the physical connections and **EVALUATE** the cause-effect relationships between the Emergency Diesel Generator System and the following systems, components or events:

- AC Distribution System (OP51.SYS.ED1.OB08)

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

Question History: Last NRC Exam CPNPP 2009

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

10 CFR Part 55 Content: 55.41 5
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 A2.01	
	Importance Rating	2.5	

Process Radiation Monitoring System: Ability to (a) predict the impacts of the following malfunctions or operations on the PRM System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Erratic or failed power supply

Proposed Question: Common 25

Given the following condition:

- The power supply to the RM-80 portion of X-RE-5895A, Control Room North Intake Radiation Monitor, has just failed.

Which of the following identifies the affect of this failure and action required to remove the Control Room Emergency Filtration from Emergency Recirculation?

A. A high radiation signal is generated.
Place X-HS-5895A, High Radiation Actuation BLOCK/ON handswitch in BLOCK.

B. A low radiation signal is generated.
Place X-HS-5895A, High Radiation Actuation BLOCK/ON handswitch in BLOCK.

C. A high radiation signal is generated.
Place X-HS-5895A, High Radiation Actuation BLOCK/ON handswitch in ON.

D. A low radiation signal is generated.
Place X-HS-5895A, High Radiation Actuation BLOCK/ON handswitch in ON.

Proposed Answer: A

Explanation:

A. Correct. When the power supply for the Control Room Intake Radiation Monitor is deenergized for any reason, it will fail safe and generate a high radiation signal. The Control Room Emergency Filtration System (CREFS) can not be taken out of Emergency Recirculation until X-HS-5895A is taken to BLOCK.

B. Incorrect. Plausible because required action is correct, however, a high radiation signal is generated.

C. Incorrect. Plausible because a high radiation signal is generated, however, placing X-HS-5895A in ON does not allow the CREFS to be taken out of Emergency Recirculation.

D. Incorrect. Plausible if it is believed that loss of power generates a loss of signal, however, a high radiation signal is generated and placing X-HS-5895A in ON does not allow the CREFS to be taken out of Emergency Recirculation.

Technical Reference(s) SOP-706, Sections 5.6 & 5.11 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the location (if applicable) of the following indications and controls, and **DESCRIBE** how each is interpreted or used to predict, monitor, or control changes in the Digital Radiation Monitoring System:

- RM-80 Microprocessor Block Handswitches (OP51.SYS.RM1.OB04)

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 11
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 K2.08	
	Importance Rating	3.1	

Service Water System: Knowledge of bus power supplies to the following: ESF actuated MOVs

Proposed Question: Common 26

Given the following conditions:

- A Loss of Station Service Water has occurred on Unit 1.
- Unit 1 Train A Station Service Water will be supplied from Unit 2 Train A Station Service Water Pump.
- Station Service Water Pump 2-01 Discharge Valve was closed using the Control Room handswitch.

Which of the following identifies the condition of the Station Service Water Pump 2-01 Discharge Valve and the appropriate action to be taken?

Station Service Water Pump 2-01 Discharge Valve is...

A. fully closed. Hold the Control Room Hand Switch in the CLOSE position for an additional 10 seconds to ensure valve closure.

B. 10% open. Open the breaker on Motor Control Center 2EB3-3 and locally CLOSE the valve the last 10%.

C. fully closed. Hold the Control Room Hand Switch in the CLOSE position for an additional 10 seconds until the breaker trips on thermal overload.

D. 10% open. Open the breaker on Motor Control Center 2EB4-3 and locally CLOSE the valve the last 10%.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that this condition would fully close the valve.
B. Correct. Per the guidance in the SOP-501A, the valve remains 10% open when closed from the Control Room and must be closed locally once the valve is deenergized. The power supply for this ESF operated valve is MCC 2EB3.
C. Incorrect. Plausible if thought that this was the action necessary to fully close the valve.
D. Incorrect. Plausible because the valve does remain 10% open and the valve must be locally close the last 10%, however, the power supply for this ESF operated valve is MCC 2EB3.

Technical Reference(s)	SOP-501A, Step 5.7.1.D, & Note OP51.SYS.SW1.LN, Page 28	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the function and operating characteristics for the following Station Service Water System components:

- SSW Pump Discharge Valve (OP51.SYS.SW1.OB05)

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	7, 8	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078 G 2.1.30	
Importance Rating	4.4	

Instrument Air System: Conduct of Operations: Ability to locate and operate components, including local controls

Proposed Question: Common 27

Given the following conditions:

- HV-2452-1 and HV-2452-2, Turbine Driven Auxiliary Feedwater Pump (TDAFW) Steam Supply Valves, were opened during a loss of instrument air.
- After two hours, Instrument Air pressure has been restored and TDAFW Pump shutdown is required.

Which of the following identifies where the valves will be operated from?

HV-2452-1 and HV-2452-2, TDAFW Pump Steam Supply Valves, will be operated...

A. locally in the TDAFW Pump Room.

B. remotely from the Control Room.

C. remotely from the TDAFW Pump Room.

D. locally in the Main Steam Header Room.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because AFW flow to the Steam Generators would be locally controlled from the TDAFW Pump room.
B. Correct. Because the Steam Supply Valve Air Accumulators are designed to last for 7 1/2 hours the valves will be closed from the Control Room.
C. Incorrect. Plausible because it could be thought that the local control panel in the TDAFW Pump Room has remote controls that are functional for the Main Steam Supply Valves.
D. Incorrect. Plausible if thought that the valves were AOVs or MOVs which could be locally controlled.

Technical Reference(s): ABN-301, Step 2.3.13; ABN-301, Attachment 9 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	**IDENTIFY** and **ILLUSTRATE** the Auxiliary Feedwater System design features which provide for the automatic starts, trips, permissives and interlocks associated with the following: • Auxiliary feedwater pump turbine steam supply valves (OP51.SYS.AF1.OB08)

Question Source:	Bank #	CPNPP LXR	
	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	8, 10	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	103 K3.02	
Importance Rating	3.8	

Containment System: Knowledge of the effect that a loss or malfunction of the Containment System will have on the following: Loss of containment integrity under normal conditions

Proposed Question: Common 28

Which of the following would cause entry into the Limiting Condition for Operation for Containment Isolation Valves while in MODE 1?

A. Damage to the operator for 1-LCV-459, LTDN ISOL VLV which prohibits closure of the valve.

B. A galled stem on 1MS-0357, SG 1-03 BLDN DNSTRM ISOL VLV which will not allow the valve to be closed.

C. A large leak from the supply side of 1-8875B, ACCUM 2 N2 SPLY/VENT VLV which prohibits closure of the valve.

D. Damage to the stem for 1-HV-3487, CNTMT INST AIR ISOL which will not allow the valve to be closed.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because valve will not close and is part of the Letdown System, however, it is not a Containment Isolation Valve.
B. Incorrect. Plausible because valve will not close and is part of the SG Blowdown System, however, it is not a Containment Isolation Valve.
C. Incorrect. Plausible because the leak exists on an Accumulator Nitrogen Supply Valve, however, it is not a Containment Isolation Valve.
D. Correct. This is a Containment Isolation Valve that closes on a Phase A signal.

Technical Reference(s) Technical Specification LCO 3.6.3 | OP51.SYS.IA1.LN, Page 27 | Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPARE** and **CONTRAST** Phase A and Phase B isolation signals including types of valves utilized to perform each function and expected system and operator response following an actuation signal which provides incomplete isolation (OP51.SYS.CY1.OB20)

LIST and **DESCRIBE** the following Technical Specifications (i.e. LCOs, action statements and conditional surveillance requirements of one hour and less, if applicable) for the Containment Systems:

- Containment Isolation Valves 3.6.3 (OP51.SYS.CY1.OB32)

Question Source: Bank # CPNPP LXR
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 9, 10
55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	071 K5.04	
	Importance Rating	2.5	

Waste Gas Disposal System: Knowledge of the operational implications of the following concepts as they apply to the Waste Gas Disposal System: Relationship of hydrogen/oxygen concentration to flammability

Proposed Question: Common 29

In accordance with RWS-201, Waste Gas Processing System, which of the following combinations of Oxygen and Hydrogen concentration (by volume) in the Waste Gas Holdup System would require immediate action to suspend all Waste Gas additions to the system?

	O_2%	H_2%
A.	2.7	3.2
B.	3.3	3.6
C.	3.5	3.8
D.	4.1	4.2

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because H_2 exceeds 3%, however, O_2 does not exceed 3%.
B. Incorrect. Plausible because O_2 exceeds 3%, however, H_2 does not exceed 4%.
C. Incorrect. Plausible because O_2 exceeds 3%, however, H_2 does not exceed 4%.
D. Correct. When O_2 is equal to or greater than 3%, H_2 cannot be greater than 4%.

Technical Reference(s) RWS-201, Section 4.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the basis for the precautions, limitations for following procedures associated with the Gaseous Waste Processing System:

- RWS-201, Gaseous Waste Processing System (OP51.SYS.GH1.OB07)

ANALYZE the risk significance of any major component being out of service. (OP51.SYS.GH1.OB11)

Question Source:	Bank #	CPNPP LXR	
	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	10, 12	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	016 K1.02	
	Importance Rating	3.4	

Non-Nuclear Instrumentation System: Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems: PZR LCS

Proposed Question: Common 30

Given the following conditions:

- A Reactor Startup is in progress.
- As Reactor power enters the Power Range, Pressurizer Level and Level Setpoint begin to rise with no apparent change in Charging or Letdown flow.

Which of the following would cause this change to occur?

A. Input to the Pressurizer Level Control System from the Power Range Nuclear Instruments.

B. Main Steam header pressure lowering when the Point of Adding Heat is reached.

C. Increased Reactor Coolant System T_{AVE} when the Point of Adding Heat is reached.

D. Pressurizer Heaters energizing when Steam Dumps throttle further open.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that Level Setpoint originated from a nuclear instrument as opposed to a temperature instrument.
B. Incorrect. Plausible because steam pressure does change, however, it rises not lowers.
C. Correct. When the Point of Adding Heat is reached, the RCS heats up and causes an expansion of liquid into the Pressurizer. Because the Level Setpoint is derived from temperature, it will also rise.
D. Incorrect. Plausible because Steam Dumps do open but not to the point were RCS pressure lowers far enough to cause PRZR Heaters to energize.

Technical Reference(s) OP51.SYS.PP1.LN, Pages 22 & 24 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** how the Pressurizer Pressure and Level Control System Main Control Board/Plant Computer controls, alarms and indications are used to predict, monitor and control changes in the system. (OP51.SYS.PP1.OB16)

Question Source:	Bank #	CPNPP LXR	
	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	5, 7	
	55.43		

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	035 G 2.1.7	
Importance Rating	4.4	

Steam Generator System: Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Proposed Question: Common 31

Given the following conditions:

- Unit 1 is at 21% power with all systems in normal alignment.
- The Main Generator is synchronized to the grid.
- Main Steam Isolation Valve 1-01 closed on a spurious signal.
- The Reactor does NOT trip.
- NO operator actions have been performed.

Which of the following describes the response of Loop ΔT and Steam Generator (SG) steam pressure in the affected loop?

Loop ΔT…

A. rises and SG steam pressure rises.

B. lowers and SG steam pressure rises.

C. rises and SG steam pressure lowers.

D. lowers and SG steam pressure lowers.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because SG steam pressure rises, however, Loop ΔT will lower.
B. Correct. In the affected loop, RCS ΔT lowers to zero and SG steam pressure rises because heat removal is minimal.
C. Incorrect. Plausible because SG steam pressure could lower if the affected loop RCP had stopped, however, Loop ΔT will lower.
D. Incorrect. Plausible because Loop ΔT will lower, however, SG steam pressure will rise.

Technical Reference(s) LO21.MCO.TA9.LN, Page 11 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** a design loss of load accident and **STATE** the major concern following a Loss of Load accident and **DESCRIBE** the conclusions with respect to DNB and system integrity (LO21.MC0.TA9.OB01)

Question Source: Bank # CPNPP LXR
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 3, 4, 5
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	033 K3.01	
	Importance Rating	2.6	

Spent Fuel Pool Cooling System: Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Area ventilation systems

Proposed Question: Common 32

Given the following conditions:

- A loss of Spent Fuel Cooling in Spent Fuel Pool #2 is in progress.
- The following PC-11, Digital Radiation Monitoring System monitors are in alarm:
  - SFP-002 LRAM SFP 2 N. WALL (X-RE-6273).
  - SFP-001 LRAM SFP 2 E. WALL (X-RE-6272).
- ABN-908, Fuel Handling Accident, has been entered and all automatic and operator actions have been completed.

Which of the following identifies the ventilation alignment in the Fuel Building?

A. All Spent Fuel Pool ventilation is automatically stopped.

B. All Spent Fuel Pool ventilation is manually stopped.

C. One supply fan in Spent Fuel Pool X-02 is running.

D. One exhaust fan in Spent Fuel Pool X-02 is running.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that the Area Radiation Monitors automatically isolated ventilation.
B. Incorrect. Plausible if thought that all ventilation is left running to clean the air, however, only one exhaust fan is kept running to maintain a negative pressure in the Fuel Building.
C. Incorrect. Plausible if thought that this would maintain the correct pressure within the Fuel Building, however, it would tend to pressurize the Fuel Building and release radiation.
D. Correct. Leaving one exhaust fan running in the Spent Fuel Pool area is required per ABN-908.

Technical Reference(s) ABN-908, Step 3.3.3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause-effect relationship between the Digital Radiation Monitoring System and the following systems, components or events:

- Plant Ventilation (OP51.SYS.RM1.OB10)

EVALUATE the effect a loss of the Spent Fuel Pool Cooling and Cleanup System has on the following:

- Area and ventilation radiation monitoring systems (OP51.SYS.SF1.OB17)

Question Source: Bank # CPNPP LXR
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 10, 12
55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	011 A1.02	
Importance Rating	3.3	

Pressurizer Level Control System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR LCS controls including: Charging and letdown flows

Proposed Question: Common 33

Given the following conditions:

- Unit 1 is at 48% power.
- Pressurizer Level Control System is in AUTO.
- 1-8106, Charging Pump to RCS Isolation Valve, fails closed due to a control switch circuit malfunction and cannot be reopened.
- Letdown has been isolated.

Which of the following describes the response of Pressurizer level to this event?

Pressurizer level will….

A. stabilize when letdown is isolated.

B. lower at a rate consistent with seal leakoff flow.

C. rise at a rate consistent with seal injection flow.

D. rise at a rate consistent with seal injection minus seal leakoff flow.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because seal injection is upstream of the isolation valves and seal leakoff is not isolated on this event, so the Pressurizer is NOT bottled up.
B. Incorrect. Plausible because seal leakoff is not isolated, however, neither is seal injection.
C. Incorrect. Plausible because seal injection flow is available with this failure, however, leak off flow must be included.
D. Correct. Seal Injection will provide makeup. Seal leakoff will be the only source of letdown for this event prior to any other actions.

Technical Reference(s) OP51.SYS.CS1.LN, Pages 58 & 88 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause-effect relationships between the CVCS and the following systems, components or events:

- Interface between charging flow path and seal return flowpath (OP51.SYS.CS1.OB10)

Question Source:	Bank #	CPNPP LXR	
	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	7	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017 K6.01	
	Importance Rating	2.7	

In-Core Temperature Monitor System: Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors

Proposed Question: Common 34

Given the following conditions:

- Several Core Exit Thermocouples (CETs) have failed.
- All failed CETs have been removed from active input to the Plant Computer.

Which of the following combinations of CETs meets the minimum OPERABILITY requirements per Technical Specifications?

A. Two (2) Train A and two (2) Train B CETs per core quadrant.

B. Two (2) Train A or two (2) Train B CETs per core quadrant.

C. Four (4) Train A and four (4) Train B CETs per core quadrant.

D. Four (4) Train A or four (4) Train B CETs per core quadrant.

Proposed Answer: A

Explanation:

A. Correct. Technical Specifications requires 2 CHANNELS OPERABLE with 2 CETs per quadrant.
B. Incorrect. Plausible because the CHANNELS required is correct, however, both Trains must be OPERABLE.
C. Incorrect. Plausible because both Trains must be OPERABLE, however, only 2 CETs each per quadrant ae required.
D. Incorrect. Plausible if thought that this was the required number of CHANNELS on either Train.

Technical Reference(s): Technical Specification Table 3.3.3-1; OPT-112A-1, Page 6 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **DESCRIBE** the following Technical Specifications (i.e., LCOs, action statements and conditional surveillance requirement of one hour and less, if applicable) for the Reactor Coolant System:

- Post Accident Monitoring Instrumentation (OP51.SYS.RC1.OB18)

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	2, 10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	002 K6.02	
	Importance Rating	3.6	

Reactor Coolant System: Knowledge of the effect that a loss or malfunction of the following will have on the RCS: RCP

Proposed Question: Common 35

Given the following conditions:

- A Small Break Loss of Coolant Accident has occurred on Unit 1.
- Reactor Coolant System pressure is 1000 PSIG and steady.
- Reactor Coolant System T_{COLD} is 540ºF and lowering.
- Containment pressure is 12 PSIG.
- Steam Generator pressures are 1000 PSIG.

Which of the following is the reason for stopping the Reactor Coolant Pumps for these plant conditions?

To prevent...

A. core damage resulting from pump operation under forced two-phase flow conditions.

B. core damage resulting from phase separation upon subsequent loss of forced flow.

C. pump damage resulting from a loss of Component Cooling Water to the motor.

D. pump damage resulting from the adverse Containment environment.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because two-phase flow conditions will occur, however, it is the separation of phases when the Reactor Coolant Pumps are secured that is the concern.
B. Correct. Given the conditions listed, this is the reason the Reactor Coolant Pumps are stopped.
C. Incorrect. Plausible if thought that CCW would be lost for the conditions listed.
D. Incorrect. Plausible because an adverse Containment does exist, however, the concern is separation of phases upon subsequent loss of flow.

Technical Reference(s) EOP-1.0A, Attachment 4 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	**DISCUSS** the bases for all steps, Cautions and Notes included in EOP-1.0. (LO21.ERG.E1A.OB105)

Question Source:	Bank #	CPNPP LXR	
	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	10	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	086 K4.02	
Importance Rating	3.0	

Fire Protection System: Knowledge of Fire Protection System design feature(s) and/or interlock(s) which provide for the following: Maintenance of fire header pressure

Proposed Question: Common 36

Given the following conditions:

- A fire has been reported in the Auxiliary Building.
- The Fire Brigade is using Hose Stations to fight the fire.

Which of the following describes the response of the Fire Pumps to decreasing fire header pressure?

A. Both Diesel Fire Pumps start 3 seconds after pressure drops below 142 PSIG, then the Electric Fire Pump starts 10 seconds after pressure drops below 140 PSIG.

B. The Electric Fire Pump starts 3 seconds after pressure drops below 142 PSIG, then one Diesel Fire Pump starts 10 seconds after pressure drops below 140 PSIG.

C. Both Diesel Fire Pumps start after pressure drops below 148 PSIG, then the Electric Fire Pump starts immediately at 142 PSIG.

D. The Electric Fire Pump starts after pressure drops below 148 PSIG, then one Diesel Fire Pump starts immediately at 142 PSIG.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because both Diesel Fire Pumps will start, however, not consistent with the times listed.
B. Correct. Per the guidance provided in SOP-904, this is the response of the Electric and Diesel Fire Pumps.
C. Incorrect. Plausible because both the Electric and Diesel Fire Pumps will start, however, not for the setpoints and sequence listed.
D. Incorrect. Plausible if thought that the Electric Fire Pump will start without delay after pressure drops below setpoint, however, the setpoint is incorrect and the Diesel Fire Pumps do not start until 140 PSIG.

