Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 K4.08	
	Importance Rating	2.7	

Knowledge of AFW design feature(s) and/or interlock(s) which provide for the following: AFW recirculation

RO Question #1

Given the following plant conditions:

- Crew is performing a shutdown, operators have transitioned from Main Feedwater to Auxiliary Feedwater
- Due to high level in 'A' Steam Generator the Control Operator throttled AOV-4007, MDAFW PUMP A DISCHARGE VLV, reducing flow from 150 gpm to 100 gpm
- Auxiliary Feedwater Pump discharge pressure changed from 1100 psig to 1200 psig

At the completion of the throttling process:

1) What is the status of AOV-4304, MDAFW PUMP A RECIRC VLV;

   AND

2) What is the parameter that causes the Recirc Valve to operate?

A. 1) Full Open
   2) discharge flow

B. 1) Full Closed
   2) discharge flow

C. 1) Full Open
   2) discharge pressure

D. 1) Full Closed
   2) discharge pressure

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible if the candidate believes that reducing flow to less than the CLOSE signal of 100 gpm would cause the Recirc valve to open. Also, the second part is correct.

B. CORRECT. Reducing flow to 100 gpm would not cause the Recirc valve to open. The OPEN signal occurs at 80 gpm.

C. INCORRECT. Plausible since other Ginna Recirc valves position is based on the associated pump's discharge pressure.

D. INCORRECT. Plausible since the first part is correct and other Ginna Recirc valves position based on the associated pump's discharge pressure.

Technical Reference(s): 10905-0699 (Rev 2)
(Attach if not previously provided, AR-H-10 (Rev 00701)
including version/revision number) ________

Proposed references to be provided to applicants during examination: None

Learning Objective: R4201C 1.07 (As available)

Question Source: Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New 165603

Question History: Last NRC Exam ________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ________

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 2.4.45	
	Importance Rating	4.1	

Ability to prioritize and interpret the significance of each annunciator or alarm.

RO Question #2

Given the following plant conditions:

- A LOCA has occurred which resulted in significant fuel failure
- Radiation levels throughout the plant are trending up
- Control Room (CR) Radiation Monitor alarms were received as follows:

0800 – R-1, Control Room Area Radiation Monitor, in Warning
0803 – R-45, A Train Control Room Radiation Monitor, in Warning
0806 – R-1, Control Room Area Radiation Monitor, in High Alarm
0809 – R-46, B Train Control Room Radiation Monitor, in Warning
0812 – R-45, A Train Control Room Radiation Monitor, in High Alarm
0815 – R-46, B Train Control Room Radiation Monitor, in High Alarm

Assuming **no operator actions**, what is the FIRST time that BOTH Trains of CREATS system will be in the Emergency mode?

A. 0803
B. 0806
C. 0812
D. 0815

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible if the candidate believes that the monitor warning value provides the signal for CREATS actuation.

B. INCORRECT. Plausible since R-1 provides indication of radiation levels inside the Control Room, but provides no automatic actions.

C. CORRECT. According to P-9, Section 6.2.18 and 6.2.19, "The high alarm setpoint causes the CREATS system to go into the Emergency mode. In accordance with 33013-1353, Sheet 7, the required logic is 1 of 2 radiation monitors (either R-45 or R-46) to start CREATS in Emergency mode.

D. INCORRECT. Plausible since other safeguards actuation signals require a 2 of 2 logic to provide the automatic action to occur

Technical Reference(s): P-9 (p8 and 21; Rev 10000)
(Attach if not previously provided, 33013-1353, Sheet 7 (Rev 4)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2201C 1.07 (As available)

Question Source: Bank # 159170
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 Ginna ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.41	
	Importance Rating	2.8	

Knowledge of the refueling process.

RO Question #3

Given the following plant conditions:

- The plant is in MODE 6
- Movement of irradiated fuel is in progress in Containment and the Auxiliary Building
- All administrative requirements for fuel movement are satisfied

Which ONE of the following conditions, if it were to subsequently occur, would require core alterations to be SUSPENDED?

A. Reactor cavity boron concentration is 2590 ppm

B. 'B' RHR Pump has been determined to be INOPERABLE

C. Refueling cavity level is 23' 8" and lowering by 0.5" per hour

D. Containment Purge has automatically been secured while performing ESFAS testing

Answer: **A**

Explanation (Optional):

A. CORRECT. According to Technical Specification 3.9.1, Boron Concentration "Boron concentrations of the RCS, refueling canal, and the refueling cavity shall be maintained within the limits specified in the COLR (COLR, 2.11 states ≥ 2750 ppm)." LCO Action Statement A "Boron concentration not within limit" has a Required Action of "Suspend CORE ALTERATIONS" with a Completion Time of "Immediately".

B. INCORRECT. Plausible since both RHR loops are required to be OPERABLE if cavity level is < 23 feet according to Technical Specification 3.9.5. Incorrect since cavity level is > 23 feet and slowly lowering, Technical Specification 3.9.4requires one RHR loop to be OPERABLE and in operation.

C. INCORRECT. Plausible Technical Specification 3.9.6 requires cavity water level to be maintained ≥ 23 feet above the top of the reactor vessel flange and requires an immediate suspension of core alterations if water level is < 23 feet. Incorrect since refueling water

cavity level is > 23 feet and slowly lowering, the operating crew has the ability to maintain water level within limits with normal sources of makeup water.

D. INCORRECT. Plausible if the candidate believes that CNMT Purge operation is required for fuel movement and since Technical Specification 3.9.3 requires "Each penetration providing direct access from the CNMT atmosphere to the outside atmosphere shall be either closed or capable of being closed by an OPERABLE CVI System" and the CNMT Purge System is in operation during outage conditions. If these requirements are not met then core alterations are immediately suspended. Incorrect since the condition given is the expected action for an ESFAS actuation regarding CNMT Purge System.

Technical Reference(s): (Attach if not previously provided, including version/revision number)

- Technical Specification 3.9.1 (Rev 122)
- COLR Cycle 39 (p7; Rev 0)
- Technical Specification 3.9.5 (Rev 118)
- Technical Specification 3.9.4 (Rev 118)
- Technical Specification 3.9.6 (Rev 122)
- Technical Specification 3.9.3 (Rev 107)
- Technical Specification Basis 3.9.3 (Rev 53)

Proposed references to be provided to applicants during examination: None

Learning Objective: RRF02C 5.01, 5.02 (As available)

Question Source: Bank # 158412

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 2007 Ginna ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 .10
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 058 AK1.01	
	Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation

RO Question #4

Given the following plant conditions:

- Plant is operating at 100% power
- While monitoring the Vital Battery Monitoring System Panel, the HCO observes:
  - 'A' Battery: +0.5 amps
  - 'B' Battery: -0.8 amps
  - TSC Battery: +1.5 amps

Based on these readings:

1) What is the status of 'B' Battery?
2) How long does the UFSAR credit a fully charged Station Battery being able to supply expected shutdown loads following a station blackout prior to falling below minimum voltage? **(assuming no operator actions taken)**

A. 1) Discharging
   2) 4 hours

B. 1) Discharging
   2) 8 hours

C. 1) Normal Float Charge
   2) 4 hours

D. 1) Normal Float Charge
   2) 8 hours

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with AR-J-31, a negative charging rate is considered discharging as the Vital Battery Monitoring System amp meters read the associated battery's amperage, not the battery charger. UFSAR Section 8.3.2.2 states "Each of the two station batteries is capable of carrying its expected shutdown loads following a plant trip and a loss of all ac power for a period of 4 hours without battery terminal voltage falling below 108.6 V."

B. INCORRECT. Plausible as the first part is correct and the time limitation is that for the EOP network if coping strategies, such as shedding DC loads, are performed.

C. INCORRECT. Plausible if the candidate believes the arrangement of the Vital Battery Monitoring System instruments and believes that the current being measured is that of the battery charger. The second part is correct.

D. INCORRECT. Plausible if the candidate believes the arrangement of the Vital Battery Monitoring System instruments and believes that the current being measured is that of the battery charger and the time limitation is that for the EOP network if coping strategies, such as shedding DC loads, are performed.

Technical Reference(s): UFSAR (p2325; Rev 26)
(Attach if not previously provided, AR-J-31 (Rev 01200)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0901C 1.09 (As available)

Question Source: Bank #
Modified Bank # 158651 (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	.8
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE W/E04 EK1.3	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment): Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Outside Containment)

RO Question #5

Given the following plant conditions:

- An SI due to a LOCA outside of Containment has occurred
- The source of RCS leakage is back-leakage from either V-853A or V-853B, RHR INLET CHECK VALVES TO THE REACTOR VESSEL CORE DELUGE LINE
- The leak location is on the common RHR HX discharge line to the reactor vessel and Loop 'B' cold leg
- The crew has transitioned from E-0, REACTOR TRIP OR SAFETY INJECTION, to the appropriate procedure

Which of the following identifies:

(1) The component that, when closed, could result in leak isolation,

AND

(2) The SI flow change as a result of the successful leak isolation.

NOTE: MOV-720, RHR PUMP DISCHARGE TO LOOP B COLD LEG
MOV-852B, RHR PUMP DISCHARGE TO REACTOR VESSEL DELUGE

A. (1) MOV-720
(2) RISES

B. (1) MOV-720
(2) LOWERS

C. (1) MOV-852B
(2) LOWERS

D. (1) MOV-852B
(2) RISES

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible because MOV-720 is checked in ECA-1.2, but it is a verification of a normally shut valve for a normal alignment. For the given conditions there would be NO change in parameters even if the valve had started from the OPEN position as back-flow leakage would not be isolated. Second part is plausible, if candidate does not know that leak isolation would cause RCS pressure to rise and SI flow to lower, due to SI pumps approaching dead head conditions.

B. INCORRECT. Plausible because MOV-720 is checked in ECA-1.2, but it is a verification of a normally shut valve for a normal alignment. For the given conditions there would be NO change in parameters even if the valve had started from the OPEN position as back-flow leakage would not be isolated. The second part is correct.

C. CORRECT. In accordance with ECA-1.2, MOV-852A and MOV-852B are closed one at a time to attempt leak isolation. Since the given leak location is upstream of failed check valve 853A or 853B, isolation is accomplished once MOV-852A or MOV-852B is closed. ECA-1.2 uses indication of RCS pressure rising for leak isolation which corresponds to SI flow lowering as repressurization occurs.

D. INCORRECT. Plausible as the first part is correct and the candidate may confuse the RCS pressure response with the applicable SI flow response.

Technical Reference(s): ECA-1.2 (Rev 00800)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REC12C 2.01 (As available)

Question Source: Bank # 158655

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments: K/A is matched because the question asks the mitigation actions contained in the EOP and the conditions indicating a LOCA outside Containment.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K6.24	
	Importance Rating	2.5	

Knowledge of the effect of a loss or malfunction on the following CVCS components: Controllers and positioners

RO Question #6

Given the following plant conditions:

- Plant at 100% power
- PT-135, NON-REGEN HX LETDN PRESS, fails low

NOTE: PCV-135, LOW PRESS LTDN PRESS CONTROL VLV
RV-203, LOOP B LETDOWN TO NRHX RELIEF VALVE TO PRT

What effect does the PT-135 failure have on the plant letdown system 5 minutes after the failure **(assume NO operator actions taken)**?

	PCV-135 position	RV-203 position
A.	OPEN	CLOSED
B.	OPEN	OPEN
C.	CLOSED	CLOSED
D.	CLOSED	OPEN

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible if the candidate does not recognize that PT-135 senses pressure upstream of PCV-135 which results in the valve closing to attempt to raise pressure. Also, plausible if the candidate does not recognize that Relief Valve 209 is located downstream of PCV-135 and therefore will not lift.

B. INCORRECT. Plausible if the candidate does not recognize that PT-135 senses pressure upstream of PCV-135 which results in the valve closing to attempt to raise pressure. Also, the second part is correct.

C. INCORRECT. Plausible since the first part is correct and if the candidate does not recognize that Relief Valve 209 is located downstream of PCV-135 and therefore will not lift.

D. CORRECT. PCV-135 controller would get a low input for letdown system pressure and compare that to the setpoint (normally 250 psig). The controller would then send a control signal to CLOSE PCV-135 in an attempt to raise pressure, but would not see a change due to the failed pressure transmitter. Eventually PCV-135 would CLOSE causing Relief Valve 203 to lift at 600 psig to the PRT.

Technical Reference(s): 33013-1264 (Rev 028)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1601C 1.10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 158671

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 K6.01	
	Importance Rating	2.7*	

Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors

RO Question #7

Given the following plant conditions:

- PT-945, CNMT PRESS XMTR, has failed and been defeated using ER-INST.1, REACTOR PROTECTION BISTABLE DEFEAT AFTER INSTRUMENTATION LOOP FAILURE
- Following PT-945 defeat, PT-949, CNMT PRESS XMTR, fails to 60 psig

What is the plant response to the failure of PT-949 with PT-945 defeated?

	Safety Injection	Steam Line Isolation
A.	Will initiate	Will NOT initiate
B.	Will NOT initiate	Will NOT initiate
C.	Will initiate	Will initiate
D.	Will NOT initiate	Will initiate

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with P-1, Containment pressure ≥ 4 psig as read on PT-945, -947, -949, at a coincidence of 2/3, will result in an automatic actuation of Safety Injection. Steam Line Isolation signals are generated from Containment pressure instruments PT-946, -948, and -950; therefore, a Steam Line Isolation signal is NOT generated.

B. INCORRECT. Incorrect since Safety Injection will receive an automatic actuation signal. Plausible since the second part is correct and both Safety Injection and Steam Line Isolation actuation signals occur with a 2/3 coincidence and the candidate may believe that the even numbered CNMT pressure detectors provide the actuation signal for Steam Line Isolation and Safety Injection.

C. INCORRECT. Incorrect since Steam Line Isolation signals for Containment high pressure (18 psig) comes from PT-946, -948, and -950. Plausible since the Safety Injection actuation is correct and both Safety Injection and Steam Line Isolation actuation signals occur with a 2/3 coincidence and the candidate may believe that the odd numbered CNMT pressure detectors provide the actuation signal for Steam Line Isolation and Safety Injection.

D. INCORRECT. Incorrect since Safety Injection will receive an automatic actuation signal. Plausible since both Safety Injection and Steam Line Isolation actuation signals occur with a 2/3 coincidence and the candidate may believe that the odd numbered CNMT pressure detectors provide the actuation signal for Steam Line Isolation and the even numbered CNMT pressure detectors provide the actuation signal for Safety Injection.

Technical Reference(s): P-1 (p16-17, 45-46; Rev 07201)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2701C 1.07, R2401C 1.07, R4001C 1.07 (As available)

Question Source: Bank # 165610
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2009 Kewaunee ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 2.2.38	
	Importance Rating	3.6	

Knowledge of conditions and limitations in the facility license.

RO Question #8

In accordance with plant technical specifications, which one of the following identifies the additional required actions if **ALL THREE** AFW trains and **BOTH** SAFW trains are rendered INOPERABLE with the plant at 100% power?

NOTE: LCO 3.7.5, AUXILIARY FEEDWATER (AFW) SYSTEM

Enter ________(1)__________ and initiate action to restore one AFW or SAFW train to an OPERABLE status _________(2)___________.

A. (1) LCO 3.0.3
   (2) immediately

B. (1) LCO 3.0.3
   (2) within one hour

C. (1) LCO 3.7.5
   (2) immediately

D. (1) LCO 3.7.5
   (2) within one hour

Answer: **C**

Explanation (Optional):

A. INCORRECT. Incorrect because LCO 3.7.3, Condition H contains a NOTE "LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one MDAFW, TDAFW, or SAFW train is restored to OPERABLE status." Plausible because in most cases when all trains of a system are INOPERABLE then entry into LCO 3.0.3 is warranted. The second part is correct.

B. INCORRECT. Incorrect because LCO 3.7.3, Condition H contains a NOTE "LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one MDAFW, TDAFW, or SAFW train is restored to OPERABLE status." Plausible because in

most cases when all trains of a system are INOPERABLE then entry into LCO 3.0.3 is warranted. Also, there are many LCO Required Actions with a Completion Time of one hour.

C. CORRECT. In accordance with Technical Specification LCO 3.7.5, Condition H, with “Three AFW trains and both SAFW trains inoperable” the Required Action is “Initiate action to restore one MDAFW, TDAFW, or SAFW train to OPERABLE status” with a Completion Time of “Immediately”.

D. INCORRECT. Plausible since the first part is correct and there are many LCO Required Actions with a Completion Time of one hour.

Technical Reference(s): Technical Specification LCO 3.7.5 (Rev 88)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4201C 1.12 (As available)

Question Source: Bank # 159168
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.23	
	Importance Rating	3.4	

Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

RO Question #9

Given the following plant conditions:

- A large Break LOCA has occurred
- The crew has transitioned to ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, and are aligning the SI System for cold leg recirculation
- The STA reports Critical Safety Function Status Trees (CSFSTs) are as follows:
  - Containment – YELLOW
  - Core Cooling – ORANGE
  - Heat Sink – RED
  - Integrity – RED
  - Inventory – YELLOW
  - Subcriticality – GREEN

Which ONE of the following identifies the crew response and the basis for this response?

A. Transition to FR-C.2, RESPONSE TO DEGRADED CORE COOLING, to preclude an extreme challenge to the Core Cooling CSFST.

B. Transition to FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, to ensure reactor vessel integrity is maintained.

C. Transition to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, to establish Auxiliary Feedwater as a means of restoring core cooling.

D. No transition required, continue performing ES-1.3 to ensure a source of water is aligned to provide core cooling.

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible because the basis for transitioning to FR-C.2 is correct. Incorrect because ES-1.3 contains a NOTE prior to step 3 that precludes transitioning to FR Procedures until after step 12 is complete. Also, FR-C.2 contains a CAUTION prior to step 13 "The following step will cause SI Accumulator injection which may result in a RED path condition in F-0.4, Integrity Status Tree." This procedure should be completed before transition to FR-P.1, Response to Imminent Pressurized Thermal Shock."

B. INCORRECT. Plausible because a RED condition on the Integrity CSFST would require a transition to FR-P.1; however, ES-1.3 contains a NOTE prior to step 3 that precludes transitioning to FR Procedures until after step 12 is complete.

C. INCORRECT. Plausible because FR-H.1 is the most urgent FR Procedure with the given conditions and would be transitioned to in most cases during the EOPs. Incorrect because ES-1.3 contains a NOTE prior to step 3 that precludes transitioning to FR Procedures until after step 12 is complete.

D. CORRECT. In accordance with ES-1.3, NOTE prior to step 3 "Steps 3 through 12 should be performed without delay. FR procedures should not be implemented prior to completion of these steps." ES-1.3 Background Document states "A suction source of water for the SI pumps must be maintained to provide core cooling. The actions of these initial alignment steps must be completed even if challenges to a CSFST occur at this time, since these steps relate to the maintenance of core cooling."

Technical Reference(s): ES-1.3 (p3-5; Rev 04600)
(Attach if not previously provided, ES-1.3 Background Document (p22; Rev 02601)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RES13C 1.02 (As available)

Question Source: Bank # 158642

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 2012 North Anna ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE 068 2.4.31	
	Importance Rating	4.2	

Knowledge of annunciator alarms, indications, or response procedures.

RO Question #10

The Control Room is being evacuated due to a fire in accordance with AP-CR.1, CONTROL ROOM INACCESSIBILITY.

Control Room actions are complete.

Which ONE of the following describes the next actions taken by the HCO?

Proceed to...

A. the AFW Pump area to transfer equipment to LOCAL control.

B. the Screenhouse to ensure 1 Service Water Pump is running in each loop.

C. the local operating station in the Charging Pump Room for MANUAL control of RCS inventory.

D. the D/G A Room to verify Busses 14 and 18 are energized prior to proceeding to the AFW Pump area to provide assistance.

Answer: A

Explanation (Optional):

A. CORRECT. In accordance with AP-CR.1, step 6 the HCO will locally establish AFW flow to S/Gs.

B. INCORRECT. Plausible because AP-CR.1, step 12 has the US (CRF) check at least one SW Pump running in each loop locally in the Screenhouse.

C. INCORRECT. Plausible because AP-CR.1, step 10 has the CO locally establish Charging flow control.

D. INCORRECT. Plausible because AP-CR.1, step 5 has the STA locally verify Emergency AC Busses 14 and 18 are energized in A D/G Room. The HCO will respond to the AFW Pump area.

Technical Reference(s): AP-CR.1 (Rev 02403)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP04C 2.01 (As available)

Question Source: Bank # 158665
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 Ginna ILT SRO Retake
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.5	
	Importance Rating	2.9*	

Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

RO Question #11

Given the following plant conditions:

- The plant is in MODE 6
- The last reactor vessel head closure bolt is being removed
- You have been tasked to develop the list of qualified personnel for the next shift

Which of the following positions will be required in accordance with Technical Specification 5.2.2, PLANT STAFF?

1) Radiation Protection Technician
2) Shift Technical Advisor

A. 1) Required
   2) Required
B. 1) Required
   2) **NOT** required
C. 1) **NOT** required
   2) Required
D. 1) **NOT** required
   2) **NOT** required

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and the STA is normally assigned, but not required, during MODE 6 operations.

B. CORRECT. In accordance with Technical Specification 5.2.2.c "An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor." According to 5.2.2.f "The STA shall be assigned to the shift crew while the plant is in MODE 1, 2, 3, or 4."

C. INCORRECT. Plausible since the Radiation Protection Technician is not required when the reactor is defueled during the outage and the STA is normally assigned, but not required, during MODE 6 operations.

D. INCORRECT. Plausible since the Radiation Protection Technician is not required when the reactor is defueled during the outage and the second part is correct.