Technical Reference(s) SOP-904, Section 4.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the functions, operation and interlocks of the following Fire Suppression components:

- Electric Fire Pump & Diesel Fire Pumps (OP51.SYS.FP1.OB05)

Question Source: Bank # CPNPP LXR
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 8
55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041 A3.02	
	Importance Rating	3.3	

Steam Dump/Turbine Bypass Control System: Ability to monitor automatic operation of the SDS, including: RCS pressure, RCS temperature, and reactor power

Proposed Question: Common 37

Given the following conditions:

- Unit 2 is operating at 60% power during ascension to 100% power.
- The following parameters are observed:
  - T_{AVE} = 575ºF.
  - Pressurizer Level = 44%.
  - Reactor Coolant System Pressure = 2235 PSIG.
  - 2-PT-505, Turbine Impulse Pressure = 55%.
  - Main Generator load = 700 MWe.
- Subsequently, PT-505 fails to 0% which equates to a T_{REF} of 557ºF.

Given the above conditions, which of the following identifies the status of the Steam Dump Valves and the indication on 2-UI-500, STM DMP DEMAND?

A. All valves CLOSED; 2-UI-500 indicates 100% demand.

B. All valves CLOSED; 2-UI-500 indicates 0% demand.

C. Group 1 valves full OPEN; 2-UI-500 indicates 50% demand.

D. Groups 1 and 2 valves full OPEN; 2-UI-500 indicates 100% demand.

Proposed Answer: A

Explanation:

A. Correct. The Steam Dump Valves remain closed because the arming signal comes from PT-506.
B. Incorrect. Plausible if thought that the reference signal (T_{REF}) lowering armed the dumps; however, PT-506 is the arming signal for C-7.
C. Incorrect. Plausible if thought that both groups only open on a plant trip vice a load rejection, however, no valves open as there is no arming signal.
D. Incorrect. Plausible if thought that the Steam Dumps are armed, the valves would be 100% open.

Technical Reference(s) ABN-709, Section 4.2 Attached w/ Revision # See Comments / Reference
LO21.SYS.SD1.LN, Page 16

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** the affect a loss of the following systems has on the Steam Dump system:

- Tref (Turbine Impulse Pressure) OP51.SYS.SD1.OB15)

Question Source: Bank #
Modified Bank # CPNPP LXR (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 7, 8
55.43

Comments / Reference: From CPNPP LXR Exam Bank	Revision # 01/16/01

The unit is operating at 60% during an ascension to 100% power. The following parameters are observed:

- Tave = 580°F
- PRZR Level = 43%
- RCS Pressure = 2250 psig
- PT-505 = 60%
- MWe = 700 MW

PT-505 fails to 45% which equates to a Tref of 571°F. What is the position of the Steam Dump valves after this failure?

A. Closed, not armed.
B. Closed, but steam dumps are armed.
C. 29% open.
D. 85% open.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068 A4.04	
	Importance Rating	3.8	

Liquid Radwaste System: Ability to manually operate and/or monitor in the control room: Automatic isolation

Proposed Question: Common 38

Which of the following will cause an AUTO closure of X-RV-5253, Liquid Waste Processing System Discharge Isolation Valve while a release is in progress?

A. RM-11 channel not responding to POLL (MAGENTA).

B. Only 2 of 4 Circulating Water Pumps running on associated Unit.

C. RM-11 channel in ALERT alarm (YELLOW).

D. Loss of counts on X-RE-5253, Liquid Effluent Radiation Monitor.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that a monitor not responding to POLL would be INOPERABLE.
B. Incorrect. Plausible because Circulating Water Pumps must be running for the valve to remain open, however, a 2 of 4 coincidence allows release to Unit aligned for discharge.
C. Incorrect. Plausible because radiation level has increased, however, it requires a high radiation level alarm to close X-RV-5253.
D. Correct. A loss of counts on the Liquid Effluent Radiation Monitor will trip X-RV-5253 (OPERATE FAILURE).

Technical Reference(s) OP51.SYS.RM1.LN, Page 39 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the functions, operation and interlocks of the following Liquid Waste Processing System components:

- X-RE-5253 (OP51. SYS.WP1.OB02)

Question Source: Bank # CPNPP LXR

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	11, 13
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	008 AK2.02	
Importance Rating	2.7	

Pressurizer Vapor Space Accident: Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Sensors and detectors

Proposed Question: Common 39

Given the following conditions:

- A Loss of Coolant Accident is in progress.
- Reactor Coolant System pressure is 1580 PSIG and slowly lowering.
- Subcooled Margin is 2ºF superheat and becoming more superheated.
- A cooldown is in progress using the Steam Dump System.
- Containment pressure is 4 PSIG and slowly rising.
- Pressurizer level is 12% and rising.
- All Safety Systems have actuated properly.
- One Control Rod failed to insert on the trip.
- Reactor Vessel Level Indication System Indication: Top 2 lights are DARK and bottom 6 lights are LIT.

Which of the following describes the location of the Reactor Coolant System leak and the status of Reactor Coolant System Inventory?

A. The leak is on the Upper Reactor Head.
Reactor Coolant System Inventory is lowering.

B. The leak is on a Reactor Coolant System Hot Leg.
Reactor Coolant System Inventory is rising.

C. The leak is on the bottom of the Reactor Vessel.
Reactor Coolant System Inventory is rising.

D. The leak is on the steam space of the Pressurizer.
Reactor Coolant System Inventory is lowering.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that the leak on the upper head would cause voiding in the head and lead to increasing level and inventory is decreasing, however, as the inventory leaves the head the Pressurizer level will drop and saturated water in the Pressurizer will flash to steam to try and maintain pressure.

B. Incorrect. Plausible because it could be thought that rising Pressurizer level was indicating that Reactor Coolant Inventory was increasing, however, as the water leaves the Hot Leg it is replaced by a loss of level in the Pressurizer and saturated water in the Pressurizer will flash to steam to try and maintain pressure. Also with pressure and subcooling still dropping then inventory is still lowering.

C. Incorrect. Plausible because it could be thought that the leak on the lower head could lower inventory in the vessel and reach saturation and cause voiding under the head, however, as the inventory leaves the head the Pressurizer level will drop and saturated water in the Pressurizer will flash to steam to try and maintain pressure. Also with pressure and subcooling still dropping then inventory is still decreasing.

D. Correct. A leak in the Pressurizer steam space results in a loss of subcooling in the Pressurizer and when pressure drops to the saturation temperature in the upper head then a void will start forming and Pressurizer level will increase. With subcooling and pressure continuing to lower, inventory is lowering.

Technical Reference(s): LO21.MCO.TAA.LN, Pages 11 & 12; LO21.MCO.TAA.LN, Pages 21 & 22; EOP-1.0A, Attachment 4, Step 6

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the bases for all Steps, Cautions, and Notes included in EOP-1.0. (LO21.ERG.E1A.OB105)

Question Source: Bank # CPNPP LXR

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 3, 5

55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009 G 2.1.27	
	Importance Rating	3.9	

Small Break LOCA: Conduct of Operations: Knowledge of system purpose and/or function

Proposed Question: Common 40

Given the following conditions:

- A Loss of Coolant Accident has occurred.
- Reactor Coolant System pressure is 1300 PSIG and slowly lowering.

Which of the following describes the status of operating Emergency Core Cooling System equipment?

A. Centrifugal Charging Pump Safety Injection flow is stable.
Safety Injection Pump discharge flow is rising.
Safety Injection Accumulator level is stable.
Residual Heat Removal injection flow is rising.

B. Centrifugal Charging Pump Safety Injection flow is rising.
Safety Injection Pump discharge flow is rising.
Safety Injection Accumulator level is stable.
Residual Heat Removal injection flow is zero.

C. Centrifugal Charging Pump Safety Injection flow is stable.
Safety Injection Pump discharge flow is zero.
Safety Injection Accumulator level is lowering.
Residual Heat Removal injection flow is zero.

D. Centrifugal Charging Pump Safety Injection flow is rising.
Safety Injection Pump discharge flow is zero.
Safety Injection Accumulator level is lowering.
Residual Heat Removal injection flow is rising.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because there is CCP SI flow and SI Pump flow is rising, however, RCS pressure is lowering which will cause CCP Pump flow to rise while still above the shutoff head of the RHR Pumps.
B. Correct. Given the pressure of the Reactor Coolant System, this is the operating condition of the ECCS Pumps and Accumulator.
C. Incorrect. Plausible because there is CCP SI flow however, RCS pressure is below the shutoff head of the SI Pumps and above that of the RHR Pumps and the SI Accumulator would not be injecting at this pressure.
D. Incorrect. Plausible because CCP SI flow is rising, however, RCS pressure is below the shutoff head of the SI Pumps and the SI Accumulator and RHR Pumps would not be injecting.

Technical Reference(s) OP51.SYS.SI1.LN, Pages 27, 30, 35 & 41 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the normal capacities, the sources of water, and the approximate flow rates of emergency core cooling components as RCS pressure decreases following a LOCA, include shutoff head values. (OP51.SYS.SI1.OB13)

Question Source: Bank #
Modified Bank # CPNPP LXR (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 7, 14
55.43

Comments / Reference: From CPNPP LXR Exam Bank	Revision # 03/22/07

Given the following conditions:

- A Loss of Coolant Accident has occurred.
- Reactor Coolant System pressure is 1650 PSIG and slowly lowering.

Which of the following describes the status of operating Emergency Core Cooling System equipment?

A. Centrifugal Charging Pump Safety Injection flow is stable.
Safety Injection Pump discharge flow is rising.
Safety Injection Accumulator level is stable.
Residual Heat Removal injection flow is rising.

B. Centrifugal Charging Pump Safety Injection flow is rising.
Safety Injection Pump discharge flow is zero.
Safety Injection Accumulator level is stable.
Residual Heat Removal injection flow is zero.

C. Centrifugal Charging Pump Safety Injection flow is stable.
Safety Injection Pump discharge flow is zero.
Safety Injection Accumulator level is lowering.
Residual Heat Removal injection flow is zero.

D. Centrifugal Charging Pump Safety Injection flow is rising.
Safety Injection Pump discharge flow is rising.
Safety Injection Accumulator level is lowering.
Residual Heat Removal injection flow is rising.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015 AK2.10	
	Importance Rating	2.8	

RCP Malfunctions: Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP indicators and controls

Proposed Question: Common 41

Given the following conditions:

- Unit 1 is at 70% power.
- Annunciator 1-ALB-5B, Window 1.5 - RCP 1 VIBR HI, has just alarmed.
- The following trend of Reactor Coolant Pump (RCP) 1-01 vibration is available:

Time	Shaft	Frame
0130	14.7 mils	3.25 mils
0200	15.0 mils	3.40 mils
0230	15.2 mils	3.52 mils
0300	15.5 mils	3.73 mils
0330	15.9 mils	3.75 mils
0400	17.7 mils	3.85 mils

Which of the following states the earliest time that RCP trip criteria were exceeded?

A. 0200

B. 0230

C. 0300

D. 0330

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because shaft is greater than 15 mils and frame is greater than 3 mils, however, shaft vibration must be increasing at greater than 1 mil/hr and frame vibration must be increasing at greater than 0.2 mils/hr.
B. Correct. Frame vibration was greater than 3 mils/hr and has risen greater than 0.2 mils/hour.
C. Incorrect. Plausible because shaft vibration is greater than 15 mils, however, it must be increasing at greater than 1 mil/hr.
D. Incorrect. Plausible because the shaft and frame rates have been excessive for 1 hour, however, it is not the earliest time.

Technical Reference(s) ABN-101, Step 6.3.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the indications and **DESCRIBE** the mitigation strategy and major steps taken relative to the Reactor Coolant System for:

- ABN-101, Reactor Coolant Pump Trip/Malfunction (OP51.SYS.RC1.OB17)

Question Source: Bank #
Modified Bank # CPNPP LXR (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 3, 10
55.43

Comments / Reference: From CPNPP LXR Exam Bank	Revision # 03/22/07

Given the following conditions:

- Unit 1 is at 85% power.
- Annunciator 1-ALB-5B, Window 1.5 - RCP 1 VIBR HI, has just alarmed.
- The following trend of Reactor Coolant Pump (RCP) 1-01 vibration is available:

Time	Shaft	Frame
0330	14.7 mils	2.65 mils
0400	15.0 mils	2.70 mils
0430	15.2 mils	2.72 mils
0500	15.5 mils	2.73 mils
0530	16.9 mils	3.05 mils
0600	17.7 mils	3.15 mils

Which of the following states the earliest time that RCP trip criteria were exceeded?

A. 0430

B. 0500

C. 0530

D. 0600

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022 AA2.01	
	Importance Rating	3.2	

<u>Loss of Reactor Coolant Makeup</u>: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exists

Proposed Question: Common 42

Given the following conditions:

- Unit 1 is at 100% power and all system alignments are normal.
- The following Annunciators have just gone into alarm:
  - 1-ALB-5C, Window 1.2 - PRZR LVL DEV LO.
  - 1-ALB-6A, Window 1.4 - REGEN HX LTDN OUT TEMP HI.
  - 1-ALB-2A, Window 2.8 - ANY CNTMT SMP PUMP RUN.
- Pressurizer level is 56% and slowly lowering.
- TCV-129, Letdown Demineralizer Bypass Valve, is in the VCT position.
- FCV-121, Charging Pump Control Valve, is full open.
- FI-121A, Charging Flow Meter, indicates 175 GPM and slowly rising.
- All Reactor Coolant Pump seal injection flows are between 8.0 and 8.5 GPM.
- All Reactor Coolant Pump seal return flows are between 2.8 and 3.2 GPM.

Which of the following identifies the location of the leak? (Drawing attached)

The leak is located in the...

A. Charging line between 1/1-8105, Containment Isolation Valve and Regenerative Heat Exchanger.

B. Letdown line between 1/1-8117, Letdown Relief Valve and Regenerative Heat Exchanger.

C. Charging line between 1/1-8106, Charging Line Isolation Valve and 1/1-HCV-182, Charging Line Flow Control Valve.

D. Letdown line between 1/1-8152, Containment Isolation Valve and Letdown Heat Exchanger.

Proposed Answer: A

Explanation:

A. Correct. With Pressurizer level lowering, a Containment Sump Pump running, and a Letdown Heat Exchanger outlet high temperature alarm the leak is located between the Charging Line Isolation Valve and the Regenerative Heat Exchanger.
B. Incorrect. Plausible because a Containment Sump Pump is running and Letdown Heat Exchanger outlet temperature is high which is indicative of a high flow rate, however, if this were the location of the leak Charging flow would exceed Letdown flow and a Pressurizer level deviation HI alarm would be received.
C. Incorrect. Plausible because Pressurizer level is lowering and Letdown Heat Exchanger outlet temperature is rising, however, a Containment Sump Pump is running which is indicative of a leak inside Containment.
D. Incorrect. Plausible because Letdown Heat Exchanger outlet temperature is high, however, if this were the location of the leak Pressurizer level deviation would be high and a Containment Sump Pump would not be running.

Technical Reference(s) ABN-105, Section 2.0 Attached w/ Revision # See Comments / Reference
OP51.SYS.CS1.FG02

Proposed references to be provided during examination: OP51.SYS.CS1.FG02

Learning Objective: **ANALYZE** the indications and **DESCRIBE** the mitigation strategy and major steps taken relative to the Chemical and Volume Control System, both initial and subsequent, for:

- ABN-105, Chemical and Volume Control System Malfunctions (OP51.SYS.CS1.OB20)

Question Source: Bank # CPNPP LXR
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 3, 5, 7
55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	025 G 2.4.9	
Importance Rating	3.8	

Loss of RHR System: Emergency Procedures/Plan: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies

Proposed Question: Common 43

Given the following conditions:

- Unit 1 is operating in Reduced Inventory.
- A large Hot Leg vent path has been aligned due to a Cold Leg opening in the Reactor Coolant System.

Which of the following describes the role of a Hot Leg vent path in mitigating the consequences of a loss of Residual Heat Removal cooling?

A. WITH the Hot Leg vent path, the core heatup and pressurization of the Reactor Coolant will force water out of the Hot Leg and cool the core.

B. WITHOUT the Hot Leg vent path, the core heatup could pressurize the Reactor Coolant causing core uncovery by forcing water out the Cold Leg.

C. WITH the Hot Leg vent path, the time required for the Reactor Coolant to reach saturation conditions in the Reactor Vessel is extended.

D. WITHOUT the Hot Leg vent path, the core heatup could pressurize the Reactor Coolant applying excessive stress at the Reactor Vessel Flange.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because a loss of RHR will cause a heatup, but the RCS should not pressurize with a Hot Leg vent path.
B. Correct. When a Cold Leg opening exists without an adequate Hot Leg vent path, subsequent pressurization of the RCS following a loss of RHR could force water from the core and out the Cold Leg opening. Once water level is below the fuel, the risk of sustaining core damage is substantially increased.
C. Incorrect. Plausible because a major concern when in Reduced Inventory Operations is time to reach saturation conditions, however, providing a Hot Leg vent path will maintain a lower pressure in the RCS and therefore result in a lower saturation temperature and shorter time to core boiling.
D. Incorrect. Plausible because the RCS would pressurize without a Hot Leg vent path, however, stresses on the Vessel Flange due to the pressure are minimal compared to full pressure conditions.

Technical Reference(s): IPO-010A, Step 3.1.4; IPO-010A, Step 5.1.26 Caution; IPO-010A, Attachment 15 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a specific set of system conditions, **PREDICT** system and instrumentation response. (OPD1.IPO.XO0.OB09)

SUMMARIZE the major action steps for each of the sections of IPO-010 considering the following:

- RHR support system requirements (OPD1.IPO.XO0.OB06)

Question Source: Bank # CPNPP LXR
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 5, 10
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	026 AA2.02	
Importance Rating	2.9	

Loss of Component Cooling Water: Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The cause of possible CCW loss

Proposed Question: Common 44

Given the following conditions:

- Component Cooling Water Pumps 2-01 and 2-02 are operating on Unit 2.
- Component Cooling Water Surge Tank Level is lowering with the following Annunciators in alarm:
  - 2-ALB-3B, Window 2.4 - CCW SRG TK TRN A LVL HI-HI/LO.
  - 2-ALB-3B, Window 1.3 - CCW SRG TK TRN A/B LVL LO-LO.
- Component Cooling Water Surge Tank levels are slowly lowering on each compartment.

Which of the following describes the operator actions that would identify the leak source?

CLOSE the...

A. Non-Safeguards Loop Isolation Valves and monitor Tank compartment levels.
Leak is on the side that continues to fall below 37% with the other side stable.

B. Non-Safeguards Loop Isolation Valves and monitor Tank compartment levels.
Leak is on the side that continues to fall below 58% with the other side stable.

C. Safeguards Loop Supply and Return Isolation Valves one Train at a time and monitor Tank compartment levels.
Leak is on the side that continues to fall below 37% with the other side stable.