Technical Reference(s): Technical Specification 5.2.2 (Rev 108)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RTS00C 4.03 (As available)

Question Source: Bank # 158410
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	002 K4.02	
	Importance Rating	3.5*	

Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: Monitoring reactor vessel level

RO Question #12

Given the following plant conditions:

- A LOCA has occurred
- RCPs have been tripped
- RCS pressure is 500 psig and lowering
- CETs indicate 450°F and lowering
- HCO is monitoring Reactor Vessel Level Indication System (RVLIS) indication

Which ONE of the following describes the operation of RVLIS?

A. Uses RCS pressure only as input for specific gravity calculation.

B. T_{COLD} and CETs are averaged for the specific gravity calculation.

C. T_{COLD} input to RVLIS is disabled and CETs are used for specific gravity calculation.

D. RCP flow function generator is provided a delta pressure input to compensate for additional head of RHR or SI Pumps.

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible because RVLIS would use only RCS pressure for specific gravity when RCPs are OFF and CET temperature exceeds T_{SAT} for the current RCS pressure. Incorrect since with the given conditions, CETs are below T_{SAT} with the given plant conditions.

B. INCORRECT. Plausible because in Natural Circulation Mode, RVLIS computes the specific gravity in the middle of the core by averaging CETs and T_{COLD}.

C. CORRECT. In accordance with P-10, Section 5.14.C.2, "If SI or RHR flow is indicated, RVLIS will cut out the T_{COLD} input, and compute the specific gravity using the CETs only." With the given RCS pressure, the candidate should determine that SI flow is occurring.

D. INCORRECT. Plausible because during RCP operation, RVLIS compensates the known "solid core D/P" to account for the head of the RCPs and the presence of voids. The candidate may believe that RVLIS uses a similar compensation method when the SI or RHR Pumps are in operation.

Technical Reference(s): P-10 (p56-57, 87-93; Rev 01901)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R6701C 1.07 (As available)

Question Source: Bank # 159174
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.5	
	Importance Rating	2.9	

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

RO Question #13

Given the following plant conditions:

- Plant is operating at 100% power
- S/G tube leakage is 0.8 gpm
- R-15, AIR EJECTOR AND GLAND STEAM EXHAUST MONITOR, is in alarm
- The operating crew enters AP-SG.1, STEAM GENERATOR TUBE LEAK
- The S/G with a tube leak has **NOT** been identified

Which ONE of the following will provide the quickest indication of which S/G has the leaking tube in accordance with ATT-16.1, ATTACHMENT SGTL?

A. R-19, STEAM GENERATOR BLOWDOWN MONITOR, after isolating flow to Blowdown HX 'A'

B. R-31 or R-32, STEAM LINE MONITORS, RMS rack indications

C. R-47, AIR EJECTOR NOBLE GAS MONITOR (using conversion table)

D. Chemistry sample for gross activity on individual S/Gs

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with ATT-16.1, step 8, the operator is directed to have the EO close inlet block valve for S/G blowdown HX 'A' outlet flow and monitor R-19 for two minutes to determine the leaking S/G.

B. INCORRECT. Plausible AP-SG.1 has the operator check R-31/32 for indication of S/G leak rate. Incorrect since the leak rate is too small to determine by the Steam Line Monitors as indicated on the RMS rack.

C. INCORRECT. Plausible AP-SG.1 has the operator check R-47 for indication of S/G leak rate. Incorrect since R-47 will not indicate a specific S/G.

D. INCORRECT. Plausible since ATT-16.1 directs the operator to have RP implement the procedure to determine S/G leak rate. Incorrect in that this will take at least 15 minutes.

Technical Reference(s): AP-SG.1 (p8; Rev 01600)
(Attach if not previously provided, ATT-16.1 (Rev 00600)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP32C 2.01; R3901C 1.04 (As available)

Question Source: Bank # 158415
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .11
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 057 2.4.21	
	Importance Rating	4.0	

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.

RO Question #14

Given the following plant conditions:

- The plant is operating at 100% power
- A loss of an Instrument Bus occurs, and appropriate action is taken to stabilize the plant
- No instrument defeat actions have been taken
- The Instrument Bus has **NOT** been transferred to another source of power

What effect, if any, does this event have on the following Rod Insertion Limit values?

(1) CALCULATED (by RIL computer);

AND

(2) ACTUAL (COLR)

A. (1) not impacted
(2) not impacted

B. (1) not impacted
(2) lowers

C. (1) lowers
(2) not impacted

D. (1) lowers
(2) lowers

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since the candidate may believe the inputs for calculated and actual Rod Insertion Limits are the same, and the second part is correct.

B. INCORRECT. Plausible since the candidate may confuse the difference between calculated and actual Rod Insertion Limits and the inputs into each.

C. CORRECT. In accordance with P-10, Section 5.12.C.1.c, the plant response to an Instrument Bus failure is "RIL will be abnormally low due to the input into Average ΔT from the failed ΔT." The COLR Rod Insertion Limit is based upon actual reactor power vice indicated reactor power; therefore, this limit does not change for loss of an Instrument Bus.

D. INCORRECT. Plausible since the first part is correct and the candidate may believe the inputs for calculated and actual Rod Insertion Limits are the same.

Technical Reference(s): P-10 (p48; Rev 01901)
(Attach if not previously provided, Cycle 39 COLR (pCOLR-11; Rev 0)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2901C 1.10 (As available)

Question Source: Bank # 159423
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A3.01	
	Importance Rating	4.1	

Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

RO Question #15

An Operations Department expectation is for the operating crew to initiate SI and CI if Containment pressure has reached 2 psig and is rising in an uncontrolled manner.

What advantage in Containment cooling and controlling Containment pressure does this initiation give the operators?

SI initiation will . . .

A. cause Service Water isolation to allow maximum Service Water cooling.

B. ensure all Service Water Pumps are running to maximize Containment cooling.

C. provide feedline isolation to isolate Feedwater leakage and stop Containment pressure rise.

D. ensures maximum Service Water flow to Containment Recirc Fan Coolers.

Answer: **D**

Explanation (Optional):

A. INCORRECT. Incorrect because for Service Water Isolation to occur there must be an undervoltage condition on either Bus 14 or 16 concurrent with an SI signal. Plausible since the purpose of Service Water isolation is to ensure that adequate cooling is provided for safeguards equipment.

B. INCORRECT. Incorrect since the Safety Injection sequencer does not ensure that all four SW Pumps are running, only the two that are selected to automatically start. Plausible since under normal conditions the non-running Service Water Pumps are selected to automatically start on a Safety Injection signal, so following a LOCA all four SW Pumps could be running.

C. INCORRECT. Plausible since the Safety Injection signal will result in a Feedwater Isolation signal. However, it will not prevent the contents of a S/G with a ruptured feed line inside Containment from being emptied into CNMT causing CNMT pressure rise.

D. CORRECT. In accordance with 33013-1353, Sheet 8, a Safety Injection signal will start all four Containment Recirc Fans. According to UFSAR, Section 9.2.1.2.4, AOV-4561 and AOV-4562 fail OPEN to ensure maximum Service Water flow is supplied to the Containment Recirc Fan Coolers.

Technical Reference(s): 33013-1353 (sh8; Rev 3)
(Attach if not previously provided, UFSAR Section 9.2.1.2.4 (Rev 26)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2201C 1.09 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165600

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments: K/A match because the operators have been placed in a situation where they are monitoring a rise in Containment pressure and have an expectation to insert a SI signal. The answer is based on criteria operators would monitor following safeguard signal actuation in E-0, ATT-27.0 regarding Containment cooling.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 A2.01	
	Importance Rating	3.5*	

Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS

RO Question #16

Given the following plant conditions:

- Plant is operating at 100% power
- AP-SW.2, LOSS OF SERVICE WATER, has been entered due to issues associated with the Service Water Pumps
- 'C' SW Pump is the only SW Pump running
- The operating crew is monitoring plant equipment

1) What is the impact to plant equipment if the given plant conditions remain unchanged?

AND

2) What actions will be required in accordance with AP-SW.2 if plant equipment temperatures continue to rise?

A. 1) Condensate Pump motor bearing temperatures will slowly rise while Condensate Booster Pump motor bearing temperatures will remain relatively steady.

   2) Perform a load reduction.

B. 1) Condensate Pump motor bearing temperatures will slowly rise while Condensate Booster Pump motor bearing temperatures will remain relatively steady.

   2) Perform a reactor trip.

C. 1) Condensate Pump and Condensate Booster Pump motor bearing temperatures will slowly rise.

   2) Perform a load reduction.

D. 1) Condensate Pump and Condensate Booster Pump motor bearing temperatures will slowly rise.

   2) Perform a reactor trip.

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with 33013-1251, Sheet 1, Service Water provides cooling for the Condensate Pump motor bearings. Therefore, with only one Service Water Pump running, reduced cooling will be supplied to the Condensate Pump motor bearings resulting in rising temperatures. The Condensate Booster Pump motor bearings are not supplied with Service Water cooling so the temperatures should remain relatively steady. AP-SW.2, step 5 RNO requires a load reduction with only one Service Water Pump running and plant equipment temperatures rising (given that plant conditions continue to degrade).

B. INCORRECT. Plausible since the first part is correct and AP-SW.2, step 2 RNO directs the operators to trip the reactor if no Service Water Pumps are running.

C. INCORRECT. Plausible since Service Water provides cooling for the Condensate Pump motor bearings and the candidate may believe that Service Water cooling is also provided to the Condensate Booster Pump motor bearings. Also, the second part is correct.

D. INCORRECT. Plausible since Service Water provides cooling for the Condensate Pump motor bearings and the candidate may believe that Service Water cooling is also provided to the Condensate Booster Pump motor bearings. Additionally, AP-SW.2, step 2 RNO directs the operators to trip the reactor if no Service Water Pumps are running.

Technical Reference(s): AP-SW.2 (Rev 00801)
(Attach if not previously provided, 33013-1251, Sheet 1 (Rev 45)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R5101C 1.04, RAP33C 2.01 (As available)

Question Source: Bank # 159171
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 Ginna ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 077 AA2.03	
	Importance Rating	3.5	

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbance: Generator current outside the capability curve

RO Question #17

Plant conditions occurred as follows:

- While at rated power, there was a disturbance on the electrical power grid which required the unit to perform a load reduction
- The CO reports that Main Generator MVARs have gone from 150 MVARs OUT to 40 MVARs IN and rising on the MCB
- Generator loading is 500 MWe

Using the provided Generator Capability Table, if current conditions continue:

1) Which Megavar limit will be met **FIRST**?

AND

2) In this condition, the grid is more ____________.

A. 1) -159 Megavars
   2) inductive

B. 1) 325 Megavars
   2) inductive

C. 1) -159 Megavars
   2) capacitive

D. 1) 325 Megavars
   2) capacitive

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and the candidate may believe that with the Main Generator taking MVARs IN that an inductive state exists.

B. INCORRECT. Plausible since the Megavars limit is correct for the generator loading in an inductive (overexcited) state and the candidate may believe that voltage is leading current when a capacitive state exists.

C. CORRECT. With MVARs coming into the Main Generator, a capacitive state exists on the electrical grid and the generator will be underexcited. In accordance with O-6.9, Attachments 16 and 17, the lower half of the curve is used (underexcited) and the limit is -159 Megavars.

D. INCORRECT. Plausible since the Megavars limit is correct for the generator loading in an inductive (overexcited) state and the second part is correct.

Technical Reference(s): O-6.9 (p35-36; Rev 037)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: O-6.9, Attachment 16

Learning Objective: R0501C 1.09 (As available)

Question Source: Bank # 158654
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2008 Ginna ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43 _____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 A4.10	
	Importance Rating	3.1*	

Ability to manually operate and/or monitor in the control room: Conditions that require the operation of two CCW coolers

RO Question #18

Which of the following describes the MINIMUM Component Cooling Water (CCW) configuration required to support a cool down of the RCS to Cold Shutdown in accordance with O-2.2, PLANT SHUTDOWN FROM HOT SHUTDOWN TO COLD CONDITIONS?

A. One CCW Train in service provides adequate cooling for the entire cooldown from MODE 1 to MODE 5.

B. One CCW Train in service until RCS is in MODE 3 and less than 400°F, then both Trains of CCW in service to MODE 5.

C. One CCW Train in service through MODE 4, then both Trains of CCW in service in MODE 5.

D. Both Trains of CCW are in service for the entire cooldown from MODE 1 to MODE 5.

Answer: B

Explanation (Optional):

A. INCORRECT. Plausible since one CCW train will provide adequate cooling, dependent upon SW temperature, to complete the RCS cooldown to cold conditions. Incorrect since O-2.2 requires the second CCW train placed in service prior to starting RHR Cooling.

B. CORRECT. In accordance with O-2.2, Steps 6.3.13 places the second CCW train in service once RCS temperature is less than 400°F. The second CCW train is not normally placed in service until just prior to achieving 350°F RCS temperature and preparing to transition to RHR Cooling.

C. INCORRECT. Plausible since the candidate may believe that more cooling is required for the RHR System in order to maintain RCS cooldown with the lower Delta-T.

D. INCORRECT. Plausible since the candidate may believe that O-2.2 places the second CCW train in service early in the cooldown process.

Technical Reference(s): O-2.2 (p29; Rev 15700)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ROP04C 1.01 (As available)

Question Source: Bank # 158674
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K5.02	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to PRTS: Method of forming a steam bubble in the PZR

RO Question #19

Given the following plant conditions:

- The plant is in Mode 4
- A bubble is being drawn in the Pressurizer
- Letdown Pressure Control Valve PCV-135 is set in AUTO to maintain RCS pressure at 325 psig
- Current PRZR temperature is 370°F
- PRZR temperature is rising at 1.2°F per minute

Assuming the current trends continue, which ONE of the following describes:

1) the approximate time before a bubble is formed;

AND

2) the indication that a bubble has been formed?

A. 1) 44 minutes;
2) large increase in Pressurizer pressure for a given change in temperature

B. 1) 44 minutes;
2) letdown flow greater than charging flow with stable or slightly rising pressure

C. 1) 49 minutes;
2) large increase in Pressurizer pressure for a given change in temperature

D. 1) 49 minutes;
2) letdown flow greater than charging flow with stable or slightly rising pressure

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible since the candidate may neglect to convert RCS pressure to psia for use in the Steam Tables. Also, the large Pressurizer pressure response would be if the plant was solid without bubble formation in progress.

B. INCORRECT. Plausible since the candidate may neglect to convert RCS pressure to psia for use in the Steam Tables. The second part is correct.

C. INCORRECT. Plausible since the first part is correct and the large Pressurizer pressure response would be if the plant was solid without bubble formation in progress.

D. CORRECT. Using Steam Tables, the saturation temperature for 340 psia (325 psig) is approximately 429°F. Therefore, at the current Pressurizer heatup rate, it will take approximately 49 minutes to reach saturation and begin forming steam. As the bubble forms, RCS pressure will begin to slowly rise causing PCV-135 to OPEN to maintain RCS pressure at setpoint and letdown flow to rise.

Technical Reference(s): O-1.1 (p59; Rev 16900)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: ROP00C 1.01 (As available)

Question Source: Bank # 158672
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	079 2.1.20	
	Importance Rating	4.6	

Ability to interpret and execute procedure steps.

RO Question #20

Given the following plant conditions:

- The reactor tripped following a turbine trip
- A loss of Offsite Power has occurred
- Both Diesel Generators failed to start and cannot be started
- The operating crew is performing ER-ELEC.5, SECURITY DIESEL FEED TO BUS 13
- The Diesel Air Compressor is out of service for maintenance

Which ONE of the following describes the purpose for performing ER-ELEC.5?

A. The Service Air Compressor is started which will be used to isolate RCP seal return.

B. The ‘A’ Instrument Air Compressor is started which will be used to allow control of the TDAFW Pump.

C. The Seal Oil Backup Pump is started to prevent hydrogen from leaking out of the Main Generator.

D. The ‘A’ Reactor Compartment Cooling Fan is started to provide cooling to the Source Range NIS detectors.

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with ER-ELEC.5, step 6.8 “IF the Diesel Air Compressor CANNOT supply Instrument Air, THEN START the service air compressor”. Step 6.9 states “WHEN service air or diesel air is supplying IA header, THEN PERFORM the following to isolate RCP seal return:”.

B. INCORRECT. Plausible since ER-ELEC.5 starts an air compressor if the Diesel Air Compressor is not available and Instrument Air provides the mode of force to control the TDAFW Pump discharge AOVs. Incorrect since the ‘A’ Instrument Air Compressor required Service Water Pumps running to provide cooling water and an EO can locally control the TDAFW Pump combined discharge valve.

C. INCORRECT. Plausible since hydrogen leakage from the Main Generator is a concern during a loss of AC power event. Incorrect since the DC Emergency Oil Pump is utilized for this concern.

D. INCORRECT. Plausible since ER-ELEC.5 starts a Shroud Fan for reactor vessel head cooling and overheating of the nuclear instrument detectors is a concern.

Technical Reference(s): ER-ELEC.5 (Rev 00601)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RER05C 2.01 (As available)

Question Source: Bank # 159177
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments:

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	003 K5.02	
Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP coastdown on RCS parameters

RO Question #21

Given the following plant conditions:

- Operators are performing a plant startup
- Reactor power is 80%
- #11 Transformer experiences a fault and lockout
- Automatic reactor trip occurred

What is the effect on RCS loop Delta-T from the transient?

5 seconds following the #11 Transformer fault, Delta-T will be __________ than the 80% value and will __________________ during RCP coast down.

A. higher; continue to rise

B. lower; then rise

C. higher; then lower

D. lower; continue to lower

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since the candidate may believe that the loss of power for the RCPs and the resulting loss of flow will cause Delta-T to initially rise. The second part is correct.

B. CORRECT. Reactor trip will occur with loss of flow from #11 transformer fault. Trip will result in lower RCS temperature and Delta-T. RCP coast down will allow rising Delta-T as the fluid flow slows down in the loops, natural circulation will bring Delta-T down as flow reestablishes.

C. INCORRECT. Plausible since the candidate may believe that the loss of power for the RCPs and that the RCP coast down will then bring loop Delta-T back down as natural circulation builds in.

D. INCORRECT. Plausible the first part is correct and the candidate may believe that the RCP coast down will then bring loop Delta-T back down as natural circulation builds in.

Technical Reference(s): UFSAR Figure 14.6-9 Sheets 1-5 (Rev 26)
(Attach if not previously provided, UFSAR Figure 14.6-10 Sheets 1-2 (Rev 26)
including version/revision number) UFSAR Section 15.3.1 (Rev 26)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1001C 1.10 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New 165597

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE 055 EA1.04	
	Importance Rating	3.5	

Ability to operate and monitor the following as they apply to a Station Blackout: Reduction of loads on the battery

RO Question #22

Given the following plant conditions:

- Station blackout occurred 30 minutes ago
- ECA-0.0, LOSS OF ALL AC POWER, has been entered
- RG&E reports that Offsite Power will be restored within the next 2 hours
- Operating crew is performing Step 20, Check DC Bus Loads

Which ONE of the following completes the statements below:

1) The CO and EO will perform DC load shed in accordance with _______________;

   AND

2) Placing the MFW Pump AC Oil Pumps to OFF will result in the MFW Pump DC Oil Pump stopping after _______.

NOTE: ATT-8.0, ATTACHMENT DC LOADS
FSG-4, ELAP DC BUS LOAD SHED/MANAGEMENT

A. 1) ATT-8.0
   2) 6 minutes

B. 1) FSG-4
   2) 6 minutes

C. 1) ATT-8.0
   2) 11 minutes

D. 1) FSG-4
   2) 11 minutes

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and the candidate may not have knowledge of the timer related to the MFW Pump AC and DC Oil Pumps.

B. INCORRECT. Plausible since ECA-0.0 has the operating crew perform FSG-4 once an ELAP is declared in progress of if DC bus voltages lower to ≤ 108.6 VDC and the candidate may not have knowledge of the timer related to the MFW Pump AC and DC Oil Pumps.

C. CORRECT. In accordance with ECA-0.0, Step 20, the operator ensures that DC load shed is in progress by referring to ATT-8.0. The background document states "The UFSAR assumes that the MFW pump DC oil pump operates for 12 minutes following loss of all AC power. This is based on a timer in the MFW pump DC oil pump control circuitry which allows the pump to operate for 11 minutes."

D. INCORRECT. Plausible since ECA-0.0 has the operating crew perform FSG-4 once an ELAP is declared in progress of if DC bus voltages lower to ≤ 108.6 VDC and the second part is correct.

Technical Reference(s): ECA-0.0 (p22; Rev 042)
(Attach if not previously provided, ECA-0.0 Background Document (p74; Rev 01901)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REC00C 2.01; R4301C 1.07 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 159422

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001 K2.02	
	Importance Rating	3.6	

Knowledge of the bus power supplies to the following: One-line diagram of power supply to trip breakers

RO Question #23

Which ONE of the following correctly delineates the order of the power train components from the associated Bus to the Control Rod Drive Mechanisms (CRDMs)?

A. Motor Generator, Trip Breakers, Logic Cabinets, CRDMs
B. Trip Breakers, Motor Generator, Logic Cabinets, CRDMs
C. Trip Breakers, Motor Generator, Power Cabinets, CRDMs
D. Motor Generator, Trip Breakers, Power Cabinets, CRDMs

Answer: **D**

Explanation (Optional):

A. INCORRECT. Incorrect as the logic cabinets are not part of the power flow to the CRDMs. Plausible since the logic cabinets provide control power to the Rod Drive System.

B. INCORRECT. Incorrect as the logic cabinets are not part of the power flow to the CRDMs. Plausible since the candidate may believe that the Reactor Trip Breakers are located on the supply side of the MG sets.