D. Safeguards Loop Supply and Return Isolation Valves one Train at a time and monitor Tank compartment levels.
Leak is on the side that continues to fall below 58% with the other side stable.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that with both compartments lowering, the source must be the common piping and the methodology would identify a leak on the Non-Safeguards header, however, the isolation of Reactor Coolant Pump cooling should not be performed if other procedurally specified lineups are available.
B. Incorrect. Plausible because it could be thought that with both compartments lowering, the source must be the common piping and the methodology would identify a leak on the Non-Safeguards header, however, the tanks are common until 37% level on Unit 2 which is where the partition plate starts.
C. Correct. The tank is common above 37% on Unit 2 and the leak cannot be identified using this methodology until level reaches 37%. This is the procedurally specified method.
D. Incorrect. Plausible because the tank is common above 58% on Unit 1 and the leak cannot be identified using this methodology until level reaches 58%. This is the procedurally specified method but the wrong Unit.

Technical Reference(s): ABN-502, Steps 3.3.1 Note & 3.3.5; OP5.1.SYS.CC1.LN, Figure 1; ALM-0032B, 2-ALB-3B, Windows 1.3 & 2.4 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the indications and DESCRIBE the mitigation strategy for the following procedures as they affect the Component Cooling Water system:

- ABN-502, Component Cooling Water System Malfunctions (OP51.SYS.CC1.OB21)

DESCRIBE any unit differences between the Unit 1 and Unit 2 Component Cooling Water System or components. (OP51.SYS.CC1.OB15)

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

Question History: Last NRC Exam CPNPP 2010

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

10 CFR Part 55 Content:
55.41 7, 8, 10
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027 AA1.01	
	Importance Rating	4.0	

Pressurizer Pressure Control Malfunction: Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunction: PZR heaters, sprays, and PORVs

Proposed Question: Common 45

Given the following conditions:

- The Unit is at 100% power.
- Pressurizer Pressure is 2235 PSIG and stable.
- 1/1-PS-455F, PRZR PRESS CTRL CHAN SELECT is in the 455/456 position
- PT-455A, Pressurizer Pressure Control Channel fails high.

Assuming NO operator action, which of the following would be the PROGRESSION of events in the primary system (including Pressurizer Pressure Control System response), following the instrument failure?

1. Both PRZR Spray Valves (PCV-455B and PCV-455C) CLOSE.
2. Both PRZR Spray Valves (PCV-455B and PCV-455C) OPEN.
3. PORV PCV-455A OPENS.
4. PORV PCV-456 OPENS.
5. PORVs PCV-455A CLOSES.
6. PORVs PCV-456 CLOSES.
7. All PRZR Heaters energize.
8. All PRZR Heaters deenergize.
9. Reactor Trip on low Pressurizer pressure.
10. Reactor Trip on high Pressurizer pressure.

A. 2, 8, 3, 5, 9

B. 1, 3, 4, 8, 10

C. 2, 7, 4, 6, 9

D. 1, 5, 6, 7, 10

Proposed Answer: A

Explanation:

A. Correct. Given Pressurizer pressure setpoint prior to the Control Channel failing high, the sequence of events is as listed.

B. Incorrect. Plausible because PORV PCV-455A will open and the will deenergize, however, the Spray Valves will open and only PORV RCV-456 would remain closed.

C. Incorrect. Plausible because the Spray Valves will open and the Reactor will trip on low pressure, however, the Pressurizer Heaters will deenergize and PORV PCV-455A is the PORV that will open then close.

D. Incorrect. Plausible the PORV PCV-455A will close and PORV PCV-456 remains closed, however, heaters will deenergize and the Reactor trips on low pressure.

Technical Reference(s): ABN-705, Step 2.2.a; OP51.SYS.PP1, Figure 5 & Slide 58 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **EXPLAIN** the Pressurizer Pressure and Level Control System design features which provide for the trips, permissives and interlocks associated with the following:

- PRZR PORVS Open Interlock in AUTO.
- PRZR High Pressure Reactor Trip.
- PRZR Low Pressure Reactor Trip (OP51.SYS.PP1.OB07)

ANALYZE the indications and **DESCRIBE** the mitigation strategy for the following procedures as they affect the Pressurizer Pressure and Level Control system:

- ABN-705, Pressurizer Pressure Malfunction (OP51.SYS.PP1.OB14)

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 7
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029 EK2.06	
	Importance Rating	2.9	

ATWS: Knowledge of the interrelations between the following and an ATWS: Breakers, relays, and disconnects

Proposed Question: Common 46

Given the following conditions:

- Reactor Trip Breaker testing is in progress on Train "A".
- Reactor Trip Breaker "A" is open.
- Reactor Trip Bypass Breaker "A" is closed.
- A transient occurs initiating an AUTOMATIC Reactor Trip signal.
- The Reactor does NOT trip from the AUTOMATIC signal.

Which of the following describes the condition that has contributed to the Reactor Trip failure?

Reactor Trip...

A. Breaker "B" Undervoltage Trip coil failed to energize.

B. Bypass Breaker "A" Undervoltage Trip coil failed to deenergize.

C. Bypass Breaker "A" Shunt Trip coil failed to energize.

D. Breaker "B" Shunt Trip coil failed to deenergize.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Breaker "B" is equipped with an undervoltage trip coil, however, trip coils are normally energized and deenergize on a trip signal.
B. Correct. Given the conditions listed, the Bypass Breaker Undervoltage Trip coil failed to deenergize.
C. Incorrect. Plausible because the Shunt Trip coil is designed to energize and trip open the breaker, however, Bypass Breaker "A" is not equipped with a Shunt Trip.
D. Incorrect. Plausible because Breaker “B” is equipped with a Shunt Trip coil, however, it energizes to trip.

Technical Reference(s) OP51.SYS.ES1.LN, Page 35 & Figure 1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the performance and design attributes of the following Solid State Protection System components, flow paths, and features:

- Reactor Trip Breakers and Reactor Trip Bypass Breakers (OP51.SYS.ES2.OB02)

DRAW a basic one line diagram of the reactor trip and bypass breaker configuration and **EXPLAIN** the operation of the reactor trip and bypass breakers, the undervoltage trip coil, the shunt trip coil, and the auto shunt trip coil and **DESCRIBE** switchgear response to a manual or automatic reactor trip signal. (OP51.SYS.ES2.OB05)

Question Source: Bank # CPNPP LXR
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 6
55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038 EA1.01	
	Importance Rating	4.5	

Steam Generator Tube Rupture: Ability to operate and/or monitor the following as they apply to a Steam Generator Tube Rupture: SG levels, for abnormal increase in any SG

Proposed Question: Common 47

Given the following condition:

- After entering EOP-3.0A, Steam Generator Tube Rupture, the crew is unable to identify the ruptured Steam Generator (SG) in Step 2, Identify Ruptured SG.

Which of the following guidance is provided for this situation?

A. Close Blowdown Isolation Valves from all Steam Generators.

B. Isolate all Steam Generators prior to identifying the ruptured SG.

C. Do NOT isolate any Steam Generator until ruptured SG identified

D. Close Auxiliary Feedwater Flow Control Valves to all Steam Generators.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this action would be performed if the ruptured Steam Generator was identified at Step 3.
B. Incorrect. Plausible because this action would be performed if the ruptured Steam Generator was identified at Step 3 and its associated MSIV could not be closed per the RNO column.
C. Correct. In this condition, the RNO column directs performance of Steps 7 through 15 in an attempt to identify the ruptured Steam Generator.
D. Incorrect. Plausible if thought that isolating Auxiliary Feedwater flow would help identify the ruptured Steam Generator, however, this action violates minimum feedwater requirements.

Technical Reference(s) EOP-3.0A, Steps 2, 3, & 7 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural step, or sequence of steps from EOP-3.0, **STATE** the purpose/bases for the step(s). (LO21.ERG.E3A.OB103)

Question Source:	Bank #	CPNPP LXR	
	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	10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	040 AK1.06	
	Importance Rating	3.7	

Steam Line Rupture: Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: High energy steam line break considerations

Proposed Question: Common 48

How would the accuracy of the Steam Generator narrow range level indication be affected under the worst-case environmental conditions following a high energy line break inside Containment?

A. Expected to consistently indicate higher than actual.

B. Environmental conditions do not affect indicated level.

C. Expected to consistently indicate lower than actual.

D. Indicated level is erratic, but accuracy is not affected.

Proposed Answer: A

Explanation:

A. Correct. When the reference leg flashes indicated level will be consistently higher than actual level.
B. Incorrect. Plausible if thought that there were no significant effects on Steam Generator level during a high energy line break.
C. Incorrect. Plausible because reference leg flashing doesn't lower level in the reference leg, however, this results in indicated level being higher than actual level.
D. Incorrect. Plausible because the Note prior to Step 1 reminds the operator to use the correct level indication due to temperature induced errors. Additionally, not only does reference leg flashing create an indicated level that is erratic but also impacts Steam Generator level accuracy.

Technical Reference(s) EOP-1.0A, Step 1 Caution Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the bases for all steps, Cautions and Notes included in EOP-1.0. (LO21.ERG.E1A.OB105)

Question Source: Bank # CPNPP LXR
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	5
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	W/E12 EK3.1	
Importance Rating	3.5	

Uncontrolled Depressurization of All Steam Generators: Knowledge of the reasons for the following responses as they apply to the Uncontrolled Depressurization of All Steam Generators: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics

Proposed Question: Common 49

Given the following conditions:

- ECA-2.1A, Uncontrolled Depressurization of All Steam Generators, is in progress on Unit 1.
- Auxiliary Feedwater (AFW) flow is 100 GPM to each Steam Generator (SG).
- Reactor Coolant Pumps have been secured.

Which of the following describes the expected plant response to the AFW flow reduction and what actions are to be taken as SG pressures decrease?

A. RCS Hot Leg temperatures will eventually increase as a result of a loss of Natural Circulation. Raise AFW flow while continuing in ECA-2.1A, Uncontrolled Depressurization of All Steam Generators.

B. RCS Hot Leg temperatures will eventually increase as a result of a loss of Natural Circulation, Raise AFW flow and transition to FRH-0.1A, Response to Loss of Secondary Heat Sink.

C. The SGs will eventually become completely depressurized due to inadequate secondary heat sink. A transition to EOP-2.0A, Faulted Steam Generator Isolation, is required.

D. The SGs will eventually become completely depressurized due to inadequate secondary heat sink. A transition to FRH-0.1A, Response to Loss of Secondary Heat Sink, is required.

Proposed Answer: A

Explanation:

A. Correct. When AFW flow is reduced to 100 GPM, eventually Hot Leg temperatures will rise when SG inventory is depleted. AFW flow is raised to restore Natural Circulation.
B. Incorrect. Plausible because Hot Leg temperatures will begin to increase as SG inventory lowers, however, FRH-0.1A conditions are not met because the AFW flow reduction was controlled by the crew.
C. Incorrect. Plausible because the SGs depressurize as long as they are faulted, however, transition to EOP-2.0A is only performed when one Steam Generator re-pressurizes.
D. Incorrect. Plausible because the SGs depressurize as long as they are faulted, however, ECA-2.1 must be performed to completion unless a SG is isolated or tubes rupture.

Technical Reference(s) ECA-2.1A, Step 2 — Attached w/ Revision # See Comments / Reference
ECA-2.1A, Attachment 4, Step 33 Bases

Proposed references to be provided during examination: None

Learning Objective: Given a major action step of ECA-2.1, **STATE** the basis for all actions taken. (LO21.ERG.C21.OB03)

Question Source: Bank # CPNPP LXR
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 5, 10
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	054 AA2.02	
	Importance Rating	4.1	

Loss of Main Feedwater: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater: Differentiation between loss of all MFW and trip of one MFW pump

Proposed Question: Common 50

Given the following condition:

- Unit 2 is at 100% power.

What effect would actuation of Train A Safety Injection have on the Main Feedwater Pumps?

A. Both Main Feedwater Pumps run until tripped by P-4, Reactor Trip.

B. Both Main Feedwater Pumps trip when the Safety Injection occurs.

C. Main Feedwater Pump 2-01 trips due to the Safety Injection Signal.
Main Feedwater Pump 2-02 is tripped by P-4, Reactor Trip.

D. Main Feedwater Pump 2-01 trips due to the Safety Injection Signal.
Main Feedwater Pump 2-02 continues to run.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because P-4 will generate a Feedwater Isolation Signal if T_{AVE} is below the low T_{AVE} setpoint of 564ºF.
B. Correct. Either Train of Safety Injection will trip both Main Feedwater Pumps.
C. Incorrect. Plausible because the Train A related Main Feedwater Pump trip and the Train B related Main Feedwater Pump trips when the interlocks associated with P-4 actuated.
D. Incorrect. Plausible if thought that Train separation existed to trip each of the Main Feedwater Pumps.

Technical Reference(s) OP51.SYS.MF1.LN, Pages 43 & 46 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **EXPLAIN** the Main Feedwater System design features which provide for the trips, permissives, and interlocks associated with the following:

- Safety Injection (OP51.SYS.MF1.OB11)

Question Source:	Bank #		
	Modified Bank #	CPNPP LXR	(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	4, 7	
	55.43		

Comments / Reference: From CPNPP LXR Exam Bank	Revision # 12/05/96
Which of the following conditions or situations will result in trip of a Main Feedwater Pump? **A. <u>Safety Injection Signal.</u>** B. Hi Main Feedwater Pump vibration. C. 21 inches of vacuum in the Main Condenser. D. LO-LO T_{AVE} with P-4.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055 EK3.02	
	Importance Rating	4.3	

Station Blackout: Knowledge of the reasons for the following responses as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power

Proposed Question: Common 51

Which of the following describes the reasons for depressurizing the Steam Generators to 270 PSIG in accordance with ECA-0.0A, Loss of All AC Power?

A. Initiates Safety Injection System Accumulator discharge and minimize Reactor Coolant Pump seal leakage.

B. Establishes Natural Circulation conditions and initiates Safety Injection System Accumulator discharge.

C. Establishes Natural Circulation conditions and minimizes secondary heat sink requirements if Auxiliary Feedwater inventory is limited.

D. Minimizes secondary heat sink requirements if Auxiliary Feed inventory is limited and minimizes RCP seal leakage.

Proposed Answer: A

Explanation:

A. Correct. Lowering RCS pressure and restoring lost inventory is the reason for depressurizing.
B. Incorrect. Plausible because RCS depressurization will assist Natural Circulation, but is not the reason for depressurization to 270 PSIG.
C. Incorrect. Plausible because Natural Circulation will be established as a byproduct of rapid depressurization. Rapid cooldown and depressurization due to limited AFW is an action that could be taken in E-3 series procedures.
D. Incorrect. Plausible because in E-3 series procedures, rapid secondary depressurizations may be performed when there is limited makeup availability.

Technical Reference(s)	ECA-0.0A, Step 18	Attached w/ Revision # See Comments / Reference
	ECA-0.0A, Attachment 7, Step 18 & Bases	

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the major action taken by the operator during a loss of all AC power to increase the time to core uncovery. (LO21.ERG.C00.OB05)

Given a procedural step, or sequence of steps from ECA-0.1, **STATE** the purpose/basis for the step(s). (LO21.ERG.C00.OB15)

Question Source:	Bank #	CPNPP LXR	
	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	5, 10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	057 AA1.03	
	Importance Rating	3.6	

Loss of Vital AC Instrument Bus: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: Feedwater pump speed to control pressure and level in SG

Proposed Question: Common 52

Given the following conditions:

- Unit 2 is at 80% power with all controls in AUTO.
- Main Feedwater Pump speeds are lowering.
- All Feedwater Flow Control Valves are opening.
- NO operator actions have occurred.

Which of the following has occurred?

A. 2-FT-512A, SG 1 STM FLO failed high.

B. 2-LT-551, SG 1 LVL (NR) CHAN 1 failed high.

C. 2-PT-507, MS HDR PRESS failed low.

D. 2-PT-508, FWP DISCH HDR PRESS failed low.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because if SG 1 STM FLO failed low Main Feedwater Pump speed would lower and the conditions listed would be the same.

B. Incorrect. Plausible because when SG 1 LVL CHAN 1 fails high, the Steam Generator 1-01 Feedwater Flow Control Valve would open, however, for the conditions listed each Steam Generator would have to have a level channel failure. Additionally, opening of the FWCV would cause ΔP to lower which would increase Main Feedwater Pump speed.

C. Correct. When Main Steam Header pressure fails low due to a Loss of Vital Instrument Bus supplied power, the Feedwater System would sense a high ΔP and would lower Main Feedwater Pump speed. Feedwater Flow Control Valves open in an attempt to maintain level consistent with the conditions listed.

D. Incorrect. Plausible because if Feedwater Pump Discharge Header Pressure failed high Main Feedwater Pump speed would lower and the conditions listed would be the same.

Technical Reference(s)
- ABN-603, Attachments 1 & 2
- ABN-709, Steps 3.1 & 3.2
- ABN-709, Steps 5.1 & 5.2
- ABN-707, Steps 2.1 & 2.2
- ABN-710, Steps 2.1 & 2.2

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the impact of the following malfunctions on operation of the Steam Generator Water Level Control System:

- Steam pressure transmitter (OP51.SYS.SN1.OB10)

Question Source:
- Bank # CPNPP LXR
- 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 5, 7
- 55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062 AA2.04	
	Importance Rating	2.5	

Loss of Nuclear Service Water: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The normal values and upper limits for the temperatures of the components cooled by SWS

Proposed Question: Common 53

Given the following conditions:

- Unit 1 is at 100% power.
- Station Service Water (SSW) Pump 1-01 has tripped.
- ABN-501, Station Service Water Malfunction, is in progress.
- SSW Pump 1-02 discharge flow lowers to zero (0) GPM.
- Train A and Train B Component Cooling Water (CCW) Heat Exchanger outlet temperatures are 125ºF and rising.

Which of the following describes the actions required per ABN-501, Station Service Water Malfunction?

Trip the...

A. CCW Pumps only.

B. Reactor and CCW Pumps only.

C. Reactor only.

D. CCW Pumps, Reactor, and Reactor Coolant Pumps.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the CCW Pumps are stopped, however, the Reactor would be tripped and RCPs are secured.
B. Incorrect. Plausible because the Reactor is tripped and the RCPs are stopped, however, the CCW Pumps would also be secured.
C. Incorrect. Plausible because the Reactor is tripped, however, the CCW Pumps and RCPs would also be secured.
D. Correct. When CCW Heat Exchanger outlet temperatures exceed 122ºF, the CCW Pumps are stopped, Reactor tripped, and RCPs secured per the RNO action of ABN-501.

Technical Reference(s): ABN-501, Step 5.3.6 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the reasons for the following responses as they apply to a SSW / CCW malfunction:

- Reactor and/or Turbine trip, manual and automatic (LO21.ABN.501.OB03)

DESCRIBE the function and operation of systems, components, and controls required to be operated in response to a SSW / CCW malfunction. (LO21.ABN.501.OB05)

Question Source:
Bank # CPNPP LXR
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 8, 10
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	065 G 2.4.4	
	Importance Rating	4.5	

Loss of Instrument Air: Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures

Proposed Question: Common 54

Given the following conditions:

- Unit 1 is at 100% power.
- A major break in the Instrument Air System has occurred.
- ABN-301, Instrument Air System Malfunction, is being implemented.
- All available Air Compressors are running.
- Instrument Air header pressure is 33 PSIG.

Which of the following describes required actions for this condition?

A. Continue efforts to restore Instrument Air pressure and simultaneously initiate a Manual Turbine Runback per ABN-302, Feedwater, Condensate, Heater Drain System Malfunction.

B. Trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection. Continue in ABN-301, Instrument Air System Malfunction, and take actions to control Charging flow locally.