C. INCORRECT. Plausible since all the correct components are listed and the candidate may believe that the Reactor Trip Breakers are located on the supply side of the MG sets.

D. CORRECT. In accordance with UFSAR Figure 7.7-4 and Section 7.7.1.2.5.1, power to the CRDM coils comes from Bus 13/15, to the M-G set motor breaker, through the M-G set generator breaker, to the Reactor Trip Breakers, to the Power Cabinets, then finally to the CRDM coils.

Technical Reference(s):	UFSAR Figure 7.7-4 (Rev 26)
(Attach if not previously provided,	UFSAR Section 7.7.1.2.5.1 (Rev 26)
including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective: R3001C 1.05 (As available)

Question Source: Bank # 159173

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 2000 Callaway ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 .7

55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EPE W/E15 EA1.2	
	Importance Rating	2.7	

Ability to operate and / or monitor the following as they apply to the (Containment Flooding): Operating behavior characteristics of the facility

RO Question #24

Which ONE of the following statements describes the Containment conditions if FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, is entered?

'B' Sump water level greater than 180 inches indicates that water from ______________________ been introduced into the Containment Sump.

A. volumes other than stored water sources (RWST, accumulators, etc.) have

B. one of the stored water sources (RWST, accumulators, etc.) has

C. a natural disaster (severe thunderstorm, tornado, etc.) has

D. the entire RCS volume (DBLOCA) has

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with FR-Z.2 Background Document "An indicated water level in the Containment greater than the maximum expected volume (design basis flood level) is an indication that water volumes other than those represented by the above noted volumes (RCS, RWST, and SI Accumulators) have been introduced into the Containment."

B. INCORRECT. Plausible since the candidate may believe that any of the stored water sources will provide entry conditions into FR-Z.2 since entry into the recirculation phase of post-LOCA cooling assumes that the stored water source volumes have entered the Containment Sump.

C. INCORRECT. Plausible since natural disasters have the ability to introduce large volumes of water in a relatively short amount of time and this water could seep into the Containment Moat.

D. INCORRECT. Plausible since the candidate may believe that any of the stored water sources will provide entry conditions into FR-Z.2 since entry into the recirculation phase of

post-LOCA cooling assumes that the stored water source volumes have entered the Containment Sump.

Technical Reference(s): FR-Z.2 Background Document (p2; Rev 1)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRZ2C 1.01 (As available)

Question Source: Bank # 158670
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 K3.02	
	Importance Rating	3.8	

Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under normal operations

RO Question #25

Given the following plant conditions:

- Plant is operating at 100% power
- A small Instrument Air leak inside Containment causes a slow rise in Containment pressure
- Containment pressure is currently +0.51 psig

In order to ensure that adequate margin to Containment Technical Specification pressure limits is maintained, which ONE of the following indicates the appropriate action to reduce Containment pressure?

A. Verify all Containment Recirculation Fans are running
B. Lower pressure in Containment using the Containment Purge System
C. Maximize Service Water cooling to the Containment Recirc Fan Coolers
D. Lower pressure in Containment using the Containment Mini-Purge System

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since verifying that all Containment Recirc Fans are running would result in lowering Containment pressure if the cause of the increased pressure was due to steam or RCS coolant.

B. INCORRECT. Plausible since the Containment Purge System is used during shutdown Modes to maintain the Containment environment, including maintaining proper Containment pressure.

C. INCORRECT. Plausible since maximizing Service Water cooling to all Containment Recirc Fans would result in lowering Containment pressure if the cause of the increased pressure was due to steam or RCS coolant.

D. CORRECT. In accordance with S-23.2.3, "This procedure will be utilized to purge the containment atmosphere for ALARA, for breathable air quality considerations, for personnel entry, or for other related reasons." This includes lowering Containment pressure when required.

Technical Reference(s): S-23.2.3 (p4; Rev 013)
(Attach if not previously provided, S-23.2.2 (4; Rev 052)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2101C 1.04 (As available)

Question Source: Bank # 165624
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2014 DC Cook ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments: K/A match because the questions asks for the operator to predict which system of control will reduce Containment pressure to prevent exceeding the design pressure (and loss of integrity).

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 A4.04	
	Importance Rating	2.6*	

Ability to manually operate and/or monitor in the control room:
PZR vent valve

RO Question #26

A trip and SI have occurred due to spurious opening of PORV-430. MOV-516, PRZR PORV BLK VLV, has been closed and isolated the RCS leak.

Present plant conditions are:

- Reactor Coolant pressure is 2200 psig and slowly rising
- PRT pressure is 9 psig and slowly lowering
- Operating crew is transitioning to ES-1.1, SI TERMINATION
- Decision has been made to vent the PRT to less than 5 psig

What must the operator do to allow venting the PRT via AOV-527, PRT VENT VALVE?

NOTE: AOV-1786, PRT RCDT ISOL
AOV-1787, PRT RCDT ISOL

A. Reduce pressure to less than 5 psig in PRT and then manually OPEN AOV-527 to the Vent Header.
B. Reset SI, CI, and X and Y relays for AOV-1786 and AOV-1787 and then manually OPEN AOV-527 to the Vent Header.
C. Reduce pressure to less than 5 psig in PRT and then manually OPEN AOV-527 to Containment.
D. Reset SI, CI, and X and Y relays for AOV-1786 and AOV-1787 and then manually OPEN AOV-527 to Containment.

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since a pressure interlock exists for AOV-527 (10 psig) and the candidate may not know the pressure interlock setpoint is met. Also, the flow path from AOV-527 is to the Vent Header.

B. CORRECT. Since PRT pressure is below the pressure interlock setpoint for AOV-527, it can be manually opened from the MCB once a flow path to the Vent Header is established. The SI and CI signals must be reset and then the X and Y relays for AOV-1786 and AOV-1787 must be reset (both valves CLOSE on a CI signal) in order to establish a complete flow path from the PRT Vent Valve to the Vent Header.

C. INCORRECT. Plausible since a pressure interlock exists for AOV-527 (10 psig) and the candidate may not know the pressure interlock setpoint is met. The candidate would have to know that the PRT vents to the Vent Header vice Containment.

D. INCORRECT. Plausible since the actions to establish a flow path are correct, but the candidate would have to know that the PRT vents to the Vent Header vice Containment.

Technical Reference(s): AR-F-9 (Rev 00800)
(Attach if not previously provided, 33013-1258 (Rev 25)
including version/revision number) 33013-1272, Sheet 1 (Rev 11) and Sheet 2 (Rev 17)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1401C 1.07, 1.08 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New 165615

Question History: Last NRC Exam _____

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 015/017 AK2.07	
	Importance Rating	2.9	

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

RO Question #27

Given the following plant conditions:

- The plant is at 100% power
- Annunciator B-10, RCP 1B LABYR SEAL LO DIFF PRESS 15" H2O, is **LIT**
- Annunciator B-18, RCP 1B NO. 1 SEAL HI-LO FLOW 5.0 GPM 1.0, is **LIT**
- 'B' RCP No. 1 seal leakoff flow indicates 8.0 gpm and rising

Which ONE of the following actions should the crew perform?

A. Trip the reactor and stop 'B' RCP.

B. Monitor 'B' RCP seal temperatures to determine if RCP should be stopped.

C. Close the 'B' RCP No. 1 seal leakoff isolation valve and have the reactor shutdown with 'B' RCP secured within the following 30 minutes.

D. Increase seal injection flow to greater than seal leakoff flow and shutdown the unit per O-2.1, NORMAL SHUTDOWN TO HOT SHUTDOWN.

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with AP-RCP.1, step 1 RNO, if #1 seal flow is greater than 8.0 gpm, and #1 seal failure is verified by a lowering labyrinth seal differential pressure or rising seal inlet/outlet temperatures, then the reactor is tripped and the affected RCP is stopped upon completion of E-0 Immediate Actions.

B. INCORRECT. Plausible since AP-RCP.1 provides this guidance if RCP total #1 seal flow is less than 0.8 gpm.

C. INCORRECT. Plausible since AP-RCP.1 contains guidance for closing the #1 seal leakoff isolation valve and requirements to shutdown the reactor for some RCP seal malfunctions.

D. INCORRECT. Plausible since AP-RCP.1 contains this guidance if RCP total #1 seal leakoff flow did not initially require a reactor trip, but has since degraded.

Technical Reference(s): AP-RCP.1 (Rev 01800)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP14C 2.02 (As available)

Question Source: Bank # 158643
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

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

Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: Refueling operations

RO Question #28

Given the following plant conditions:

- The plant is in Mode 6, cavity has been filled to greater than 23 feet, Source Range response checks are in progress
- 'A' RHR Pump is in service
- 'B' RHR Pump is out of service for oil change, expected to be returned to service in 2 hours
- RCS temperature is 80°F
- 'A' RHR Pump trips for an unknown reason

Which ONE of the following describes the actions <u>required by Technical Specifications</u> associated with this event?

A. Immediately initiate Containment closure.

B. Immediately initiate Containment Ventilation Isolation.

C. Immediately suspend all operations that could dilute RCS boron concentration.

D. Immediately isolate Letdown and known RCS drain paths.

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible since Containment closure is a mitigating action when the plant is in Mode 6 during reduced inventory conditions and this action would be required by AP-RHR.2.

B. INCORRECT. Plausible since Containment Ventilation is required to be OPERABLE in this condition and Containment Ventilation Isolation is a mitigating action for most accidents located inside Containment.

C. CORRECT. In accordance with Technical Specification LCO 3.9.4, Condition A.1 "Immediately suspend operations that would cause introduction of coolant into the RCS

with boron concentration less than required to meet the boron concentration of LCO 3.9.1 (≥ 2750 ppm)."

D. INCORRECT. Plausible since AP-RHR.2 isolates Letdown and known drain paths with a loss of RHR during reduced inventory conditions.

Technical Reference(s): Technical Specification 3.9.4 (Rev 122)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2501C 1.12 (As available)

Question Source: Bank # 165613
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2011 Harris ILT
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE W/E05 EA2.2	
	Importance Rating	3.7	

Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

RO Question #29

Given the following plant conditions:

- FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, is in progress
- Bleed and Feed has been established
- 'B' MDAFW Pump has been made available
- CNMT pressure is 8.5 psig
- Hot leg temperatures are 565°F and lowering
- Both S/G narrow range levels are 0%
- Both S/G wide range levels are 90 inches

Which ONE of the following describes:

1) the allowable initial AFW flow rate in accordance with ATT-22.0, ATTACHMENT RESTORING FEED FLOW, **AND**
2) the required S/G level necessary to secure Bleed and Feed?

A. 1) < 100 gpm
   2) 7%
B. 1) < 100 gpm
   2) 25%
C. 1) as desired
   2) 7%
D. 1) as desired
   2) 25%

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and the second part is plausible since this would be the necessary level if Containment was not adverse.