C. Align the Unit Instrument Air Cross-tie to assist in pressure recovery. If loss of air operated valve control is observed, trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection.

D. Continue efforts to restore Instrument Air pressure per ABN-301, Instrument Air System Malfunction, and when pressure reaches 25 PSIG, trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because loss of Instrument Air has a significant impact on the secondary plant and it could be thought that rapidly reducing power would minimize the effects, however, ABN-301 requires a Unit trip when pressure reaches 35 PSIG.

B. Correct. ABN-301, Instrument Air Malfunction, requires a Unit trip when pressure reaches 35 PSIG. The EOP is entered but the actions of ABN-301, Instrument Air Malfunction are still performed which includes local control of Charging flow due to failed open valves.

C. Incorrect. Plausible if thought that the only trip criteria was loss of air operated valve control and that would be the proper actions until loss of control was observed, however, ABN-301 requires a Unit trip when pressure reaches 35 PSIG.

D. Incorrect. Plausible if thought that the pressure criteria for tripping was 25 PSIG and the actions to trip at that point would be correct.

Technical Reference(s) ABN-301, Steps 5, 7, & 14 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** all system limitations and precautions associated with responding to Instrument Air System malfunctions. (LO21.ABN.301.OB06)

Question Source: Bank # CPNPP LXR
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 5, 10
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	W/E04 EK1.1	
Importance Rating	3.5	

LOCA Outside Containment: Knowledge of the operational implications of the following concepts as they apply to the LOCA Outside Containment: Components, capacity, and function of emergency systems

Proposed Question: Common 55

Given the following conditions:

- A Loss of Coolant Accident has occurred on Unit 2.
- Actions of ECA-1.2B, LOCA Outside Containment, are being performed.
- The break was not isolated by the actions of ECA-1.2B, and a transition to ECA-1.1B, Loss of Emergency Coolant Recirculation, is required.

Which of the following identifies the effect that a transition to ECA-1.1B, Loss of Emergency Coolant Recirculation, has on mitigating the accident?

Actions are taken to...

A. ensure all Containment Isolation Phase A and Containment Isolation Phase B valves are closed.

B. minimize Refueling Water Storage Tank depletion by reducing injection flow.

C. increase the injection flow rate to restore Reactor Coolant System pressure.

D. stabilize RCS pressure to prevent the Safety Injection Accumulators from dumping out the break.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this action is performed, however, it is done during EOP-1.0B entry.
B. Correct. Reducing injection flow will minimize RWST depletion and pumps would be stopped in the event the RWST is emptied.
C. Incorrect. Plausible if thought that this would help mitigate the problem, however, actions contained in ECA-1.1B are taken to restore Reactor Coolant System mass.
D. Incorrect. Plausible because for certain events this action is performed, however, the ECA-1.1B Flowchart dumps the accumulators if there is no other source of water.

Technical Reference(s) ECA-1.1B, Flowchart; ECA-1.2B, Step 3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, Note or Caution from ECA-1.1, **DISCUSS** the reason or basis for the Step, Note or Caution (LO21.ERG.C11.104)

Give a procedural Step, Note or Caution, from ECA-1.2, **DISCUSS** the reason or bases for the Step, Note, or Caution. (LO21.ERG.C12.OB04)

Question Source: Bank # CPNPP LXR

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 8, 10

55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	077 AK3.02	
Importance Rating	3.6	

<u>Generator Voltage and Electric Grid Disturbances</u>: Knowledge of the reasons for the following responses as they apply to the Generator Voltage and Electric Grid Disturbances: Actions contained in the abnormal operating procedure for voltage and grid disturbances

Proposed Question: Common 56

Given the following conditions:

- Unit 2 Turbine load was at 1265 MWe prior to a degradation of Grid frequency.
- ABN-601, Response to a 138/345 KV System Malfunction, Section 9.0, Grid Frequency Fluctuations/Loss of QSE Generation Controller Communications, is in progress.
- All four High Pressure Control Valves have fully opened.
- Grid frequency has lowered to 57.0 HZ.
- QSE Generation Controller is unable to raise Grid frequency.

Which of the following identifies the action required by ABN-601, Response to a 138/345 KV System Malfunction?

A. Ensure the Reactor is tripped and go to EOP-0.0B, Reactor Trip and Safety Injection.

B. Perform a Normal Start of both Diesel Generators and attempt to stabilize frequency on Safeguards Buses.

C. Take actions to stabilize plant power level for a load reduction.

D. Coordinate with QSE Generation Controller to trip the Reactor.

Proposed Answer: A

Explanation:

A. Correct. Main Turbine load has increased in response to decreasing grid frequency with the Turbine not in LOAD CONTROL. Grid frequency is less than 57.2 Hz which is the setpoint for the RCP under frequency trip.

B. Incorrect. Plausible because this action is correct as Grid Frequency lowers, however, less than 57 Hz requires a Reactor Trip.

C. Incorrect. Plausible because Main Turbine load will change, however, this action is required on a high Grid frequency.

D. Incorrect. Plausible because notification of the QSE Generation Controller is desirable, however, not when a Reactor Trip is required.

Technical Reference(s): ABN-601, Section 9.2; ABN-601, Steps 9.3,.1 & 9.3.3 RNO; ABN-601, Steps 9.3.4 RNO & 9.3.7 RNO — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DIAGNOSE** plant conditions and **IDENTIFY** the indications which are entry-level conditions for ABNs required to be implemented in response to a Station Electrical malfunction. (LO21.ABN.601.OB04)

EVALUATE all system limitations and precautions associated with responding to a Station Electrical malfunction. (LO21.ABN.601.OB12)

Question Source: Bank # CPNPP LXR
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 5, 10
55.43 ______

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	051 AA2.02	
Importance Rating	3.9	

<u>Loss of Condenser Vacuum</u>: Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip

Proposed Question: Common 57

Given the following conditions:

- Main Condenser vacuum has been degrading on Unit 1.
- ABN-304, Main Condenser and Circulating Water System Malfunction, is in progress.
- Condenser vacuum indicates 23.5 inches Hg and is degrading by 0.25 inches Hg per minute.
- The crew is reducing load at approximately 2% per minute.
- Power is currently at 40%.

Assuming the current trends continue, which of the following describes the action that must be taken, and the LATEST time the action must be taken, in accordance with ABN-304, Main Condenser and Circulating Water System Malfunction?

Trip the...

A. Reactor within 10 minutes.

B. Turbine within 10 minutes.

C. Reactor within 15 minutes.

D. Turbine within 15 minutes.

Proposed Answer: A

Explanation:

A. Correct. In 10 minutes vacuum would be < 21 inches Hg, and power is > 10%, therefore, trip the Reactor.
B. Incorrect. Plausible because tripping the Turbine is the correct action if power level is less than 10%, however, in 10 minutes power would still exceed 10%.
C. Incorrect. Plausible because power would be at 10% in 15 minutes but vacuum will be 21” in 10 minutes, requiring a Reactor Trip in 10 minutes.
D. Incorrect. Plausible because power would be at 10% in 15 minutes but vacuum will be 21” in 10 minutes, requiring a Reactor Trip in 10 minutes.

Technical Reference(s) ABN-304, Step 3.3.3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** all system limitations and precautions associated with responding to a Main Condenser, Circ Water, and TPCW malfunction. (LO21.ABN.304.OB12)

Question Source: Bank # CPNPP LXR
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 10
55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	060 AK2.02	
	Importance Rating	2.7	

Accidental Gaseous Radwaste Release: Knowledge of the interrelations between Accidental Gaseous Radwaste Release and the following: Auxiliary building ventilation system

Proposed Question: Common 58

Given the following conditions:

- X-HCV-0014, Waste Gas Discharge Control Valve, was inadvertently opened while transferring Waste Gas Decay Tanks.
- Radiation levels in the Ventilation System are rapidly rising.

Which of the following Radiation Monitors will cause an automatic closure of X-HCV-0014, Waste Gas Discharge Control Valve?

1. X-RE-5250, Waste Gas.
2. X-RE-5567A/B, South/North Vent Stack Discharge Noble Gas.
3. X-RE-5570A/B, South/North Wide Range Gas Monitor Effluent.
4. X-RE-5701, Auxiliary Building Vent Duct.

A. 1 and 2.

B. 1 and 3.

C. 2 and 4.

D. 3 and 4.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that X-RE-5250 Radiation Monitor provides an input to close the Waste Gas Discharge Control Valve, however, it is used to monitor gaseous activity at the Waste Gas Compressor suction. Additionally, noble gases would be released and end in the Vent Stack flow path, however, this function is not provided by X-RE-5567A/B.

B. Incorrect. Plausible because X-RE-5570A/B will close X-HCV-0014, however, X-RE-5250 does not.

C. Incorrect. Plausible because X-RE-5701 will close X-HCV-0014, however, X-RE-5567A/B does not.

D. Correct. Per ABN-902, both of these Radiation Monitors will close X-HCV-0014.

Technical Reference(s)		
	ABN-902, Steps 2.1.b & 2.2	Attached w/ Revision # See Comments / Reference
	OP51.SYS.GH1.LN, Page 33	

Proposed references to be provided during examination: None

Learning Objective: **STATE** the functions, operation and interlocks of the following Gaseous Waste Processing System components:

- X-HCV-0014, GWPS Discharge to Plant Vent Stacks (OP51.SYS.GH1.OB02)

Question Source: Bank # CPNPP LXR
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 11, 14
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	037 AK1.02	
	Importance Rating	3.5	

Steam Generator Tube Leak: Knowledge of the operational implications of the following concepts as they apply to the Steam Generator Tube Leak: Leakrate versus pressure drop

Proposed Question: Common 59

Given the following conditions with Steam Generator Tube Leakage of 200 gallons per day (GPD):

- Reactor Coolant System pressure is 2200 PSIG and lowering.
- Steam Generator pressure is ~900 PSIG.

The Unit is tripped and plant parameters following the trip are:

- Reactor Coolant System pressure is 1700 PSIG and lowering.
- Steam Generator pressure is ~1100 PSIG.

Based on the two sets of data, which of the following describes the effect on primary-to-secondary leakage?

Leakage following the trip is approximately...

A. 50% of the initial leak rate or about 100 GPD.

B. 67% of the initial leak rate or about 133 GPD.

C. 100% of the initial leak rate or about 200 GPD.

D. 150% of the initial leak rate or about 300 GPD.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that half the pressure drop yielded half the flow rate.
B. Correct. Flow is proportional to the square root of the differential pressure; therefore, leakage following the trip is approximately 67% of the initial leak rate.
C. Incorrect. Plausible if thought that leakage was not going to change.
D. Incorrect. Plausible if a math error is made when calculating square root of the differential pressure.

Technical Reference(s) LO21.GFT. FSD, Page 26 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: NRC Generic Fundamentals Equation Sheet

Learning Objective: **SOLVE** problems using the continuity of fluid flow equation. (OP51.GFT.FSD.OB13)

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 14
55.43 ____

Comments / Reference: Calculation	Revision # N/A
200 GPD @ 2200 PSIG – 900 PSIG = ΔP of 1300 PSIG X GPD @ 1700 PSIG – 1100 PSIG = ΔP of 600 PSIG X GPD = 200 GPD/square root of 1300 x square root of 600 X GPD = 200 GPD/36 x 24 = 133.3 GPD @ 1700 PSIG	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076 AK3.05	
	Importance Rating	2.9	

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

Proposed Question: Common 60

Given the following conditions:

- Unit 1 is at 70% power following a Turbine Runback due to Heater Drain Pump 1-01 trip.
- 1-RE-0406, Gross Failed Fuel Monitor (FFL-160), indication has been rising for the past hour, and is now in alarm.

Which of the following describes the action required and reason?

A. Ensure Letdown flow is 120-140 GPM to reduce Reactor Coolant System contamination.

B. Isolate Letdown to minimize radiation levels in the Auxiliary Building.

C. Bypass the mixed bed demineralizer to allow the Letdown filter to remove CRUD.

D. Alternate mixed bed demineralizers to minimize radiation levels in the Auxiliary Building.

Proposed Answer: A

Explanation:

A. Correct. Letdown flow is raised to increase purification of the Reactor Coolant System.
B. Incorrect. Plausible because it is desirable to minimize radiation levels in the Auxiliary Building, however, personnel would be informed and the RCS must be cleaned up.
C. Incorrect. Plausible because CRUD release may have occurred, however, bypassing the mixed bed demineralizer is not an appropriate action.
D. Incorrect. Plausible because it is always desirable to minimize radiation levels, however, when this occurs personnel are notified and entry into the Auxiliary Building would be minimized.

Technical Reference(s) ABN-102, Steps 2.1.b & Section 2.3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the reasons for the following responses as they apply to a high reactor coolant activity and excessive reactor coolant leakage:

- Corrective actions as a result of high fission-product radioactivity level in the RCS. (LO21.ABN.103.OB03)

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 12, 13
55.43 ________

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	W/E08 EA1.3	
Importance Rating	3.6	

RCS Overcooling-PTS: Ability to operate and/or monitor the following as they apply to Pressurized Thermal Shock: Desired operating results during abnormal and emergency situations

Proposed Question: Common 61

FRP-0.1A, Response to Imminent Pressurized Thermal Shock (PTS) Condition directs the operator to check if Emergency Core Cooling System (ECCS) flow can be terminated.

Which of the following conditions is required with respect to this step?

A. ECCS flow should not be terminated until adequate Pressurizer level is established to support restarting a Reactor Coolant Pump.

B. Emergency Core Cooling System flow must be terminated to stabilize Reactor Coolant System Hot Leg temperature.

C. Monitor RVLIS and Core Exit Thermocouples to determine if adequate Reactor Coolant System inventory exists such that core cooling can be verified.

D. Subcooling and Pressurizer level must be closely monitored because Safety Injection termination criteria are more restrictive in this procedure.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because starting an RCP could be a desired action, however, the concerns are adequate inventory and sub cooling.
B. Incorrect. Plausible because flow may need to be terminated, however, the reason is not that listed.
C. Correct. Core inventory and adequate subcooling are requirements that must be met prior to Safety Injection (SI) termination.
D. Incorrect. Plausible because these are the two criteria that must be met for SI flow to be terminated, however, they are <u>less</u> restrictive than other Functional Recovery Procedures.

Technical Reference(s) FRP-0.1A, Attachment 4, Step 7 Bases Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a major action step of FRP-0.2A/B, **STATE** the basis for all actions taken. (LO21.ERG.FP2.OB03)

Question Source:	Bank #	CPNPP LXR	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	CPNPP 2009	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	W/E14 EA2.1	
Importance Rating	3.3	

Loss of Containment Integrity: Ability to determine and interpret the following as they apply to the High Containment Pressure: Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question: Common 62

Given the following conditions:

- Unit 2 entered EOP-0.0B, Reactor Trip or Safety Injection, due to a Reactor Trip and Safety Injection with Containment Spray actuation.
- The actions of EOP-0.0B have been completed, including Attachment 2, Safety Injection Actuation Alignment, and a diagnosis has been made that a Loss of Reactor Coolant inside Containment exists.
- At the time of entry into EOP-1.0B, Loss of Reactor or Secondary Coolant, an ORANGE Path condition is noted on the Containment Status Tree.
- Annunciator 1-ALB-4B, Window 1.8 - RWST 2 of 4 LVL LO-LO alarms and level is confirmed at 33%.

Which of the following is the proper course of actions to implement?

A. Complete Steps 1, 2 and 3 of EOS-1.3B, Transfer to Cold Leg Recirculation, then review Critical Safety Function Status Trees. If ORANGE path on Containment still exists, then transition to FRZ-0.1B, Response to High Containment Pressure.

B. Transition to FRZ-0.1B, Response to High Containment Pressure, and complete the actions without delay so that EOS-1.3B, Transfer to Cold Leg Recirculation, can be implemented to realign Emergency Core Cooling System injection.

C. Complete Steps 1, 2 and 3 of EOS-1.3B, Transfer to Cold Leg Recirculation, and EOP-1.0B, Loss of Reactor or Secondary Coolant. FRZ-0.1B, Response to High Containment Pressure, is not required as EOP-0.0B verified conditions for Containment Spray.

D. Transition to FRZ-0.1B, Response to High Containment Pressure, and concurrently implement actions of EOS-1.3B, Transfer to Cold Leg Recirculation. When completed, return to EOP-1.0B, Loss of Reactor or Secondary Coolant.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, this is the correct set of actions to complete.
B. Incorrect. Plausible because the Critical Safety Function ORANGE Path is immediately addressed, however, entry and completion of EOS-1.3B takes priority.
C. Incorrect. Plausible because completing the actions of EOS-1.3B is correct, however, the Critical Safety Function ORANGE Path cannot be ignored.
D. Incorrect. Plausible because the last two actions listed are correct, however, EOP-1.0B is not implemented at this time.

Technical Reference(s): FRZ-0.1B, CSFST Flowchart
EOP-1.0B, Attachment 1.A, Foldout Page
ODA-407, Attachment 8.A, Step 10
EOS-1.3B, Step 1 CAUTION

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a set of plant conditions, **IDENTIFY** the proper transitions through/out of EOS-1.3. (LO21.ERG.E13.OB06)

Given the Containment CSFST (and necessary parameter data or access to the data), **DETERMINE** whether entry into FRZ-0.1A/B is applicable, and, if so, **STATE** the severity of the challenge. (LO21.ERG.FZ1.OB02)

Question Source: Bank # CPNPP LXR
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 10
55.43 ____

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	W/E02 EK3.3	
Importance Rating	3.9	

SI Termination: Knowledge of the reasons for the following responses as they apply to the SI Termination: Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations

Proposed Question: Common 63

Given the following conditions:

- A Reactor Trip and Safety Injection have occurred as a result of a Faulted Steam Generator outside Containment.
- EOS-1.1A, Safety Injection Termination, is in progress.
  - Centrifugal Charging Pump 1-01 is running and aligned to the normal Charging flow path.
  - Centrifugal Charging Pump 1-02 is OFF and in AUTO.
  - Safety Injection Pumps 1-01 and 1-02 are OFF and in AUTO.
  - Residual Heat Removal Pumps 1-01 and 1-02 are OFF and in AUTO.
- FRI-0.1A, Response to High Pressurizer Level, was started due to a YELLOW path Critical Safety Function Status Tree (CSFST) for Inventory.
- While performing FRI-0.1A, subcooling approaches 20ºF.

Which of the following action is required and reason for the action?

A. Reestablish a Pressurizer bubble per FRI-0.1A to restore subcooling.

B. Close FCV-121, Charging Flow Control Valve to minimize Charging flow.

C. Manually start ECCS Pumps to restore subcooled margin.

D. Manually initiate Safety Injection to immediately start all ECCS Pumps.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this action is performed here, however, subcooling has been lost and EOS-1.1A Foldout Page requires starting ECCS Pumps.
B. Incorrect. Plausible if thought that correcting high Pressurizer level was the desired action.
C. Correct. Per the EOS-1.1A, Attachment 1.A, Foldout Page.
D. Incorrect. Plausible because SI has been reset, however, not per EOS-1.1A Foldout Page guidance.

Technical Reference(s)	EOS-1.1A, Attachment 1.A, Foldout Page ODA-407, Attachment 8.A, Steps 4 & 10	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the items on EOS-1.1 Foldout page including any equipment, parameter, set point or condition. (LO21.ERG.E11.OB107)

Question Source:	Bank #	CPNPP LXR	
	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	10
	55.43	

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	061 G 2.1.28	
Importance Rating	4.1	

ARM System Alarms: Conduct of Operations: Knowledge of the purpose and function of major system components and controls

Proposed Question: Common 64

Which of the following identifies the status of an Area Radiation Monitor channel with a dark blue background on the PC-11, Digital Radiation Monitoring System?

A. PC-11 POLL STATUS.

B. OPERATE FAILURE.

C. Channel ALERT alarm.

D. Channel HIGH alarm.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because there is an alarm associated with PC-11 POLL STATUS, however, it is a white indication.
B. Correct. Designated in the Stem as a dark blue background because an EQUIPMENT FAILURE alarm is a light blue background.
C. Incorrect. Plausible because there is an alarm associated with Channel ALERT, however, it is a yellow indication.
D. Incorrect. Plausible because there is an alarm associated with Channel HIGH, however, it is a red indication.

Technical Reference(s) ALM-3200, Attachment 8 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the location (if applicable) of the following indications and controls, and **DESCRIBE** how each is interpreted or used to predict, monitor, or control changes in the Digital Radiation Monitoring System:

- PC-11 Color/Intensity Status Cues (OP51.SYS.RM1.OB04)

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	11
	55.43	______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	059 AK1.02	
Importance Rating	2.6	

Accidental Liquid Radwaste Release: Knowledge of the operational implications of the following concepts as they apply to the Accidental Liquid Radwaste Release: Biological effects on humans of various types of radiation, exposure levels that are acceptable for nuclear power plant personnel, and the units used for radiation intensity measurements and for radiation exposure levels

Proposed Question: Common 65

Given the following conditions:

- A Waste Holdup Tank has ruptured in the Auxiliary Building.
- The event has critically injured an operator.
- High dose rates exist in the area and the Emergency Coordinator has implemented EPP-305, Emergency Exposure Guidelines and Personnel Dosimetry.

Which of the following describes an eligible volunteer's dose limit for this life-saving rescue?

A volunteer may receive...

A. a maximum of 5 REM TEDE.

B. a maximum of 10 REM TEDE.

C. a maximum of 25 REM TEDE..

D. greater than 25 REM TEDE.

Proposed Answer: D

Explanation:
A. Incorrect. Plausible because this is the limit for all activities.
B. Incorrect. Plausible because this is the limit for protecting valuable property.
C. Incorrect. Plausible because this is the limit for life saving or protecting large populations (non-volunteer).
D. Correct. An eligible volunteer is not limited in dose for a life-saving action.

Technical Reference(s) EPP-305, Step 4.3.2 and Attachment 1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	**STATE** the effect of radiation on cells. (RWT009) **STATE** the Federal Dose Limits for Total Effective Dose Equivalent (TEDE). (RWT017)

Question Source:	Bank #	CPNPP LXR	
	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	12
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	1	
	K/A #	G 2.1.38	
	Importance Rating	3.7	

Conduct of Operations: Knowledge of the station's verbal requirements when implementing procedures

Proposed Question: Common 66

Given the following conditions:

- Unit 2 Reactor tripped from 100% power.
- The crew transitioned to EOS-0.1B, Reactor Trip Response.
- The transition brief was completed.
- The Continuous Action at Step 1 is being directed.

Identify the expected verbal communication based on CPNPP verbal requirements during implementation of Emergency Response Guidelines within the Control Room?

A. US: "Continuous Action, Joe, Check RCS temperature stable at or trending to five five seven degrees."
RO: "Dave, RCS temperature is five six one degrees and slowly lowering."
US: "Joe, RCS temperature is five six one degrees and slowly lowering."
RO: "That is correct."

B. US: "Joe, this next step is a Continuous Action step."
RO: "Dave, the next step is a Continuous Action."
US: "Joe, that is correct, Check RCS temperature stable at or trending to five five seven degrees."
RO: "Dave, RCS temperature is five six one degrees and slowly decreasing."

C. US: "Joe, Check RCS temperature stable at or trending to five five seven degrees."
RO: "RCS temperature is five six one degrees and slowly lowering."
US: "Five sixty one and slowly lowering."
RO: "That's right."

D. US: "Attention in the Control Room. Continuous Action step. End of announcement."
US: "Joe, Check RCS temperature stable at or trending to five five seven degrees."
RO: "I understand that this step is a Continuous Action."
RO: "Dave, RCS temperature is five six one degrees and slowly decreasing."

Proposed Answer: A

Explanation:

A. Correct. Per the guidance identified and the Site Communication Guide, Operations Procedure Guideline, and Operations Department Procedure Use and Adherence.
B. Incorrect. Plausible because the Continuous Action Step must be identified, first names are used, and value with trend is included in the communication, however, decreasing should not be used and acknowledging a Continuous Action Step is not required.
C. Incorrect. Plausible because the communication listed is correct, however, the Reactor Operator failed to state the Unit Supervisor's first name.
D. Incorrect. Plausible because the Continuous Action Step must be identified and value with trend is included in the communication, however, decreasing should not be used, first names are not used consistently, and acknowledging a Continuous Action Step is not required.

Technical Reference(s): NMG-114, Pages 1 & 2; NMG-114, Attachment 1; OPG-3, Section 2.3; ODA-403, Attachment 8.A — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** requirements for Conduct of Operations in accordance with ODA-102, ODA-407 and Operations Guideline 3. (LO21.ADM.XA3.OB01)

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 10
55.43 ___

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	1	
	K/A #	G 2.1.34	
	Importance Rating	2.7	

Conduct of Operations: Knowledge of primary and secondary plant chemistry limits

Proposed Question: Common 67

Which of the following is the setpoint and MODE limitation for Dose Equivalent Iodine-131 (I-131) in the Secondary System?

A. ≤ 0.10 μCi/gm for MODES 1 and 2 only.

B. ≤ 1.00 μCi/gm for MODES 1 and 2 only.

C. ≤ 0.10 μCi/gm for MODES 1, 2, 3, and 4 only.

D. ≤ 1.00 μCi/gm for MODES 1, 2, 3, and 4 only.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the setpoint is correct, however, the MODE limitation is not.
B. Incorrect. Plausible if thought that secondary system dose equivalent I-131 was only applicable in MODES 1 and 2.
C. Correct. Per Technical Specification LCO 3.7.18.
D. Incorrect. Plausible because the MODE limitation is correct, however, the setpoint is not.

Technical Reference(s) Technical Specification LCO 3.7.18 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** indications for system operating parameters, met as a result of high secondary activity, which are entry level conditions for Technical Specifications, **ASSESS** the limiting conditions for operations and safety limits which may be entered, and **APPLY** TS action statements which require response within one hour or less. (LO21.ABN.106.OB07)

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	10
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	1	
	K/A #	G 2.1.25	
	Importance Rating	3.9	

Conduct of Operations: Ability to interpret reference materials, such as graphs curves, tables, etc.

Proposed Question: Common 68

Given the following conditions:

- Unit 1 was operating at 85% power with a Main Steam Safety Valve leaking by on Steam Generator 1-02 when a Steam Generator Tube Rupture occurred.
- Prior to the Reactor Trip and Safety Injection the Main Steam Line Radiation Monitors read as follows;
  - Steam Generator 1-01, 4.56E-02 µci/ml
  - Steam Generator 1-02, 3.78E-02 µci/ml
  - Steam Generator 1-03, 3.45E-02 µci/ml
  - Steam Generator 1-04, 3.05E+02 µci/ml
- The ruptured Steam Generator has been isolated per EOP-3.0A, Steam Generator Tube Rupture.
- The following conditions exist:
  - Containment pressure is 1.3 PSIG.
  - Containment radiation levels are 2.2 REM/hr.
  - Steam Generator 1-01 pressure is 860 PSIG.
  - Steam Generator 1-02 pressure is 790 PSIG.
  - Steam Generator 1-03 pressure is 820 PSIG.
  - Steam Generator 1-04 pressure is 880 PSIG.

Based on the above plant conditions, which of the following is the required Core Exit Temperature for the cooldown? (Table provided.)

A. 425ºF

B. 440ºF

C. 445ºF

D. 460ºF

TABLE 1	
LOWEST -RUPTURED SG PRESSURE (PSIG)	CORE EXIT TEMPERATURE (°F)
1200	495°F (475°F for Adverse Containment)
1150	490°F (470°F for Adverse Containment)
1100	485°F (465°F for Adverse Containment)
1000	475°F (455°F for Adverse Containment)
900	460°F (440°F for Adverse Containment)
800	445°F (425°F for Adverse Containment)
700	430°F (410°F for Adverse Containment)
600	415°F (395°F for Adverse Containment)
500	390°F (375°F for Adverse Containment)
420	370°F (350°F for Adverse Containment)

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that the Adverse Containment number should be used or that the RCS should be cooled to the temperature corresponding to 700 PSIG.
B. Incorrect. Plausible if thought that the Adverse Containment number should be used.
C. Correct. This is the correct temperature for cooling down Steam Generator 1-04 with an initial pressure of 880 PSIG.
D. Incorrect. Plausible because it could be thought that Steam Generator 1-04 was already less than the required pressure.

Technical Reference(s): EOP-3.0A, Step 6.b; EOP-3.0A, Attachment 6, Step 6 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables; EOP-3.0A, Step 6.b, Page 8; EOP-3.0A, Attachment 6, Step 6 Bases

Learning Objective: Given a set of plant conditions, **IDENTIFY** the proper transitions through/out of EOP-3.0. (LO21.ERG.E3A.OB106)

Question Source:
Bank # CPNPP LXR
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 10, 14
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	2	
K/A #	G 2.2.13	
Importance Rating	4.1	

Equipment Control: Knowledge of tagging and clearance procedures

Proposed Question: Common 69

Given the following condition:

- During the implementation of a Clearance that requires a manually operated valve to be danger tagged in the CLOSED position, it is noted that the valve is already danger tagged in the OPEN position on another Clearance.

Which of the following actions is required?

A. CLOSE the valve and attach the Danger Tag and note this in the Comment Section of the Clearance.

B. STOP the Clearance implementation and immediately notify the Unit Supervisor.

C. ATTACH the Danger Tag leaving the valve in the OPEN position and note this in the Comments Section of the Clearance.

D. CONTINUE with the next tag on the Clearance and report the discrepancy to the Control Room after completion.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the valve is positioned per the Clearance, however, the other Danger Tag may still be applicable and could endanger personnel.
B. Correct. As outlined in OWI-110.
C. Incorrect. Plausible if thought that hanging two Danger Tags on the same valve in opposite positions is appropriate.
D. Incorrect. Plausible because the other Danger Tag may not be applicable, however, the Operations Work Control procedure requires that the Unit Supervisor be immediately notified.

Technical Reference(s): OWI-110, Step 6.2.3.1.C — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: At the conclusion of this lesson the Operator will **STATE** the requirements of STA-605, STA-606 and supporting procedures related to the initiation and processing of Work Orders and Clearances. (OPD1.ADM.XAA.OB00)

Question Source:
Bank # CPNPP LXR
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 10
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	2	
	K/A #	G 2.2.35	
	Importance Rating	3.6	

Equipment Control: Ability to determine Technical Specification Mode of Operation

Proposed Question: Common 70

Which of the following conditions defines MODE 3 per Technical Specifications?

A. k_{eff} < 0.99 T_{AVG} < 350ºF

B. k_{eff} ≥ 0.99 T_{AVG} ≥ 350ºF

C. k_{eff} ≥ 0.99 T_{AVG} < 350ºF

D. k_{eff} < 0.99 T_{AVG} ≥ 350ºF

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because k_{eff} is correct, however, temperature is not.
B. Incorrect. Plausible because temperature is correct, however, k_{eff} is not.
C. Incorrect. Plausible because k_{eff} is correct for MODE 2 and temperature is correct for MODE 4.
D. Correct. When k_{eff} is < 0.99 and T_{AVG} is ≥ 350ºF the Unit is in MODE 3, HOT STANDBY.

Technical Reference(s): Tech Spec Definitions, Table 1.1.1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the terms defined in Technical Specifications. (LO21.RLS.SL1.OB01)

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 10
- 55.43

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Category #	3	
K/A #	G 2.3.14	
Importance Rating	3.4	

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

Proposed Question: Common 71

Given the following conditions:

- Unit 2 is in MODE 6 and Containment Fan Coolers are being alternated per SOP-801B, Containment Ventilation System.

Which of the following describes a CAUTION that exists applicable to rotating the Containment Fan Coolers?

A. Always stop the running fan prior to starting the standby fan to protect the discharge ductwork.

B. Always stop the running fan prior to starting the running fan to protect the suction ductwork.

C. Monitor to ensure air flows into Containment from the Equipment Hatch and not from Containment out of the Equipment Hatch.

D. Alternating Fan Coolers can cause changes in indicated radiation levels due to noble gases in stagnant pockets of air.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that restrictions on the number of fans discharging into the ductwork existed, however, there is no CAUTION pertaining to this issue.
B. Incorrect. Plausible because it could be thought that restrictions on the number of fans taking suction from a suction ductwork existed, however, there is no common suction ductwork.
C. Incorrect. Plausible because this is a concern in ventilation changes that change air flow out of or into Containment, however, these fans recirculate air and alternating these fans does not change the air flow through the Equipment Hatch.
D. Correct. The CAUTION is concerned with stagnant pockets of air with noble gases that will be introduced into the Containment atmosphere and the potential to cause automatic Containment Isolation.

Technical Reference(s): SOP-801B, Step 5.1.3 Caution; OP51.SYS.CL1.LN, Page 8 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the basis for the precautions and limitations, and given major procedure steps relative to the Containment Ventilation system, **PLACE** them in the proper sequence for:

- SOP-801, Containment Ventilation System (OP51.SYS.CL1.OB13)

Question Source: Bank # CPNPP LXR
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 9, 12
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	3	
	K/A #	G 2.3.11	
	Importance Rating	3.8	

Radiation Control: Ability to control radiation releases

Proposed Question: Common 72

Given the following conditions with Unit 1 in MODE 6:

- A Containment Purge is in progress.
- Cavity fill per SOP-102A, Residual Heat Removal System Operation, is in progress.
- 1-RE-5502, Containment Particulate Radiation Monitor is in HI alarm.
- 1-RE-5503, Containment Gas Radiation Monitor has a stable trend.
- X-RE-5567A, Plant Vent Gas Monitor shows no increase in activity.
- X-RE-5570A, Wide Range Gas Monitor Low shows no increase in activity.
- 1-RE-6251 and 1-RE-6253, Refueling Cavity Area Radiation Monitors are alarming.

Which of the following describes the equipment actions to take first for these conditions?

A. Manually secure the Containment Purge and stop the Cavity fill.

B. Slow the Cavity fill and place the Containment Pre-Access Filtration Units in service.

C. Verify Containment Ventilation Isolation has occurred and evacuate Containment.

D. Leave Containment Purge in service as long as X-RE-5567A, Plant Vent Gas Monitor is not showing an increasing trend and evacuate Containment.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that Containment Isolation only occurred on the Containment Gas channel because it is the channel required by Technical Specifications and stopping the Containment Purge and cavity fill would be prudent, however, automatic Containment Isolation occurs on either channel and Containment Evacuation is required.

B. Incorrect. Plausible because it could be thought that the cause was the cavity fill and reducing the fill rate would reduce the particulate activity seen by the monitor, however, while reducing or stopping the fill would be prudent and starting the Pre-Access Filtration Units is required later by procedure an automatic Containment Isolation occurs on Containment Particulate High alarm and Containment evacuation is required.

C. Correct. An automatic Containment Isolation occurs on Containment Particulate High alarm and Containment evacuation is required per ABN-902, Release of Radioactive/Toxic Gas.

D. Incorrect. Plausible because if the Containment Radiation Monitors were not in High Alarm then the Purge could stay operating while monitoring the Plant Vent Monitors and Containment evacuation, however, an automatic Containment Isolation occurs on Containment Particulate High alarm and Containment evacuation is required.

Technical Reference(s) ABN-902, Steps 2.3.1 to 2.3.4 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given conditions that warrant a radioactive effluent release, **EVALUATE** and **DETERMINE** the proper methodology for review and authorization of the release permit. (OPD1.ADM.XA8.OB106)

Question Source: Bank # CPNPP LXR
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 11,12
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	4	
	K/A #	G 2.4.19	
	Importance Rating	3.4	

Emergency Procedures / Plan: Knowledge of EOP layout, symbols, and icons

Proposed Question: Common 73

Which of the following identifiers is used to alert the operator of an Emergency Response Guideline (ERG) step that, when performed, could create a radiation hazard?

The procedure step is annotated with a (an)…

A. (*)

B. [R]

C. [CV]

D. (HRA)

Proposed Answer: B

Explanation:

A. Incorrect. This is the designator for a Continuous Action Step.
B. Correct. ERG steps that are performed in an area where radiation hazards are positively identified or which, when performed, create a radiation hazard are annotated as shown.
C. Incorrect. This annotation is used for Concurrent Verification (CV).
D. Incorrect. This annotation is used for a High Radiation Area (HRA), but is not stated in the ERG Rules of Usage. HRA, LHRA, and VHRA are used by Radiation Protection personnel.

Technical Reference(s): STA-660, Steps 4.1, 4.4 & 4.11; ODA-407, Attachment 8.A, Steps 5.A & 5.B; ODA-407, Step 6.2.J — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Upon completion of this lesson, the student should be able to **DESCRIBE** how to use and comply with station procedures. (OPD1.ADM.XA3.OB300/400)

Question Source:
Bank # CPNPP LXR
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	10, 12
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	4	
K/A #	G 2.4.49	
Importance Rating	4.6	

Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls

Proposed Question: Common 74

Given the following:

- A Reactor Trip has occurred on Unit 1.
- All High Pressure Turbine Stop Valves indicate OPEN after attempting a manual Main Turbine Trip.

Which of the following is the NEXT action required?

A. Place all Electro Hydraulic Control Pumps in PULLOUT.

B. Dispatch an operator to locally trip the Main Turbine.

C. Manually runback the Main Turbine using Load Control.

D. Close Main Steam Isolation Valves and Bypass Valves.

Proposed Answer: A

Explanation:

A. Correct. This is the RNO action outlined in EOP-0.0A, Reactor Trip or Safety Injection.
B. Incorrect. Plausible because this action could be performed, however, time does not meet need to limit steam demand.
C. Incorrect. Plausible because the Main Turbine would eventually be taken off-line, however, Runback time does not meet need to limit steam demand.
D. Incorrect. This action is required after the EHC Pumps are stopped and the Main Turbine is not tripped.

Technical Reference(s) EOP-0.0A, Step 2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **PERFORM**, without reference to procedures, the actions that require immediate operation of system components and controls as a result of a Reactor Trip or Safety Injection. (LO21. ERG.E0A.OB06)

Question Source:	Bank #	CPNPP LXR	
	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	10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	4	
	K/A #	G 2.4.5	
	Importance Rating	3.7	

Emergency Procedures / Plan: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question: Common 75

Which of the following procedures are designed to protect specific fission product barriers?

A. Abnormal Conditions Procedures (ABN).

B. Alarm Response Procedures (ALM).

C. Functional Restoration Guidelines (FRG).

D. Optimal Recovery Guidelines (ORG).

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because ABNs address specific radiation and contamination issues, such as, ABN-102, High Reactor Coolant Activity, ABN-106, High Secondary Activity, and ABN-903, Accidental Release of Radioactive Liquid, however, it is the FRG that protects specific fission product barriers.
B. Incorrect. Plausible because numerous ALMs address specific radiation issues, however, it is the FRG that protects specific fission product barriers.
C. Correct. FRGs protect specific fission product barriers via the Critical Safety Function Status Trees.
D. Incorrect. Plausible because ORGs are part of the Emergency Procedures network, however, they act as a conduit for the Frgs.