B. CORRECT. In accordance with ATT-22.0, since hot leg temperatures are greater than 550°F, Bleed and Feed has been initiated, RCS temperature is lowering, step 2.c states "Feed flow is restricted to less than or equal to 100 gpm to affected S/G. When S/G level greater than 50 inches [100 inches adverse CNMT], THEN fill as desired to restore narrow range level greater than 7% [25% adverse CNMT]." FR-H.1 step 29 requires narrow range level in at least one S/G to be greater than 7% [25% adverse CNMT] in order to proceed to the steps for securing Bleed and Feed.

C. INCORRECT. Plausible since ATT-22.0 would allow the operator to feed as desired if hot leg temperatures were < 550°F with Bleed and Feed established. The second part is plausible since this would be the necessary level if Containment was not adverse.

D. INCORRECT. Plausible since ATT-22.0 would allow the operator to feed as desired if hot leg temperatures were < 550°F with Bleed and Feed established. The second part is correct.

Technical Reference(s): FR-H.1 (p26 and 28; Rev 04100)
(Attach if not previously provided, Att-22.0 (Rev 00700)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRH1C 2.01 (As available)

Question Source: Bank #
Modified Bank # 158656 (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EPE W/E06 EA2.1	
	Importance Rating	3.4	

Ability to determine and interpret the following as they apply to the (Degraded Core Cooling): Facility conditions and selection of appropriate procedure during abnormal and emergency operations.

RO Question #30

The plant was operating at 100% power when a small break LOCA occurred.

The following plant conditions exist:

- Bus 14 is de-energized
- 'C' SI Pump failed to start, attempts to restart have been unsuccessful
- Both RCPs were tripped
- Average Core Exit Thermocouples are 675°F
- RCS Pressure is 800 psig
- RVLIS indicates 50%
- Both S/G levels are 20% and rising
- Both S/G pressures are 1000 psig and stable
- RWST level 80%
- CNMT Pressure 6 psig
- CNMT Radiation 10 R/hr
- The operators have just entered ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION

Which ONE of the following describes the required procedure flowpath?

A. Remain in ES-1.2
B. Transition to FR-C.1, RESPONSE TO INADEQUATE CORE COOLING
C. Transition to FR-C.2, RESPONSE TO DEGRADED CORE COOLING
D. Transition to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since ES-1.2 performance is the expected flowpath during a small break LOCA when no challenges exist to the Critical Safety Function Status Trees (CSFSTs).

B. INCORRECT. Plausible since transition to FR-C.1 would be made once Core Exit Thermocouples exceed 700°F with the remainder of the given plant conditions.

C. CORRECT. In accordance with F-0.2, Core Cooling CSFST, with CETs < 1200°F, RCS subcooling less than FIG-1.0, Figure Min Subcooling, requirements, no RCPs running, CETs < 700°F, and RVLIS less than 55%, the operator is directed to transition to FR-C.2.

D. INCORRECT. Plausible since both S/G water levels are below the adverse water level entry requirement for FR-H.1; however, since plant conditions have not been given that AFW is malfunctioning, the operator must assume that AFW is operating as designed and therefore ≥ 200 gpm AFW flow is available.

Technical Reference(s): F-0.2 (Rev 00600)
(Attach if not previously provided, FIG-1.0 (Rev 00200)
including version/revision number)

Proposed references to be provided to applicants during examination: FIG-1.0

Learning Objective: RFRC2C 1.02 (As available)

Question Source: Bank #
Modified Bank # 158666 (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 027 2.4.46	
	Importance Rating	4.2	

Ability to verify that the alarms are consistent with the plant conditions.

RO Question #31

Given the following plant conditions:

- The plant is operating at 100% power
- RCS Pressure Channels (PT-420 & PT-420A) are 2260 psig and rising
- PRZR Backup Heaters are ON in AUTO
- The following Annunciators are LIT:
  - F-10, PRESSURIZER LO PRESS 2205 PSI
  - F-23, RCS OTΔT CHANNEL ALERT
  - F-27, PRESSURIZER LO PRESS CHANNEL ALERT 1873 PSI

Which ONE of the following identifies the failed channel AND includes the system response to this failure, assuming **NO OPERATOR ACTION** is taken?

A. PT-429 failed; the PRZR Spray Valves will modulate OPEN
B. PT-429 failed; one PRZR PORV will OPEN
C. PT-449 failed; the PRZR Spray Valves will modulate OPEN
D. PT-449 failed; one PRZR PORV will OPEN

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since there are two control channels for PRZR pressure control. PT-429 is not normally selected for control. Spray valves will not OPEN due to the controlling channel failing low.

B. INCORRECT. Plausible since there are two control channels for PRZR pressure control. PT-429 is not normally selected for control. Only one PORV (PCV-430) will OPEN on RCS high pressure since PCV-431C will see its control signal low.

C. INCORRECT. Plausible since the failed channel is correct and with RCS pressure given as high, the candidate may believe that the PRZR Spray Valves will modulate OPEN as they normally would.

D. CORRECT. In accordance with P-10, Instrument Failure Reference Manual, Section 5.2, PT-449 failing low will result in PRZR heaters ON, PORV-431C is prevented from opening, and PORV-430 will still operate. Annunciator F-10 alarms due to PRZR Pressure Controller, PCV-431K, seeing that PT-449 failing low. Annunciators F-23 and F-27 alarm due to PT-449 failing low.

Technical Reference(s): P-10 (p12-15; Rev 01901)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC02C 1.06 (As available)

Question Source: Bank # 158645
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 Ginna ILT RO Retake
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 A1.01	
	Importance Rating	2.8	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW flow rate

RO Question #32

Given the following plant conditions:

- Operators are performing O-2.2, PLANT SHUTDOWN FROM HOT SHUTDOWN TO COLD CONDITIONS, and are preparing to place RHR in service
- A second CCW Pump has been started and both MOV-738A and MOV-738B, CCW SUPPLY TO RHR HEAT EXCHANGERS, have been opened
- The HCO is monitoring PPCS point F0619, COMPONENT COOLING LOOP TOTAL FLOW, and reports that flow is 5000 gpm

1) The crew will reduce CCW flow to ___________ maximum.

AND

2) What is the reason for these actions?

A. 1) < 4900 gpm
   2) CCW Heat Exchanger tube vibration

B. 1) < 4900 gpm
   2) CCW Pump runout

C. 1) < 2400 gpm
   2) CCW Heat Exchanger tube vibration

D. 1) < 2400 gpm
   2) CCW Pump runout

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with O-2.2 step 6.3.13.4 "WHEN two CCW HXs are in service, THEN ENSURE CCW flow is less than or equal to 4900 GPM". According to P-4, step 6.2.12 "Maximum CCW flow is as follows: Less than 4900 gpm with two CCW Heat Exchangers (two CCW Pumps in service". P-4, step 6.2.13 states "These limits were established to prevent heat exchanger tube damage and wall thinning due to flow induced vibration."

B. INCORRECT. Plausible since the first part is correct and CCW Pump runout could be considered as CCW flow rates are well above normal values.

C. INCORRECT. Plausible since the maximum CCW flow for one CCW Heat Exchanger (one CCW Pump in service) is < 2400 gpm according to P-4. Also, the second part is correct.

D. INCORRECT. Plausible since the maximum CCW flow for one CCW Heat Exchanger (one CCW Pump in service) is < 2400 gpm according to P-4. Additionally, CCW Pump runout could be considered as CCW flow rates are well above normal values.

Technical Reference(s): P-4 (p8; Rev 02602)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2801C 1.09 (As available)

Question Source: Bank # 165616
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43 ______

Comments:

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

Knowledge of the operation implications of the following concepts as they apply to Steam Generator Tube Leak: Leak rate vs. pressure drop

RO Question #33

The operating crew is shutting down the plant from 100% Rated Thermal Power in response to a Steam Generator tube leak per AP-SG.1, STEAM GENERATOR TUBE LEAK.

Which ONE of the following correctly states the trend during the SHUTDOWN **AND** the reason?

Actual leak rate will ________ during the shutdown. (Assume that the geometric size of the flaw remains constant).

A. Rise; RCS temperature is lowered, causing greater density of primary fluid.

B. Rise, because the primary to secondary pressure difference is reduced as power is lowered.

C. Lower; RCS temperature is lowered, causing greater density of primary fluid.

D. Lower; because the primary to secondary pressure difference is reduced as power is lowered.

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since the RCS density will rise as power is reduced and temperature lowers in the RCS. However, this does not have much of an effect on leak rate.

B. INCORRECT. Plausible since the candidate may believe that since the pressure difference is lower that the leak rate will rise.

C. INCORRECT. Plausible since the first part is correct and RCS density will rise as power is reduced and temperature lowers in the RCS. However, this does not have much of an effect on leak rate.

D. CORRECT. As reactor power is reduced, S/G pressure rises resulting in a lower differential pressure across the tube leak causing S/G leak rate to lower. In accordance with AP-SG.1 Background Document for step 7 "There is information that suggests that merely reducing power below 50% significantly reduces tube leakage and reduces the probability of a tube rupture."

Technical Reference(s): (Attach if not previously provided, including version/revision number) AP-SG.1 Background Document (p6; Rev 00801)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP32C 1.03 (As available)

Question Source: Bank # 165608

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 2012 Vogtle ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 .8

55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011 A2.09	
	Importance Rating	2.9?	

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High ambient reflux boiling temperature effect or indicated PZR level

RO Question #34

ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, is being performed. Prior to cooldown of 100°F/hr being performed, the procedure has a step to turn all Pressurizer Heaters OFF.

Which ONE of the following describes why Pressurizer Heaters are turned OFF in ES-1.2?

A. Pressurizer level will be lowering during cooldown and the heaters would trip.

B. Refill of the Pressurizer with colder water could cause damage to the heaters from thermal shock.

C. Dissolution of hydrogen in the reference leg could have occurred causing inaccurate high level indication and allow heaters to be uncovered.

D. A large delta temperature across the Surge Line will be created and cause thermal stresses on the Surge Line.

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible since ES-1.2 steps will be directing a 100°F/hr cooldown and the possibility of Pressurizer level lowering and the heater automatically tripping on low level could occur.

B. INCORRECT. Plausible since refilling of the Pressurizer with colder SI water from the RWST will result in cold water entering Pressurizer. The candidate may believe this could cause heater damage from changing temperatures.

C. CORRECT. In accordance with ES-1.2 Background Document, the purpose of the CAUTION before step 7 is "To ensure that the PRZR heaters are not energized when the pressurizer water level indication may be affected by large positive level measurement errors caused by dissolution of hydrogen in the reference leg."

D. INCORRECT. Plausible since Surge Line delta-T is a concern for some types of transients.

Technical Reference(s): ES-1.2 Background Document (p28; Rev 013)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RES12C 1.02 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 158664

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035 K5.01	
	Importance Rating	3.4	

Knowledge of operational implications of the following concepts as they apply to the S/GS: Effect of secondary parameters, pressure, and temperature on reactivity

RO Question #35

Given the following plant conditions:

- A reactor startup is in progress following a mid-cycle outage
- Reactor power is 2% when a S/G ARV fails OPEN
- RCS T_{AVG} lowers and stabilizes at 538°F

Which ONE of the following:

1) Predicts the plant response?