Technical Reference(s) LO21.ERG.XG1.LN, Pages 14 & 15 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the primary function of the Functional Restoration Guides. (LO21.ERG.XG1.OB07)

Question Source: Bank # CPNPP LXR
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 10
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	058 AA2.03	
Importance Rating		3.9

Loss of DC Power: Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems

Proposed Question: SRO 76

Given the following condition:

- Safeguards DC Bus voltage is at 105 VDC during a Loss of All AC Power.

Which of the following should be performed?

Conduct actions per…

A. ABN-601, Response to a 138/345 KV System Malfunction, because DC power is not required to support AC power restoration.

B. ABN-602, Response to a 6900/480V System Malfunction, because the 480 VAC Battery Charger is an immediate need with voltage at minimum levels on the batteries.

C. ABN-603, Loss of Protection or Instrument Bus, because Safeguards DC bus voltage is insufficient for the current plant conditions.

D. ECA-0.0, Loss of All AC Power, in order to determine the necessity for load shedding of additional DC loads.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because ABN-601 is referenced when power is not restored to one AC Safeguards Bus, however, DC power is required for control power to Vital AC breakers for restoration activities.
B. Incorrect. Plausible because Battery Chargers are required, however, the immediate need is to reduce load on the Battery per ECA-0.0A.
C. Incorrect. Plausible because ABN-603 is referenced in the Load Shedding Attachment of ECA-0.0A, however, 105 VDC is the threshold below which many components fail to work, such as breaker control power.
D. Correct. At 105 VDC the DC Bus is below the threshold at which many components fail to work, actions per ECA-0.0, Loss of All AC Power must be performed.

Technical Reference(s)	ECA-0.0A, Attachment 2 ECA-0.0A, Flow Chart	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a major action Step, Note or Caution of ECA-0.0, **STATE** the basis for the Step, Note or Caution. (LO21.ERG.C00.OB11)

Question Source:	Bank #	CPNPP LXR	
	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

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	056 G 2.2.40	
Importance Rating		4.7

Loss of Offsite Power: Equipment Control: Ability to apply Technical Specifications for a system

Proposed Question: SRO 77

Given the following conditions:

- A Loss of Offsite Power has occurred with Unit 2 in MODE 3.
- All offsite circuits have been declared INOPERABLE.
- Both Emergency Diesel Generators are powering their respective Safeguards Buses.
- No other equipment was previously out of service.

Per Technical Specification LCO 3.8.1, which of the following COMPLETION TIMES would apply to restore one (1) required offsite circuit to OPERABLE status?

Restore one (1) required offsite circuit to OPERABLE status within …

A. 2 hours.

B. 12 hours.

C. 24 hours.

D. 72 hours

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this is the COMPLETION TIME when both Emergency Diesel Generators are INOPERABLE per Technical Specification LCO 3.8.1.E.

B. Incorrect. Plausible because this is the COMPLETION TIME when one required offsite circuit and one Emergency Diesel Generator are INOPERABLE per Technical Specification LCO 3.8.1.D.

C. Correct. When two required offsite circuits are INOPERABLE, restore one required offsite circuit to OPERABLE status within 24 hours per Technical Specification LCO 3.8.1.C.

D. Incorrect. Plausible because this is the COMPLETION TIME when one required offsite circuit is INOPERABLE per Technical Specification LCO 3.8.1.A.

Technical Reference(s): Technical Specification LCO 3.8.1.A & C; Technical Specification LCO 3.8.1.D & E — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **DESCRIBE** the following Technical Specifications (i.e. LCO's, Action Statements and conditional surveillance requirements of one hour or less, if applicable) for the AC Distribution 6.9 Kv and 480 V Systems:

- 3.8.1, AC Sources-Operating (OP51.SYS.AC2.OB16)

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 2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	007 EA2.05	
	Importance Rating		3.9

Reactor Trip - Stabilization - Recovery: Ability to determine and interpret the following as they apply to a reactor trip: Reactor trip first-out indication

Proposed Question: SRO 78

Given the following conditions:

- Unit 1 is operating at 3% power.
- First Out Panel annunciator 1-ALB-6C, Window 2.5 - IR FLUX HI, goes into alarm.

Which of the following describes the immediate response to this alarm?

Enter...

A. ABN-107, Emergency Boration, and immediately commence a boration greater than 30 GPM.

B. ABN-702, Intermediate Range Instrument Malfunction, and verify Reactor power greater than P-6 setpoint.

C. EOP-0.0A, Reactor Trip or Safety Injection, and verify the Reactor is tripped.

D. Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation, and perform actions for a loss of an Intermediate Range Nuclear Instrument.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because one of the probable causes of this alarm is an inadvertent dilution, however, this first-out indication requires an immediate Reactor Trip.
B. Incorrect. Plausible because this is an entry condition into ABN-702, however, receipt of this alarm indicates one channel of Intermediate Range Nuclear Instrument amps is greater than 25%. This action is correct unless the Reactor is tripped.
C. Correct. One channel of Intermediate Range Nuclear Instrument amps is greater than 25%. An immediate Reactor Trip and entry into EOP-0.0A is required.
D. Incorrect. Plausible because Technical Specifications would be entered if one channel has failed, however, the channel indicates high and a Reactor Trip is required.

Technical Reference(s)	ABN-702, Step 2.1.a, 2.2.a, 2.3.1 & 2.3.2	Attached w/ Revision # See Comments / Reference
	ABN-107, Steps 2.1.b	
	Technical Specification LCO 3.3.1	

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** the following as they apply to a Reactor Trip or Safety Injection:

- Reactor trip first-out panel indicator (LO21. ERG.E0A.OB03)

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	W/E11 EA2.2	
	Importance Rating		4.2

Loss of Emergency Coolant Recirculation: Ability to determine and interpret the following as they apply to the Loss of Emergency Coolant Recirculation: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

Proposed Question: SRO 79

Given the following conditions:

- A Loss of Coolant Accident is in progress on Unit 1.
- EOS-1.3A, Transfer to Cold Leg Recirculation was entered.
- Containment Emergency Sump blockage has prevented an Emergency Core Cooling System flowpath from being established.
- A transition to ECA-1.1A, Loss of Emergency Coolant Recirculation is in effect.
- Refueling Water Storage Tank level is 25%.
- Containment pressure has increased to 20 PSIG.

Which of the following procedures would be used, per the ERG Rules of Usage, and what action should be taken in response to the High Containment Pressure?

A. Transition to FRZ-0.1A, Response to High Containment Pressure, and remain in FRZ-0.1A and operate all Containment Spray Pumps until Containment pressure is < 3 PSIG, then stop the Containment Spray Pumps.

B. Transition to FRZ-0.1A, Response to High Containment Pressure, and operate two (2) Containment Spray Pumps to conserve RWST inventory per ECA-1.1A, Loss of Emergency Coolant Recirculation.

C. Remain in ECA-1.1A, Loss of Emergency Coolant Recirculation. Operate all Containment Spray Pumps until Containment pressure is <18 PSIG, then stop and start Containment Spray Pumps as required to maintain pressure less than 18 PSIG.

D. Remain in ECA-1.1A, Loss of Emergency Coolant Recirculation. Do not run the Containment Spray Pumps due to a loss of emergency coolant recirculation capability unless Containment pressure reaches 50 PSIG.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the transition to FRZ-0.1A is correct, however, this procedure sends you to ECA-1.1A when it is determined that a Loss of Emergency Coolant Recirculation has occurred.

B. Correct. Enter FRZ-0.1A and exit at Step 4d when it is determined that ECA-1.1A entry is required. Containment Spray Pumps are operated per ECA-1.1 until sump blockage is overcome.

C. Incorrect. Plausible if thought that remaining in ECA-1.1A is correct, however, only two Containment Spray Pumps would be operated due to emergency sump blockage.

D. Incorrect. Plausible if thought that remaining in ECA-1.1A is correct, however, two Containment Spray Pumps must be operated when Containment pressure is greater than 18 PSIG.

Technical Reference(s): ECA-1.1A, Symptoms or Entry Conditions; ECA-1.1A, Step 10; EOS-1.3A, Step 3; FRZ-0.1A, Step 4.d — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the applicability of ECA-1.1 to include: Modes, RHR/SI Status, and the required action of ECA-1.1 as otherwise used. (LO21.ERG.C11.102)

Question Source: Bank # CPNPP LXR
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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	011 EA2.01	
	Importance Rating		4.7

Large Break LOCA: Ability to determine and interpret the following as they apply to a Large Break LOCA: Actions to be taken, based on RCS temperature and pressure – saturated or superheated

Proposed Question: SRO 80

The Critical Safety Function Status Trees are being reviewed in order to verify the CORE COOLING Critical Safety Function during a Large Break Loss of Coolant Accident. Plant conditions are as follows:

- Containment pressure is 10 PSIG and stable.
- Reactor Coolant System (RCS) Saturation Margin indicates 5ºF SUPERHEAT.
- Reactor Vessel Level Indication System (RVLIS) indication: 11 inches above the Core Plate light is LIT all other RVLIS lights are DARK.
- Core Exit Thermocouples are 500ºF and stable.

Which of the following procedures would be used to mitigate the CORE COOLING Critical Safety Function and what is the basis for that action?

Enter procedure...

A. FRC-0.2, Response to Degraded Core Cooling, due to RCS subcooling less than 25ºF and Core Exit Thermocouples less than 750ºF.

B. FRC-0.3, Response to Saturated Core Cooling, due to RCS subcooling less than 25ºF and Core Exit Thermocouples less than 750ºF.

C. FRC-0.2, Response to Degraded Core Cooling, due to RCS subcooling less than 55ºF and RVLIS indication at 11 inches above the Core Plate LIT.

D. FRC-0.3, Response to Saturated Core Cooling, due to RCS subcooling less than 55ºF and RVLIS indication at 11 inches above the Core Plate LIT.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that both RCS subcooling and Core Exit Thermocouples temperatures were being met.
B. Incorrect. Plausible because procedure entry is correct and Core Exit Thermocouples temperatures are being met, however, RCS subcooling does not meet Adverse Containment conditions.
C. Incorrect. Plausible because RCS subcooling does not meet Adverse Containment conditions, however, with RVLIS indication at 11 inches above the Core Plate LIT and subcooling not met, entry into FRC-0.3A, Response to Saturated Core Cooling is required.
D. Correct. Given that Adverse Containment conditions exist then RCS subcooling must be >55ºF. With CET temperature less than 750ºF and RVLIS indication at 11 inches above the Core Plate LIT, entry into FRC-0.3A, Response to Saturated Core Cooling is correct.

Technical Reference(s) FRC-0.3A, CSFST   Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the indications and **DESCRIBE** the mitigation strategy and major steps taken relative to the Core Cooling Monitor System/RVLIS, both initial and subsequent, for:

- FRC-0.3, Response to Saturated Core Cooling (OP51.SYS.RC3.OB13)

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

Question History: Last NRC Exam CPNPP 2010

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: From CPNPP LXR Exam Bank	Revision # 03/22/10

The Critical Safety Function Status Trees are being reviewed in order to verify the CORE COOLING Critical Safety Function during a Large Break Loss of Coolant Accident. Plant conditions are as follows:

- Containment pressure is 16 PSIG and stable.
- Reactor Coolant System Saturation Margin indicates 300ºF SUPERHEAT.
- RVLIS indication 11 inches above the Core Plate is NOT lit.
- Core Exit Thermocouples are 770ºF and stable.

Which of the following procedures would be used to mitigate the CORE COOLING Critical Safety Function and what is the basis for that action?

Enter procedure...

A. FRC-0.2, Response to Degraded Core Cooling, due to RCS subcooling less than 55ºF and Core Exit Thermocouples greater than 750ºF.

B. FRC-0.3, Response to Saturated Core Cooling, due to RCS subcooling less than 55ºF and Core Exit Thermocouples less than 1200ºF.

C. FRC-0.2, Response to Degraded Core Cooling, due to RCS subcooling less than 25ºF and RVLIS indication at 11 inches above the Core Plate NOT lit.

D. FRC-0.3, Response to Saturated Core Cooling, due to RCS subcooling greater than 25ºF and RVLIS indication at 11 inches above the Core Plate NOT lit.

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #	W/E05 G 2.1.32	
Importance Rating		4.0

Inadequate Heat Transfer - Loss of Secondary Heat Sink: Conduct of Operations: Ability to explain and apply system limits and precautions

Proposed Question: SRO 81

Given the following conditions:

- A Reactor Trip and Safety Injection have occurred on Unit 1.
- EOP-0.0A, Reactor Trip or Safety Injection, actions have been performed.
- Auxiliary Feedwater (AFW) flow cannot be established.
- Containment pressure is 6 PSIG.
- All Steam Generator narrow range levels are off-scale low.
- All Steam Generator wide range levels are at approximately 60%.
- FRH-0.1A, Response to Loss of Secondary Heat Sink, has been entered.
- Reactor Coolant System (RCS) pressure is 1175 PSIG and stable.
- Highest reading Core Exit Thermocouple is 568ºF and stable.
- Intact Steam Generator (SG) pressures are 975 PSIG and slowly trending down.
- Reactor Vessel Level Indication System (RVLIS) indication: All lights are DARK.

Which of the following describes the plant conditions and action required?

Steam Generators are...

A. required for RCS heat removal. Continue in FRH-0.1A, Response to Loss of Secondary Heat Sink, and initiate Bleed and Feed.

B. NOT required for RCS heat removal. Transition to EOP-1.0A, Loss of Reactor or Secondary Coolant, and ensure correct diagnosis of the event prior to continuing.

C. required for RCS heat removal. Continue in FRH-0.1A, Response to Loss of Secondary Heat Sink, and attempt to restore feedwater to the Steam Generators.

D. NOT required for RCS heat removal. Transition to FRC-0.2A, Response to Degraded Core Cooling, due to current RVLIS indication and loss of AFW flow.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Steam Generators are required for RCS heat removal, however, bleed and feed criteria is not met because SG wide range level is not less than 40% (in at least 3 SGs), per FRH-0.1A, Step 3.

B. Incorrect. Plausible because a LOCA is occurring and entry into EOP-1.0A would be made, however, current RCS pressure requires a heat sink.

C. Correct. SGs are required for RCS heat removal because RCS pressure is higher than intact SG pressure. Bleed and Feed criteria is not met because SG wide range level is not less than 40% (in at least 3 SGs), therefore, the procedure should continue with attempts to establish feedwater to any Steam Generator.

D. Incorrect. Plausible because when RVLIS 11 inch core plate light is NOT lit, one of the criteria for FRC-0.2A entry has been met, however, Core Exit Thermocouple temperature is only high enough to warrant FRC-0.3C, Response to Saturated Core Cooling, entry which is a YELLOW Path Critical Safety Function.

Technical Reference(s): FRH-0.1A, Steps 1 & 3; FRC-0.2A, CSFST; EOP-1.0A, Entry Conditions — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the Symptoms or Entry Conditions for FRH-0.1. (LO21.ERG.FH1.OB03)

Question Source: Bank # CPNPP LXR
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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	W/E15 EA2.2	
	Importance Rating		3.3

Containment Flooding: Ability to determine and interpret the following as they apply to the Containment Flooding: Adherence to appropriate procedures and operation within limitations in the facility's license and amendments

Proposed Question: SRO 82

Given the following conditions:

- A Main Steam Line Break with fuel failure has occurred on Unit 2.
- EOP-1.0B, Loss of Reactor or Secondary Coolant, is in progress.
- The initial scan of the Critical Safety Function Status Tree parameters indicate the following:
  - Pressurizer level is 0%.
  - Containment water level indicates 817 feet and slowly rising.
  - Containment Spray has automatically actuated and was verified in EOP-0.0B, Reactor Trip or Safety Injection.
  - Containment pressure is 14 PSIG and slowly lowering.
  - Containment Radiation Monitors are in ALARM at 25 REM/hr.
  - Reactor Vessel Level Indication System has no lights LIT.
  - Core Exit Thermocouples are 292ºF.
  - Loop Cold Leg temperatures are 270ºF.

Which of the following procedures must be entered to address the above conditions?

A. FRZ-0.2B, Response to Containment Flooding.

B. FRP-0.2B, Response to Anticipated Pressurized Thermal Shock Condition.

C. FRI-0.2B, Response to Low Pressurizer Level.

D. FRZ-0.3B, Response to High Containment Radiation Level.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, entry into FRZ-0.2B is correct due to a Containment Flooding condition existing.
B. Incorrect. Plausible because the cooldown rate exceeded 100ºF per hour and Loop Cold Leg temperatures are less than 350ºF, however, it is a Yellow Path condition in the CSFST hierarchy
C. Incorrect. Plausible because Pressurizer level is low and the requirements to enter FRI-0.2B are met, however, it is a Yellow Path condition in the CSFST hierarchy.
D. Incorrect. Plausible because Containment radiation level is high and entry conditions for FRZ-0.3B are met, however, it is a Yellow Path condition in the CSFST hierarchy.

Technical Reference(s): FRZ-0.2B, CSFST; FRI-0.2B, CSFST; FRP-0.2B, CSFST — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given the Containment CSF Status Tree and a description of applicable plant conditions, **DETERMINE** if entry into FRZ-0.2A/B is indicated, and **STATE** the severity of the challenge, if any. (LO21.ERG.FZ2.OB03)

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

Question History: Last NRC Exam CPNPP 2010

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

10 CFR Part 55 Content:
55.41 ____
55.43 5

Comments / Reference: From CPNPP LXR Exam Bank	Revision # 03/22/10

Given the following conditions:

- A Large Break Loss of Coolant Accident has occurred on Unit 2.
- EOP-1.0B, Loss of Reactor or Secondary Coolant, is in progress.
- The initial scan of the Critical Safety Function Status Tree parameters indicate the following:
  - Pressurizer level is 0%.
  - Containment water level indicates 817 feet and slowly rising.
  - Containment Spray has automatically actuated and was verified in EOP-0.0B, Reactor Trip or Safety Injection.
  - Containment pressure is 14 PSIG and slowly lowering.
  - Containment Radiation Monitors are in ALARM at 25 REM/hr.
  - Reactor Vessel Level Indication System has no lights LIT.
  - Core Exit Thermocouples are 292ºF.

Which of the following procedures must be entered to address the above conditions?

A. FRZ-0.2B, Response to Containment Flooding.
B. FRZ-0.1B, Response to High Containment Pressure.
C. FRI-0.2B, Response to Low Pressurizer Level.
D. FRZ-0.3B, Response to High Containment Radiation Level.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	W/E16 G 2.2.42	
Importance Rating		4.6

High Containment Radiation: Equipment Control: Ability to recognize system parameters that are entry level conditions for Technical Specifications

Proposed Question: SRO 83

Given the following conditions:

- Unit 1 is in MODE 3.
- 1-RE-6290A, Containment High Range Radiation Monitor, has failed.

Which of the following is the Technical Specification (TS) impact?

A. Enter a Tracking LCOAR for TS LCO 3.3.3, Post Accident Monitoring (PAM) Instrumentation for 1 of 3 Channels INOPERABLE.

B. Enter a Tracking LCOAR for TS LCO 3.3.4, Remote Shutdown System for 1 of 3 Channels INOPERABLE.

C. Enter an Active LCOAR for TS LCO 3.3.3, Post Accident Monitoring (PAM) Instrumentation for 1 of 2 Required Channels INOPERABLE.

D. Enter an Active LCOAR for TS LCO 3.3.4, Remote Shutdown System for 1 of 2 Required Channels INOPERABLE.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the Technical Specification is correct, however, an Active LCOAR must be initiated and only one of 2 channels is INOPERABLE.
B. Incorrect. Plausible if thought that Containment High Range Radiation was a Technical Specification LCO 3.3.4 entry.
C. Correct. An Active LCOAR must be initiated and only one of 2 channels is INOPERABLE for Technical Specification LCO 3.3.3.
D. Incorrect. Plausible because an Active LCOAR must be initiated and only one of 2 channels is INOPERABLE, however, the Technical Specification entry is incorrect.

Technical Reference(s) Technical Specification Table 3.3.3-1, #10
Technical Specification LCO 3.3.3.A
ODA-308, Steps 6.2.1.A & 6.2.2.A

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **DESCRIBE** the following Technical Specifications (i.e., LCOs, action statements and conditional surveillance requirement of one hour and less, if applicable) for the Reactor Coolant System:

- 3.3.3, Post Accident Monitoring Instrumentation (OP51.SYS.RC1.OB18)

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	W/E06 & E07 G 2.4.20	
	Importance Rating		4.7

Inadequate Core Cooling: Emergency Procedures/Plan: Knowledge of EOP mitigation strategies

Proposed Question: SRO 84

Given the following conditions:

- A Small Break Loss of Coolant Accident has occurred on Unit 2.
- The highest reading Core Exit Thermocouple (CET) temperature is 775ºF and rising.
- Reactor Vessel Level Indication System (RVLIS) Indication: All lights are DARK.
- Reactor Coolant System pressure is 1020 PSIG and lowering.
- Emergency Core Cooling System (ECCS) total injection flow is 0 GPM.
- Efforts to establish ECCS flow have been unsuccessful.
- Reactor Coolant System (RCS) is superheated.
- All Steam Generator (SG) narrow range levels are 0%.
- Total Auxiliary Feedwater flow is 570 GPM.
- Reactor Coolant Pumps 2-01, 2-02 and 2-03 are running.
- Containment pressure is 6 PSIG.
- Critical Safety Function Status Trees are being addressed.

Which of the following should be performed after attempts to establish Emergency Core Cooling System flow are unsuccessful?

A. Open the Pressurizer and Reactor Vessel Head Vents to depressurize the RCS and increase ECCS flow.

B. Transfer to Cold Leg Recirculation to ensure the Reactor Core remains covered.

C. Stop Reactor Coolant Pumps 2-01, 2-02, and 2-03 then initiate Bleed and Feed.

D. Restore intact SG narrow range levels to greater than 18% in at least one SG to provide core cooling.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought this would help cool the core, however, it is imperative to close all RCS and PRZR vents to conserve inventory.
B. Incorrect. Plausible if thought that Transferring to Cold Leg Recirculation would help the condition, however, RWST level is not known and ECCS flow cannot be established.
C. Incorrect. Plausible because RCPs would be stopped later in FRC-0.2A prior to depressurization, however, with Auxiliary Feedwater flow available Core Cooling will be regained by raising Steam Generator narrow range level and steaming that SG.
D. Correct. With Auxiliary Feedwater available and no ECCS flow, FRC-0.2A directs the restoration of at least one Steam Generator narrow range level to greater than 18% (Adverse Containment) to deal with a Degraded Core Cooling condition.

Technical Reference(s) FRC-0.2B, CSFST; FRC-0.2B, Steps 1 to 10 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the indications and **DESCRIBE** the mitigation strategy and major steps taken relative to the Core Cooling Monitor System/RVLIS, both initial and subsequent, for:

- FRC-0.2, Response to Degraded Core Cooling (OP51.SYS.RC3.OB13)

Question Source:
Bank # CPNPP LXR
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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	032 AA2.07	
	Importance Rating		3.4

Loss of Source Range NI: Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Maximum allowable channel disagreement

Proposed Question: SRO 85

Given the following conditions:

- Unit 1 is in a REFUELING outage with core onload in progress.
- 1-NI-50A-2, Gamma-Metrics Source Range Neutron Flux is out-of-service.
- 1-NI-31B, Westinghouse Source Range Neutron Flux, has a lower voltage than 1-NI-32-B, Westinghouse Source Range Neutron Flux.

Which of the following describes the effect on 1-NI-31B due to the lower voltage and the applicable Technical Specification requirement if 1-NI-31B were declared out-of-service?

With a lower voltage the count rate should be...

A. lower. Technical Specifications require two (2) OPERABLE Channels and 1-NI-50B-2 and 1-NI-32B are both available.

B. higher. Technical Specifications require two (2) OPERABLE channels and with only one CORE ALTERATIONS must stop immediately.

C. higher. Technical Specifications require two (2) OPERABLE Channels and 1-NI-50B-2 and 1-NI-32B are both available.

D. lower. Technical Specifications require two (2) OPERABLE Channels and with only one CORE ALTERATIONS must stop immediately.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because when Core offload or onload are not in progress this is the CHANNEL CHECK for the Westinghouse SR Channels, however, Technical Specifications are not being met.
B. Incorrect. Plausible because when Core offload or onload is not in progress this is the CHANNEL CHECK for the Westinghouse SR Channels. The Technical Specification REQUIRED ACTIONS are correct.
C. Incorrect. Plausible because the shiftly CHANNEL CHECK is correct for Core onload, however, Technical Specifications are not being met.
D. Correct. CHANNEL CHECK requirements state that counts should be consistent with Core configuration. Credit can only be taken for matched sets of channels; two Westinghouse or two Gamma-Metrics, therefore, CORE ALTERATIONS must stop immediately.

Technical Reference(s): Technical Specification LCO 3.9.3.A; Technical Specification LCO 3.9.3 Bases; OPT-102A-6, Page 2 of 6

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** the effect a loss of the Excore Instrumentation System has on the following:

- Refueling operations (OP51.SYS.EC1.OB22)

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

Question History: Last NRC Exam CPNPP 2009

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

10 CFR Part 55 Content:
55.41 ______
55.43 2

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	076 G 2.4.45	
Importance Rating		4.3

Service Water System: Emergency Procedures/Plan: Ability to prioritize and interpret the significance of each annunciator or alarm

Proposed Question: SRO 86

Given the following conditions:

- Unit 1 is in MODE 4.
- Train A Component Cooling Water Pump (CCWP) and Train A Station Service Water Pump (SSWP) are in service.
- Train B CCWP and Train B SSWP are secured.
- The following Annunciators are in alarm:
    - 1-ALB-1, Window 1.7 - SSW TRN A/B HDR PRESS LO.
    - 1-ALB-1, Window 1.8 - SSWP 1/2 OVRLOAD/TRIP.
- The BOP determines that SSWP 1-01 has tripped.
- NO other automatic action has occurred.

Which of the following describes the system response and what Technical Specification REQUIRED ACTIONS should be implemented?

A. 1.) Start SSWP 1-02 and CCWP 1-02 because they should have started in response to low SSW header pressure.
   2.) Restore SSW Pump 1-01 to OPERABLE status within 72 hours per LCO 3.7.8.B, Station Service Water System, One SSWS train inoperable.

B. 1.) Start SSWP 1-02 only because it should have started in response to low SSW header pressure.
   2.) Restore SSW Pump 1-01 to OPERABLE status within 7 days per LCO 3.7.8.B, Station Service Water System, One SSWS train inoperable.

C. 1.) Start SSWP 1-02 and CCWP 1-02 because they should have started in response to low SSW header pressure.
   2.) Restore SSW Pump 1-01 to OPERABLE status within 7 days per LCO 3.7.8.B, Station Service Water System, One SSWS train inoperable.

D. 1.) Start SSWP 1-02 only because it should have started in response to low SSW header pressure.
   2.) Restore SSW Pump 1-01 to OPERABLE status within 72 hours per LCO 3.7.8.B, Station Service Water System, One SSWS train inoperable.

Proposed Answer: A

Explanation:

A. Correct. A low SSW header pressure starts the standby SSW and CCW Pumps. Additionally, this is the Technical Specification REQUIRED ACTION.
B. Incorrect. Plausible because the Train B SSW Pump should be started, however, both SSW and CCW Pumps AUTO start on low SSW header pressure and the Technical Specification COMPLETION TIME is 72 hours vice 7 days.
C. Incorrect. Plausible because a low SSW header pressure starts the standby SSW and CCW Pumps, however, the Technical Specification COMPLETION TIME is 72 hours vice 7 days.
D. Incorrect. Plausible because the Technical Specification REQUIRED ACTION is correct and the Service Water Pump will AUTO start on low header pressure, however, both SSW and CCW Pumps should AUTO start on low header pressure.

Technical Reference(s): ALM-0011A, 1-ALB-1, Windows 1.7 & 1.8; ABN-501, Steps 2.3.2, & 2.3.3; Technical Specification LCO 3.7.8.A & B — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the Technical Specifications, Technical Requirements Manual, or ODCM associated with the components, parameters, and operation of the Station Service Water System, including (Tech Specs provided for Action Requirements > 1 hour). (OP51.SYS.SW1.OB13)

Question Source: Bank # CPNPP LXR
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, 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	006 A2.02	
Importance Rating		4.3

Emergency Core Cooling System: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of flowpath

Proposed Question: SRO 87

Given the following conditions:

- A Plant Heatup is in progress on Unit 2 per IPO-001B, Plant Heatup from Cold Shutdown to Hot Standby.
- Current Unit 2 conditions are as follows:
  - Reactor Coolant System temperature is 360ºF.
  - Pressurizer pressure is 450 PSIG.
  - All four Reactor Coolant Pumps are in operation.
- Centrifugal Charging Pump 2-01 tripped on low lube oil pressure.
- Centrifugal Charging Pump 2-02 was started and plant conditions were stabilized.
- The Positive Displacement Charging Pump is tagged out per IPO-001B, Plant Heatup from Cold Shutdown to Hot Standby.

Which of the following completely describes the impact on Technical Specification / Technical Requirement Manual compliance due to this failure and what action is required?

A. Only one of the required Boron Injection flow paths is INOPERABLE. Restore Boration Injection flow path to OPERABLE status within 72 hours.

B. Only one of the required Emergency Core Cooling Trains is INOPERABLE. Restore Emergency Core Cooling System Train to OPERABLE status within 7 days.

C. One Boron Injection flow path and one Emergency Core Cooling Train are INOPERABLE. Restore the Boration flow path and ECCS Train within 7 days.

D. Align the Positive Displacement Charging Pump within 72 hours to restore the Boration Injection flow path to OPERABLE status.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the Boration Injection flow path is INOPERABLE, however, the ECCS Train is also INOPERABLE.

B. Incorrect. Plausible because the ECCS Train is INOPERABLE and the Technical Specification LCO 3.5.2.A REQUIRED ACTION is correct, however, a Boration Injection flow path is also INOPERABLE.

C. Correct. Technical Specification LCO 3.5.2.A and Technical Requirement 13.1.31.B are both impacted. The correct COMPLETION TIME for both systems is 7 days.

D. Incorrect. Plausible because the Positive Displacement Charging Pump (PDP) can be used in lieu of one of the required Centrifugal Charging Pumps, however, an Emergency Core Cooling Train is still INOPERABLE and the PDP can only be substituted when in MODE 4 per the Note in Technical Requirement 13.1.31.

Technical Reference(s): Technical Specification LCO 3.5.2.A; TRM LCO 13.1.31.B — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **DESCRIBE** the following Technical Specifications (i.e. LCOs, action statements and conditional surveillance requirements of one hour and less, if applicable) for the Emergency Core Cooling System:

- ECCS Subsystems 3.5.2 & 3.5.3 (OP51.SYS.SI1.OB30)

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 1, 2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	059 A2.07	
	Importance Rating		3.3

Main Feedwater System: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Tripping of MFW pump turbine

Proposed Question: SRO 88

Given the following conditions:

- Unit 2 is at 100% power.
- Main Feedwater Pump 2A tripped.
- ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, has been entered.
- Steam Generator levels initially stabilized but now are lowering in an uncontrolled manner.
- Main Feedwater Pump 2B speed dropped to 2000 rpm and steam header pressure is 150 PSIG greater than feedwater header pressure.

Which of the following identifies the impact on the Unit and what actions should be taken to mitigate the situation?

A. 1.) A Turbine Runback to 700 MWe will be initiated.
   2.) Initiate an additional manual Turbine Runback of 50 MWe then trip the Main Turbine per ABN-302, Feedwater System Malfunction.

B. 1.) A Reactor Trip and entry into EOP-0.0B, Reactor Trip or Safety Injection is required.
   2.) Start Auxiliary Feedwater (AFW) Pumps and control AFW flow per EOP-0.0B, Reactor Trip or Safety Injection, Attachment 1A, Foldout Page.

C. 1.) A Turbine Runback to 700 MWe will be initiated.
   2.) Trip the Main Turbine and start both Motor Driven AFW Pumps per ABN-302, Feedwater System Malfunction.

D. 1.) A Reactor Trip and entry into EOP-0.0B, Reactor Trip or Safety Injection is required.
   2.) Start all available AFW Pumps at full flow in an attempt to restore Steam Generator level per ABN-302, Feedwater System Malfunction.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because a Turbine Runback is initiated and the 2nd Runback could allow Steam Dumps to control RCS temperature, however, the Reactor must be tripped and Auxiliary Feedwater flow initiated.
B. Correct. Given the conditions listed, a Reactor Trip is required and the Auxiliary Feedwater System is operated as shown.
C. Incorrect. Plausible because a Turbine Runback is initiated and the Motor Driven AFW Pumps will be started, however, a Reactor Trip is required.
D. Incorrect. Plausible because a Reactor Trip is required, however, this is not the appropriate response using the Auxiliary Feedwater System.

Technical Reference(s) ABN-302, Step 2.3.3 and Note; ABN-302, Step 2.3.1 and Note — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DIAGNOSE** plant conditions and **IDENTIFY** the indications which are entry-level conditions for ABNs required to be implemented in response to a Feedwater, Condensate, Heater Drain System Malfunction. (LO21.ABN.302.OB07)

EVALUATE the following as they apply to a Feedwater, Condensate, Heater Drain System malfunction:

- Differentiation between loss of all MFW and trip of one MFW pump (LO21.ABN.302.OB05)

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

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	003 G 2.4.4	
Importance Rating		4.7

Reactor Coolant Pump System: Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters that are entry level conditions for emergency and abnormal procedures

Proposed Question: SRO 89

Given the following conditions:

- Unit 1 is at 50% power.
- Annunciator 1-ALB-5A, Window 1.3 - RC LOOP 1 1 OF 3 FLO LO is in alarm.
- 1-FI-414, RC LOOP 1 FLO CHAN I indication is reading 50%.

Which of the following procedures should be entered for this condition?

A. SOP-108A, Reactor Coolant Pump.

B. ABN-713, RCS Loop Flow Instrument Malfunction.

C. EOP-0.0A, Reactor Trip or Safety Injection.

D. ABN-101, Reactor Coolant Pump Trip/Malfunction.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because if it is thought that at 50% power that 50% flow was expected and the procedure for system operation would be entered.
B. Correct. With only one channel failed, 1-ALB-5A, Window 1.3 is the correct procedure to enter.
C. Incorrect. Plausible because Reactor power is greater than 48% which means that the P-8 permissive is satisfied, however, a 2nd channel would have to be reading less than 90% in order to enter EOP-0.0.
D. Incorrect. Plausible because this annunciator is an entry condition for ABN-101, however, Reactor power would have to be less than 48% in order to meet the criteria for a Reactor Coolant Pump Malfunction as opposed to a Reactor Trip.

Technical Reference(s): ABN-713, Steps 1 & 2; ABN-101, Sections 2.1 and 2.2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the indications and **DESCRIBE** the mitigation strategy and major steps taken relative to the Reactor Coolant System for:

- ABN-101, Reactor Coolant Pump Trip/Malfunction (OP51.SYS.RC1.OB17)

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 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013 G 2.1.30	
	Importance Rating		4.0

Engineered Safety Features Actuation System: Conduct of Operations: Ability to locate and operate components, including local controls

Proposed Question: SRO 90

Given the following conditions:

- Unit 1 is at 100% power.
- Power has been lost to 1PC3, Protection Bus III, and actions are being taken in accordance with ABN-603, Loss of Protection or Instrument Bus.
- Before power is locally transferred to the alternate supply, 1-PT-934, Containment Pressure (IR) Channel IV, fails high.

Which of the following describes the effect on the Safety Injection (SI) and Containment Spray (CS) Systems, and the action required?

A. Safety Injection and Containment Spray both actuate.
Enter EOP-0.0A, Reactor Trip or Safety Injection, and terminate SI and CS per EOS-1.1A, Safety Injection Termination.

B. Safety Injection and Containment Spray both actuate.
Enter EOP-0.0A, Reactor Trip or Safety Injection, and terminate SI per EOS-1.1A, Safety Injection Termination and CS per SOP-204A, Containment Spray Operation.

C. Safety Injection actuates but Containment Spray does NOT actuate.
Enter EOP-0.0A, Reactor Trip or Safety Injection, and terminate SI per EOS-1.1A, Safety Injection Termination.

D. Safety Injection actuates but Containment Spray does NOT actuate.
Enter EOP-0.0A, Reactor Trip or Safety Injection, and initiate SI reduction per FRI-0.1A, Response to High Pressurizer Level.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the procedure entries are correct had both Safety Injection and Containment Spray actuated, however, only SI will actuate.
B. Incorrect. Plausible because the procedure entries are correct for a Safety Injection Actuation, however, Containment Spray does not actuate and it would not be terminated using SOP-204A although it is plausible as it contains a Section for placing Containment Spray in standby.
C. Correct. Given the conditions listed, Safety Injection actuates and Containment Spray does not. Procedure entries are correct as listed.
D. Incorrect. Plausible because Safety Injection actuates and Containment Spray does not, however, only the entry into EOP-0.0A is correct.

Technical Reference(s): EOS-1.1A, Flow Chart; FRI-0.1A, Flow Chart; SOP-204A, Step 5.1.1; OP51.SYS.ES1.LN, Page 66 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DETERMINE** from indications available when an SI should be manually actuated. (OP51.SYS.SI1.OB23)
IDENTIFY the proper transitions through/out of EOS-1.1. (LO21.ERG.E11.OB105)

Question Source: Bank # CPNPP LXR
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

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	001 G 2.2.42	
Importance Rating		4.6

Control Rod Drive System: Equipment Control: Ability to recognize system parameters that are entry level conditions for Technical Specifications

Proposed Question: SRO 91

Given the following conditions:

- Power Range Channel N-41 indicates 81%, N-42 indicates 80%, N-43 indicates 79% and N-44 indicates 81%.
- Annunciator 1-ALB-6D, Window 2.10 - AVE TAVE HI is in alarm.
- Control Bank D Rods stepped in at 72 steps/minute.
- 1/1-RBSS, Control Rod Bank Select Switch was placed in MANUAL to stop rod movement.

Which of the following actions should be taken?

Enter procedure...

A. ABN-704, Tc/N16 Instrumentation Malfunction, and refer to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation.

B. ABN-703, Power Range Instrument Malfunction, and refer to Technical Specification LCO 3.2.4, Quadrant Power Tilt Ratio.

C. ABN-709, Turbine 1st Stage Pressure Instrument Malfunction, and refer to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation.