   AND

2) Operators must restore T_{AVG} above the Technical Specification minimum or be subcritical in MODE 2 within ____________?

A. 1) reactor power rises
   2) 15 minutes

B. 1) reactor power rises
   2) 30 minutes

C. 1) reactor power lowers
   2) 15 minutes

D. 1) reactor power lowers
   2) 30 minutes

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and there are some Technical Specification Conditions with a 15 minute Completion Time required.

B. CORRECT. The failed OPEN S/G ARV causes the associated S/G pressure to lower resulting in T_{AVG} lowering. Due to the negative Moderator Temperature Coefficient (MTC), positive reactivity is added causing reactor power to rise. In accordance with Technical Specification 3.4.2, Condition A "T_{AVG} in one or both RCS loops not within limit (≥540°F)", the Required Action is "Be in MODE 2 with $K_{eff} < 1.0$" with a Completion Time of "30 minutes).

C. INCORRECT. Plausible since a reactor with a positive MTC would result in reactor power lowering, but this would not be the condition given that the plant is at mid-cycle. The second part is plausible since there are some Technical Specification Conditions with a 15 minute Completion Time required.

D. INCORRECT. Plausible since a reactor with a positive MTC would result in reactor power lowering, but this would not be the condition given that the plant is at mid-cycle. The second part is correct.

Technical Reference(s): Technical Specification 3.4.2 (Rev 122)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1001C 1.12 (As available)

Question Source: Bank # 165625
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 Ginna ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5

55.43 ____________ ____________

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 A1.10	
	Importance Rating	2.9*	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Air ejector PRM

RO Question #36

Given the following plant conditions:

- Plant is operating at 100% power
- PPCS R-47G indicates > 75 gpd and rising at 50 gpd/hr
- Operating crew entered AP-SG.1, STEAM GENERATOR TUBE LEAK
- A 1%/min downpower has commenced to lower reactor power to < 50% within an hour
- The CO reports R-47G indication has been rising during the downpower and recommends load reduction be raised to 3%/min
- R-47G indication is currently 1300 gpd and rising

NOTE: AP-TURB.5, RAPID LOAD REDUCTION

1) Which ONE of the following will the Unit Supervisor perform based on R-47G indication?
2) Which procedure provides the guidance for manipulating Main Turbine EHC controls for the associated load reduction rate?

A. 1) Raise load reduction rate to 3%/min
   2) AP-SG.1

B. 1) Raise load reduction rate to 3%/min
   2) AP-TURB.5

C. 1) Maintain load reduction rate of 1%/min
   2) AP-TURB.5

D. 1) Maintain load reduction rate of 1%/min
   2) AP-SG.1

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible since AP-SG.1 contains steps that direct operators to perform a load reduction at 3 %/min. Incorrect because according to AP-SG.1 NOTE prior to step 7 "R-47 should not be used to determine if the rate of power reduction should be changed."

B. INCORRECT. Plausible since AP-TURB.5 is used to perform load reductions during performance of AP-SG.1 and AP-TURB.5 can be used for load reductions rates up to 5 %/min. Incorrect because according to AP-SG.1 NOTE prior to step 7 "R-47 should not be used to determine if the rate of power reduction should be changed."

C. CORRECT. In accordance with AP-SG.1, step 7.b RNO, a load reduction is commenced to reduce reactor power to less than 50% within one hour by referring to AP-TURB.5. This would require a load reduction rate of 1 %/min. The NOTE prior to step 7 states "once the power reduction has begun, R-47 should not be used to determine if the rate of power reduction should be changed."

D. INCORRECT. Plausible since the load reduction rate is correct and AP-SG.1 contains steps that direct operators to perform a load reduction.

Technical Reference(s): AP-SG.1 (p1-11; Rev 01600)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP32C 1.03 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165602

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE 003 AK1.04	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: Effects of power level and control position on flux

RO Question #37

Given the following plant conditions:

- Plant is being brought online following a forced outage
- Operating crew is performing O-1.2, PLANT STARTUP FROM HOT SHUTDOWN TO FULL LOAD
- Reactor power is 70% and slowly rising
- EHC controls are in OPER. PAN – IMP IN
- The following alarms are received:
  - C-5, PPCS ROD SEQUENCE OR ROD DEVIATION
  - C-14, ROD BOTTOM
  - F-16, AVERAGE TAVG – TREF DEVIATION ± 5°F
- HCO reports that Control Rod D-4 has dropped

1) What is the effect on total reactor power with no operator actions?

AND

2) How will the crew **initially** raise T_{AVG} to match T_{REF}?

A. 1) goes down and stabilizes at a lower level
   2) dilute the RCS

B. 1) goes down and stabilizes at a lower level
   2) manually reduce turbine load

C. 1) initially goes down and returns to approximately 70%
   2) dilute the RCS

D. 1) initially goes down and returns to approximately 70%
   2) manually reduce turbine load

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible since a dropped rod with the plant operating at power will result in a lower total reactor power since the turbine EHC controls are in OPER PAN – IMP OUT. Also, AP-RCC.3, step 6 allows for dilution of the RCS to raise T_{AVG} after the initial load reduction to match temperatures is completed in step 3.

B. INCORRECT. Plausible since a dropped rod with the plant operating at power will result in a lower total reactor power since the turbine EHC controls are in OPER PAN – IMP OUT. The second part is correct.

C. INCORRECT. Plausible since the first part is correct and AP-RCC.3, step 6 allows for dilution of the RCS to raise T_{AVG} after the initial load reduction to match temperatures is completed in step 3.

D. CORRECT. Reactor power initially drops due to the negative reactivity insertion from the control rod. Since the turbine EHC controls are in OPER PAN – IMP IN, the Turbine Control Valves will OPEN to maintain turbine first stage pressure constant. This results in total reactor power returning to approximately the initial value. In accordance with AP-RCC.3 RNO, step 3 with T_{AVG} less than T_{REF}, the operator is instructed to lower turbine load manually to match temperatures.

Technical Reference(s): AP-RCC.3 (p3-5; Rev 010)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP29C 2.01 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 158658

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 A2.18	
	Importance Rating	2.6*	

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of premature opening of breaker under load

RO Question #38

Given the following plant conditions:

- Plant is operating at 100% power
- A loss of Offsite Power has occurred
- Operating crew has entered AP-ELEC.1, LOSS OF 12A AND/OR 12B BUSSES
- Offsite Power has been restored to Busses 12A and 12B
- Crew is ready to transfer 480 VAC safeguards busses to Normal supply
- Redundant loads have been started on Bus 16
- The CO inadvertently trips D/G A BUS 14 SUPPLY BREAKER

Which ONE of the following lists:

1) the consequence of the D/G Supply Breaker being tripped at this time;
   AND
2) the procedure that will be used to restore normal electrical alignment?

NOTE: AP-ELEC.14/16, LOSS OF SAFEGUARDS BUS 14/16
ER-ELEC.1, RESTORATION OF OFFSITE POWER

A. 1) Bus 14 Normal Feed Breaker will CLOSE
   2) ER-ELEC.1

B. 1) Bus 14 Normal Feed Breaker will CLOSE
   2) AP-ELEC.14/16

C. 1) Bus 14 D/G A Supply Breaker will reclose once control switch is released to the AFTER-TRIP position
   2) ER-ELEC.1

D. 1) Bus 14 D/G A Supply Breaker will reclose once control switch is released to the AFTER-TRIP position
   2) AP-ELEC.14/16

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since ER-ELEC.1 places Bus 14 back on its normal power supply; however, the normal feed breaker control switch must be held in the CLOSE position and the D/G BUS 14 Supply Breaker OPEN for the normal feed breaker to CLOSE. The second part is correct.

B. INCORRECT. Plausible since ER-ELEC.1 places Bus 14 back on its normal power supply; however, the normal feed breaker control switch must be held in the CLOSE position and the D/G BUS 14 Supply Breaker OPEN for the normal feed breaker to CLOSE. Also, AP-ELEC.14/16 is plausible since this is the procedure the operator would use for a loss of Bus 14.

C. CORRECT. According to 10905-0101, Sheet 1, once the D/G Bus 14 Supply Breaker control switch is taken to After-Trip position, since the breaker is open, the D/G is up to speed and voltage, the X Relay would energize causing the Closing Coil to energize closing the D/G Bus 14 Supply Breaker. According to 10905-0065, Sheet 1, the control switch for the Bus 14 Normal Feed Breaker must be taken to the CLOSE position to energize the X Relay and subsequently the Closing Coil to close the breaker. ER-ELEC.1, Section 6.5 directs the operator actions necessary to transfer Bus 14 from the D/G to Offsite Power.

D. INCORRECT. Plausible since the first part is correct and AP-ELEC.14/16 is plausible since this is the procedure the operator would use for a loss of Bus 14.

Technical Reference(s): ER-ELEC.1 (p17-26; Rev 01801)
(Attach if not previously provided, 10905-0101 Sheet1 (Rev 12), Sheet 2 (Rev 2)
including version/revision number) 10905-0065 Sheet1 (Rev 7), Sheet 2 (Rev 2)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0701C 1.07, RER18C 1.04 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165605

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE 028 AK2.03	
	Importance Rating	2.6	

Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Controllers and positioners

RO Question #39

Given the following plant conditions:

- 100% Reactor power
- LT-428 Pressurizer level channel has failed to 0%

Which ONE of the following describes:

1) The Charging Pump in AUTO will ________ Charging line flow;

AND

2) Pressurizer Backup Heaters will _________.

A. 1) raise
   2) de energize

B. 1) lower
   2) de energize

C. 1) raise
   2) energize due to level deviation

D. 1) lower
   2) energize due to level deviation

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with P-10, LT-428 is normally selected for Charging Pump speed control. A failure low will result in Charging Pump speed (in AUTO) rising and Pressurizer Heaters are secured due to Pressurizer low level cutout.

B. INCORRECT. Plausible since this would be the correct response had LT-428 failed high. The second part is correct.

C. INCORRECT. Plausible since the first part is correct and Pressurizer Backup Heaters energize on a ± 5% level deviation from program.

D. INCORRECT. Plausible since this would be the correct response had LT-428 failed high and Pressurizer Backup Heaters energize on a ± 5% level deviation from program.

Technical Reference(s): P-10 (p16-18; Rev 01901)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC03C 1.06 (As available)

Question Source: Bank # 165607
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2011 Harris ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 K1.02	
	Importance Rating	4.1	

Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: Cooling water

RO Question #40

Given the following plant conditions:

- Plant was at 100% power, when a Designed Basis LOCA occurred concurrent with a Loss of Offsite Power
- Operators are in E-0, REACTOR TRIP OR SAFETY INJECTION, and have verified Containment Spray Pumps are running

What is the status of cooling to Containment Spray Pumps?