D. ABN-712, Rod Control System Malfunction, and refer to Technical Specification LCO 3.1.6, Control Bank Insertion Limits.

Proposed Answer: A

Explanation:

A. Correct. ABN-704 entry is required because a temperature instrument has failed. Reference to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation, is required.

B. Incorrect. Plausible because the Technical Specification reference is correct with power greater than 75%, however, the wrong procedure is being referenced due to incorrect diagnosis.

C. Incorrect. Plausible because there are T_{AVE} alarms associated with a Turbine 1st Stage Pressure failing low and rod movement is in the correct direction, however, it is T_{REF}.

D. Incorrect. Plausible if thought that a Rod Control System Malfunction was in progress. This would be a correct analysis if rods were stepping out and Technical Specification LCO 3.1.4 would need to be referenced.

Technical Reference(s): ABN-704, Steps 2.2 & 3.1; ABN-709, Steps 4.1, 4.2, & 4.3.9; ABN-712, Steps 2.3.2, 2.3.2, 2.3.4 & RNOs; ABN-703, Steps 2.2 & 2.3.2

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** interrelationships between the failed instrument in ABN-703 and ABN-704 and other plant systems. (LO21.ABN.703.OB03)

IDENTIFY applicable Technical Specifications and **DETERMINE** actions required to meet the Limiting Condition for Operation (LCO). (LO21.ABN.703.OB05)

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, 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	029 A2.03	
	Importance Rating		3.1

Containment Purge System: Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Startup operations and the associated required valve lineups

Proposed Question: SRO 92

Given the following conditions:

- Unit 1 is in MODE 5.
- A Containment Purge is planned.
- Containment pressure is 1.1 PSIG.

Which of the following identifies the impact on the Containment Purge and what actions should be taken?

A. 1.) Containment Ventilation Isolation signal must be bypassed until the purge is initiated.
   2.) Refer to SOP-706, Digital Radiation Monitoring System, Section for Bypassing Containment Ventilation Isolation.

B. 1.) Outage Auxiliary Containment Cooling Unit Fan should be redirected towards any highly contaminated areas to maximize purge impact.
   2.) Contact Health Physics personnel for guidance on redirecting discharge flow per SOP-801A, Containment Ventilation System.

C. 1.) Containment pressure must be reduced using the Containment Pressure Relief System.
   2.) Refer to SOP-801A, Containment Ventilation System, and lower Containment pressure to atmospheric.

D. 1.) Fuel Transfer Tube Gate Valves must be closed to ensure Spent Fuel Pool level is unaffected by the purge.
   2.) Refer to RF0-404, Refueling Gate Operation, for guidance regarding positioning of the Fuel Transfer Tube Gate Valves.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because starting a Containment Purge can generate spurious Containment Ventilation Isolation signals, however, CVI must remain operational during core alterations or movement of irradiated fuel per Technical Specification LCO 3.9.4.

B. Incorrect. Plausible because health physics personnel should be contacted if this fan is to be used, however, Precautions of SOP-801A do not allow this action.

C. Correct. With Containment pressure greater than atmospheric, pressure must be reduced per SOP-801A.

D. Incorrect. Plausible because a Precaution does exist regarding the differential pressure created and resultant change in Spent Fuel Pool and Refueling Pool levels, however, this valve need not be closed. The Fuel Transfer Tube Gate Valves are closed for Transfer Canal and Refueling Cavity Lift Gate operations.

Technical Reference(s): SOP-801A, Step 5.6.5; SOP-801A, Precautions & Step 4.1.A; SOP-801A, Step 5.1.1.C — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause-effect relationships between the Containment Systems and the following systems, components or events:

- Containment Purge and Pressure Relief (OP51.SYS.CY1.OB22)

LIST and **DESCRIBE** the following Technical Specifications (i.e. LCOs, action statements and conditional surveillance requirements of one hour and less, if applicable) for the Containment Systems.

- Containment Building Penetrations 3.9.4 (OP51.SYS.CY1.OB32)

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 4, 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034 K4.01	
	Importance Rating		3.4

Fuel Handling Equipment System: Knowledge of Fuel Handling Equipment design feature(s) and/or interlock(s) that provide for the following: Fuel protection from binding and dropping

Proposed Question: SRO 93

Given the following conditions:

- Refueling is in progress.
- A fuel assembly is being lowered into the core.
- The hoist load decreased in excess of the Operator Load Limits prior to the fuel assembly reaching the bottom.

Which of the following identifies the action required?

A. Lower the hoist speed to jog and continue to insert the assembly.

B. Laterally position the crane hoist and slowly continue to insert the fuel assembly into the core location.

C. Check the alignment of the lifting tool with respect to the fuel assembly and the reactor core location and reposition the hoist as necessary.

D. If the hoist load decreased by more than 800 pounds, the fuel assembly and adjacent assemblies shall be examined for evidence of damage prior to continuing.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because this condition would have been allowed if hoist speed was in JOG at the time the hoist load decreased, however, it cannot be placed in a lower speed after the fact.
B. Incorrect. Plausible if thought that this would help the positioning of the fuel assembly, however, it does not meet the criteria established by CPNPP.
C. Incorrect. Plausible if thought that this would help the positioning of the fuel assembly, however, it does not meet the criteria established by CPNPP.
D. Correct. Per the guidance outlined in our RFO-302.

Technical Reference(s) RFO-302, Step 6.12 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the axial loading limitation and the location axial loading may be applied to the fuel assembly, as defined in RFO-302. (LO21.RFO.FH2.OB03)

STATE the location that lateral loads may be applied to the fuel assembly as described in RFO-302. (LO21.RFO.FH2.OB02)

STATE the Operator Load Limits, defined in RFO-302, that must be observed when inserting or withdrawing fuel assemblies from any location. (LO21.RFO.FH2.OB07)

Question Source: Bank # CPNPP LXR

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 7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		1
	K/A #	G 2.1.39	
	Importance Rating		4.3

Conduct of Operations: Knowledge of conservative decision making practices

Proposed Question: SRO 94

As a supervisor, which of the following elements must be carefully considered during conservative decision making as it has the greatest potential to increase risk during performance of a particular activity?

New employees...

A. with a “can do” attitude.

B. with a healthy questioning attitude.

C. stopping an activity when unsure of the outcome.

D. applying error prevention techniques during an activity.

Proposed Answer: A

Explanation:

A. Correct. As a supervisor, new employees should be monitored and mentored on conservative decision making but be cautious of a "can-do" attitude.
B. Incorrect. Plausible if thought that a healthy questioning attitude of a new employee was adverse to conservative decision making.
C. Incorrect. Plausible if thought that stopping an activity when unsure of the outcome was adverse to conservative decision making.
D. Incorrect. Plausible if thought that applying error prevention techniques was inadvisable because the employee was new.

Technical Reference(s) Nuclear Policy Statement 208, Pages 2 & 3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **CONDUCT** the required on-shift training and **CONTROL** the actions of trainees in accordance with station procedures. (OPD1.ADM.XA1.OB06)

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	5
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		1
K/A #	G 2.1.35	
Importance Rating		3.9

Conduct of Operations: Knowledge of fuel handling responsibilities of SROs

Proposed Question: SRO 95

Given the following conditions:

- Unit 1 is in MODE 6 with core off-load in progress.
- A Fuel Handling Accident has occurred in Containment.
- Containment Particulate and Gas Radiation Monitors are alarming on PC-11, Digital Radiation Monitoring System.
- ABN-908, Fuel Handling Accident, has been entered.

Which of the following describes responsibilities of the Fuel Handling Supervisor in the Containment Building?

Along with placing the Fuel in a secured position, other responsibilities include:

A.
- Ensure the Upender is horizontal.
- Ensure all Containment penetrations are isolated.
- Ensure the Fuel Transfer Cart is in Containment.

B.
- Ensure the Upender is deenergized.
- Direct personnel to exit Containment via any available means.
- Sound the local Radiological Emergency alarm.

C.
- Ensure all Containment penetrations are isolated.
- Ensure no loads are suspended from the Manipulator Crane.
- Initiate a Containment Purge once Personnel Airlocks are secured.

D.
- Ensure the Upender is horizontal.
- Direct personnel to exit to the Safeguards Building control point.
- Ensure the Fuel Transfer Cart is in the Fuel Building.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the first two actions are correct, however, the Fuel Transfer Cart is moved to the Fuel Building and the Fuel Transfer Tube Gate Valve is closed.
B. Incorrect. Plausible if thought that exiting via the nearest exit would be the safest choice, however, personnel are required to exit into the Safeguards Building control point to allow radiological monitoring.
C. Incorrect. Plausible because the first two actions are correct, however, Containment Purge would be isolated upon receipt of Containment Particulate and Gas High Radiation alarms.
D. Correct. These actions are listed in the responsibilities of the Fuel Handling Supervisor during a Fuel Handling Accident within Containment.

Technical Reference(s) ABN-908, Steps 2.1, 2.2, & 2.3.8 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** those positions which are considered secure for storing a fuel assembly during a fuel handling accident as stated in ABN-908. (LO21.RFO.FH2.OB09)
Given a specific fuel handling accident, be able to **OUTLINE** the Initial Operator Actions per ABN-908. (LO21.RFO.FH2.OB10)

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 4, 7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		2
	K/A #	G 2.2.1	
	Importance Rating		4.4

Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity

Proposed Question: SRO 96

Given the following conditions:

- Unit 2 is in MODE 3.
- Control Rod referencing is in progress per IPO-002B, Plant Startup from Hot Standby.
- Source Range Channel N-31 suddenly fails low.
- Source Range Channel N-32 is indicating 100 cps.
- Control Rod motion was stopped.

Which of the following actions is required?

A. Continue the Reactor Startup until both Intermediate Range Channels reach $1x10^{-10}$ amps then block both Source Range Channels per IPO-002B, Plant Startup from Hot Standby.

B. Restore N-31 to OPERABLE status or ensure all Control Rods inserted and made incapable of withdrawal within 49 hours per ABN-701, Source Range Instrument Malfunction.

C. Block Source Range Channels N-31 and N-32 because they are above the P-6 setpoint and are therefore not required for continued startup per ABN-701, Source Range Instrument Malfunction

D. Immediately ensure both Reactor Trip Breakers are OPEN per ABN-701, Source Range Instrument Malfunction and continue in IPO-002B, Plant Startup from Hot Standby.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because remaining in IPO-002B would be an appropriate action if the initial power level were higher, however, Technical Specification REQUIRED ACTIONS are not being met.
B. Correct. These are the appropriate actions for one Source Range Channel INOPERABLE and Rod Control capable of rod withdrawal.
C. Incorrect. Plausible because the Source Range Channels can eventually be blocked, however, Technical Specification REQUIRED ACTIONS are not being met.
D. Incorrect. Plausible because this would be an appropriate action for two failed Source Range Channels, however, the wrong procedure is referenced.

Technical Reference(s): IPO-002A, Steps 4.1.1, 4.1.2 & 4.1.3
ABN-701, Section 2.3
ABN-701, Attachment 1, Steps 1 & 3
Technical Specification LCOs 3.3.1.I, J, K

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the basis for the precautions and limitations, and given major procedure steps, **PLACE** them in the proper sequence, for:

- IPO-002, Plant Startup from Hot Standby (OP51.SYS.EC1.OB24)

ANALYZE the indications and **DESCRIBE** the mitigation strategy and major steps taken relative to the Excore Instrumentation system, both initial and subsequent, for:

- ABN-701, Source Range Instrumentation Malfunction (OP51.SYS.EC1.OB24)

LIST and **DESCRIBE** the following Technical Specifications (i.e., LCOs, action statements and conditional surveillance requirements of one hour and less, if applicable) for the Excore Instrumentation System:

- 3.3.1, Reactor Trip System Instrumentation (OP51.SYS.EC1.OB26)

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, 5, 6

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		2
K/A #	G 2.2.11	
Importance Rating		3.3

Equipment Control: Knowledge of the process for controlling temporary design changes

Proposed Question: SRO 97

Who shall review Temporary Modifications to ensure compatibility with existing plant conditions, current operating mode and impact on the opposite unit's operation prior to authorizing installation per STA-602, Temporary Modifications and Transient Equipment Placements?

A. Station Operations Review Committee (SORC).

B. Plant Manager.

C. Temporary Modification Coordinator.

D. Shift Manager.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible since SORC must review specific temporary modifications prior to installation per STA-602 Section 5.4.
B. Incorrect. Plausible since Plant Manager approves all Temporary Modifications prior to installation per STA-602 Section 5.5.
C. Incorrect. Plausible since the TMC has multiple responsibilities per STA-602; however, he does not authorize installation per STA-602, Section 5.3.
D. Correct. In accordance with STA-602, Step 6.5.3.

Technical Reference(s): STA-602, Sections 5.3 & 5.4
STA-602, Steps 5.5, 5.6.3, & 6.5.3

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **REVIEW** and **APPROVE** the installation and restoration of Temporary Modifications, and **MAINTAIN** files and documentation of Temporary Modifications. (OPD1.ADM.XA1.OB14)

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

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		3
K/A #	G 2.3.13	
Importance Rating		3.8

Radiation Control: Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: SRO 98

Given the following conditions:

- The Unit is in MODE 1.
- STA-620, Containment Entry, is in progress.
- The gasket on the Inner Air Lock Door was reported damaged.
- There is a gouge across the entire sealing surface approximately 1/8 inch deep and 1/2 inch wide and air flow could be heard through the gouge before pressure was equalized.

Which of the following are the REQUIRED ACTIONS per Technical Specifications?

Verify that...

A. at least the Outer Air Lock Door is CLOSED within 1 hour, and within 24 hours LOCK the Outer Air Lock Door CLOSED. Operation may then continue provided the Outer Air Lock Door is verified LOCKED CLOSED at least once per 31 days.

B. the Outer Air Lock Door is CLOSED within 1 hour. Within 24 hours LOCK the Inner Air Lock Door CLOSED. Be in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

C. the Outer Air Lock Door is CLOSED within 1 hour, and within 24 hours LOCK the Outer Air Lock Door CLOSED. Be in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

D. the Outer Air Lock Door is CLOSED within 1 hour. Within 24 hours LOCK the Inner Air Lock Door is CLOSED. Operation may then continue provided that the Inner Air Lock Door is verified to be LOCKED CLOSED at least once per 31 days.

Proposed Answer: A

Explanation:

A. Correct. Technical Specification LCO 3.6.2.A REQUIRED ACTION is to verify an OPERABLE Air Lock Door is closed within one hour, locked in 24 hours, and surveillance performed every 31 days.

B. Incorrect. Plausible because the Outer Air Lock Door must be closed within 1 hour. The MODE changes are necessary only if the REQUIRED ACTION and COMPLETION TIME of TS LCO 3.6.2.A are not met.

C. Incorrect. Plausible because the initial actions are correct, however, changing MODES is only required if the REQUIRED ACTION and COMPLETION TIME are not met.

D. Incorrect. Plausible because the 1st action is correct, however, REQUIRED ACTIONS and COMPLETION TIMES listed for the Inner Air Lock Door are required for the Outer Air Lock Door.

Technical Reference(s): Technical Specification LCO 3.6.2.A — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **DESCRIBE** the following Technical Specifications (i.e. LCOs, action statements and conditional surveillance requirements of one hour and less, if applicable) for the Containment Systems:

- Containment Air Locks 3.6.2 (OP51.SYS.CY1.OB32)

Question Source:
Bank # CPNPP LXR
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 1, 2

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		4
K/A #	G 2.4.27	
Importance Rating		3.9

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

Proposed Question: SRO 99

Given the following conditions:

- Both Units are at 100% power.
- A fire occurs in the Unit 1 Cable Spreading Room.
- Both Units are required to be shut down.
- The Shift Manager is injured and cannot perform his duties.

Who will perform the duties of the Shift Manager and which Unit will control systems common to both Units per ABN-803A, Response to a Fire in the Control Room or Cable Spreading Room?

A. Unit 1 Unit Supervisor; Unit 1 will control.

B. Unit 1 Unit Supervisor; Unit 2 will control.

C. Unit 2 Unit Supervisor; Unit 1 will control.

D. Unit 2 Unit Supervisor; Unit 2 will control.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Unit 1 will control, however, the Unit 2 Unit Supervisor will act as Shift Manager.
B. Incorrect. Plausible if thought that Unit 1 Unit Supervisor should act as Shift Manager since the fire is in the Unit 1 Cable Spreading Room, however, the opposite Unit's Unit Supervisor performs this function.
C. Correct. Per the guidance set forth in ABN-803A.
D. Incorrect. Plausible because the Unit 2 Unit Supervisor will act as Shift Manager, however, even though the fire is in the Unit 1 Cable Spreading Room, Unit 1 controls common systems.

Technical Reference(s) ABN-803A, Steps 2.3.1 & 2.3.3 Notes Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	**ASSESS** the implementation of ABNs entered in Response To A Fire In The Control Room Or Cable Spreading Room in conjunction with EOPs. (LO21.ABN.803.OB09) **EVALUATE** all system limitations and precautions associated with Response To A Fire In The Control Room Or Cable Spreading Room. (LO21.ABN.803.OB06)

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		4
	K/A #	G 2.4.1	
	Importance Rating		4.8

Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps

Proposed Question: SRO 100

Given the following conditions:

- A Steam Generator Tube Rupture has occurred on Unit 1.
- Due to equipment failures, ECA-3.2A, SGTR With a Loss of Reactor Coolant-Saturated Recovery Desired, is being performed.
- The STA informs you that the Critical Safety Function Status Trees (CSFST) are all GREEN with the exception of the following:
  - CORE COOLING CSFST is YELLOW based on Reactor Vessel Level.
  - INVENTORY CSFST is YELLOW based on Reactor Vessel Level.

Which of the following describes the implementation of procedures for this event?

A. Remain in ECA-3.2A. Do not address either CSFST YELLOW path as implementation is not allowed in the ECA procedures.

B. Hand off both CSFST YELLOW paths to other operators while remaining in ECA-3.2A. The actions of all procedures are to be implemented.

C. Transition from ECA-3.2A and address both CSFST YELLOW paths. The actions for CORE COOLING take precedence over the actions for INVENTORY.

D. Remain in ECA-3.2A. Address INVENTORY actions as desired and do not perform the actions for CORE COOLING due to conflict with ECA-3.2A actions.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because remaining in ECA-3.2A is correct, however, the only limitation imposed on the YELLOW path procedures is for the CORE COOLING Safety Functions.
B. Incorrect. Plausible because remaining in ECA-3.2A is correct, however, the limitation imposed on the YELLOW path procedures is for the CORE COOLING Safety Functions.
C. Incorrect. Plausible because FRC-0.3A is a higher priority, however, it cannot be performed as it conflicts with ECA-3.2A.
D. Correct. Per the guidance contained in FRC-0.3A, Response to Saturated Core Cooling. FRC-0.3A directs a reestablishment of RCS subcooling via Safety Injection Flow. This is inconsistent with the actions of ECA-3.2A which reduce RCS subcooling via ECCS flow reduction in order to minimize primary to secondary leakage during a SGTR.

Technical Reference(s): FRC-0.3A, Step 1 Note; FRC-0.3A, Attachment 2 Bases; ECA-3.2A, Flow Chart; FRI-0.3A, CSFST

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the two paths through which the ERG network may be entered, and **DESCRIBE** the conditions that require/allow entry for each. (LO21.ERG.XD2.OB21)

DESCRIBE the expectations associated with performing immediate actions for the US/RO. (LO21.ERG.XD2.OB29)

Question Source: Bank # CPNPP LXR
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