____________ is providing cooling water to the Containment Spray Pump ___________.

A. CCW; seals

B. RWST water; seals

C. CCW; bearings

D. Service Water; bearings

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since CCW provides the cooling medium for the Containment Spray Pump Seal Heat Exchanger. However, in the conditions given the CCW Pumps are locked out due to a Safety Injection signal and Undervoltage signal on Buses 14 and 16 (EDGs supplying power).

B. CORRECT. During the injection phase of the accident, RWST water will provide the cooling medium for the Containment Spray Pump seals when CCW is not available.

C. INCORRECT. Plausible since other ECCS Pumps utilize Bearing Heat Exchangers to provide cooling to the pump bearings using CCW as the cooling medium.

D. INCORRECT. Plausible since other pumps utilize Bearing Heat Exchangers to provide cooling to the pump bearings using Service Water as the cooling medium.

Technical Reference(s): 33013-1246, Sheet 2 (Rev 14)
(Attach if not previously provided, 33013-1250, Sheet 2 (Rev 51)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2401C 1.04 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165614

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .8
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE 011 EA2.06	
	Importance Rating	3.7*	

Ability to determine or interpret the following as they apply to a Large Break LOCA: That fan in slow speed and dampers are in accident mode during LOCA

RO Question #41

Given the following plant conditions:

- RCS pressure is 100 psig and lowering
- S/G levels are 10% and rising
- CNMT pressure is 35 psig and rising
- CNMT temperature is 282°F and stable

Which ONE of the following procedures will be used FIRST to verify the expected Containment cooling system alignment?

A. E-0, REACTOR TRIP OR SAFETY INJECTION

B. E-1, LOSS OF REACTOR OR SECONDARY COOLANT

C. E-2, FAULTED STEAM GENERATOR ISOLATION

D. ES-1.3, TRANSFER TO COLD LEG RECIRCULATION

Answer: A

Explanation (Optional):

A. CORRECT. In accordance with E-0, step 5 RNO, CNMT Spray is verified initiated if Containment pressure is greater than 28 psig. E-0, step 6 directs the operator to perform ATT-27.0, Attachment Automatic Action Verification. ATT-27.0, step 2 directs the operator to verify CNMT Recirc Fans are running and the charcoal filter dampers are aligned for post-accident conditions. ATT-27.0, step 6.c directs the operator to verify CNMT Recirc Fan Coolers SW outlet valves are OPEN.

B. INCORRECT. Plausible as the candidate will have to perform E-0 prior to transitioning to E-1 and Containment cooling systems are checked in E-0.

C. INCORRECT. Plausible as the candidate will have to perform E-0 prior to transitioning to E-2 and Containment cooling systems are checked in E-0.

D. INCORRECT. Plausible as the candidate will have to perform E-0 prior to transitioning to ES-1.3 and Containment cooling systems are checked in E-0.

Technical Reference(s): E-0 (p6-7; Rev 048)
(Attach if not previously provided, ATT-27.0 (p2 & 4; Rev 00400)
including version/revision number) E-0 Background (p24-25, 73, 79-80; Rev 020)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP00C 1.06 (As available)

Question Source: Bank # 159179
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2001 Wolf Creek ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments: K/A match as E-0, step 6 directs the operator to perform ATT-27.0, ATTACHMENT AUTOMATIC ACTION VERIFICATION, which will verify that the Containment Recirculation Fans and Dampers are in the proper status for SI actuation.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 062 2.4.11	
	Importance Rating	4.0	

Knowledge of abnormal condition procedures.

RO Question #42

Given the following plant conditions:

- The plant is operating at 100% power
- All systems are in a normal configuration
- Service Water Pumps A, C and D are running

The following sequence occurs:

- Computer Point P2160, SERVICE WATER PUMP A & B HEADER, alarms LOW
- PI-2160, SW LOOP A HEADER PRESS, is indicating 45 psig
- PI-2161, SW LOOP B HEADER PRESS, is indicating 56 psig
- NO other Annunciators are currently LIT
- The crew is taking action in an attempt to raise Service Water pressure

Which ONE (1) of the following describes:

1) the procedure that was entered to address this event,

   AND

2) the action required if the condition **cannot** be corrected?

NOTE: AP-SW.1, SERVICE WATER LEAK
AP-SW.2, LOSS OF SERVICE WATER
E-0, REACTOR TRIP OR SAFETY INJECTION
AP-TURB.5, RAPID LOAD REDUCTION

A. 1) AP-SW.1
   2) Trip the reactor and enter E-0

B. 1) AP-SW.1
   2) Initiate a controlled shutdown (AP-TURB.5)

C. 1) AP-SW.2
   2) Trip the reactor and enter E-0

D. 1) AP-SW.2
   2) Initiate a controlled shutdown (AP-TURB.5)

Answer: B

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and AP-SW.1, step 3.a RNO contains guidance to trip the reactor and enter E-0 if three Service Water Pumps are running and either loop header pressure is less than 40 psig.

B. CORRECT. In accordance with AP-SW.1, Symptoms include AR-PPCS-P2160 which directs the operator to either AP-SW.1 or AP-SW.2. AP-SW.1, step 3.b RNO states "IF either SW loop pressure is less than 55 psig with three SW pumps running AND cause can NOT be corrected, THEN initiate a controlled shutdown while continuing with this procedure (Refer to AP-TURB.5, RAPID LOAD REDUCTION)."

C. INCORRECT. AP-SW.2 is incorrect as "the procedure provides the necessary instructions to respond to a loss of SW Pumps." Plausible since AR-PPCS-P2160 directs the operator to either AP-SW.1 or AP-SW.2. AP-SW.2 Symptoms include AR-PPCS-P2160. Additionally, AP-SW.1, step 3.a RNO contains guidance to trip the reactor and enter E-0 if three Service Water Pumps are running and either loop header pressure is less than 40 psig.

D. INCORRECT. AP-SW.2 is incorrect as "the procedure provides the necessary instructions to respond to a loss of SW Pumps." Plausible since AR-PPCS-P2160 directs the operator to either AP-SW.1 or AP-SW.2. AP-SW.2 Symptoms include AR-PPCS-P2160. The second part is correct.

Technical Reference(s): AP-SW.1 (p1-5; Rev 02300)
(Attach if not previously provided, AP-SW.2 (p2; Rev 00801)
including version/revision number) AR-PPCS-P2160 (Rev 00001)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP19C 2.01 (As available)

Question Source: Bank # 158652

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .10

55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068 K6.10	
	Importance Rating	2.5	

Knowledge of the effect of a loss or malfunction on the following will have on the Liquid Radwaste System: Radiation monitors

RO Question #43

Given the following plant conditions:

- A release of the 'A' Monitor Tank is in progress
- R-18, LIQUID WASTE DISPOSAL, Radiation Monitor fails HIGH

Which ONE of the following completes the statement:

The release will be _____(1)_____ terminated and the Monitor Tank Pump will be _____(2)_____.

A. (1) manually
   (2) running

B. (1) manually
   (2) tripped

C. (1) automatically
   (2) running

D. (1) automatically
   (2) tripped

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible if the candidate believes that a malfunction of the radiation monitor will not result in the associated automatic functions occurring.

B. INCORRECT. Plausible if the candidate believes that a malfunction of the radiation monitor will not result in the associated automatic functions occurring. The second part is correct.

C. INCORRECT. Plausible since the first part is correct and the candidate believes that a malfunction of the radiation monitor will not result in the associated automatic functions occurring.

D. CORRECT. In accordance with P-8, Section 2.17 "The following will trip if R-18 goes into alarm: Laundry Pump, Monitor Tank Pump, and Waste Condensate Pumps A & B". Section 2.18 states "If open, RCV-18 will close if R-18 goes into alarm, and will not reopen until the alarm clears." 33013-1271 shows that when RCV-18 closes, the liquid discharge to the Discharge Canal is stopped.

Technical Reference(s): P-8 (p 3; Rev 19)
(Attach if not previously provided, 33013-1268 (Rev 23)
including version/revision number) 33013-1271 (Rev 20)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.07; R3801C 4.01 (As available)

Question Source: Bank # 165626
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K4.07	
	Importance Rating	3.0	

Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: First-out indication

RO Question #44

Given the following plant conditions:

- Plant startup is in progress
- Reactor power is 6%
- PT-449, PRZR PRESS XMTR, has been defeated due to instrument drift
- I&C personnel are calibrating PT-449
- PT-430, PRZR PRESS XMTR, power supply fails de-energizing the associated loop circuitry

Which ONE of the following lists the plant response?

A. No reactor trip occurs

B. Reactor trip with D-3, OTΔT TRIP, as first out annunciator

C. Reactor trip with D-19, PRESSURIZER LO PRESS SI 1750 PSIG, as first out annunciator

D. Reactor trip with D-20, PRESSURIZER LO PRESS TRIP 1873 PSI, as first out annunciator

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since some of the high power trips are bypassed below 8% reactor power and not all four pressure transmitters provide input to every reactor trip function.

B. CORRECT. In accordance with P-1, all four Pressurizer pressure instruments supply an input into a channel of OTΔT protection. Therefore; when PT-430 loses power, the second channel of OTΔT is tripped resulting in a reactor trip. OTΔT is not bypassed by permissive P-7.

C. INCORRECT. Incorrect since PT-449 does not provide an input into Pressurizer low pressure SI. Plausible since two Pressurizer pressure channels have failed which would meet the proper coincidence (2/3) if both supplied an input for the trip.

D. INCORRECT. Incorrect since Pressurizer low pressure trip is bypassed below the P-7 permissive setpoint of 8%. Plausible since the required coincidence is met (2/4) for the Pressurizer low pressure reactor trip setpoint if reactor power was above 8%.

Technical Reference(s): P-1 (p16-23, 40-42, 45; Rev 07201)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC02C 1.06 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165599

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	027 K1.01	
	Importance Rating	3.4*	

Knowledge of the physical connections and/or cause-effect relationships between the CIRS and the following systems: CSS

RO Question #45

Given the following plant conditions:

- Plant is operating at 100% power
- It has been determined that the Sodium Hydroxide (NaOH) tank was inadvertently refilled with a solution concentration less than that required by Technical Specifications

Which ONE of the following correctly answers the statements:

1) NaOH enters the Containment Spray System via ________________ at the suction of the Containment Spray Pumps.
2) If a LOCA were to occur, the reduced NaOH solution would result in increased _____ _________ inside Containment.

A. 1) eductors
   2) Iodine levels

B. 1) eductors
   2) Sump 'B' pH

C. 1) pumps
   2) Iodine levels

D. 1) pumps
   2) Sump 'B' pH

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with Technical Specification Bases 3.6.6, "The NaOH mixture is injected into the CS flowpath via a liquid educator during the injection phase of an

accident." According to UFSAR, Section 6.5.2.1 Iodine can be re-emitted from the sump water for the period of time after a LOCA until sump pH reaches 8.0. With a lower concentration in the NaOH Tank, it will take longer for the sump pH to rise to a value of 8.0 resulting in higher iodine levels in the Containment atmosphere.

B. INCORRECT. Plausible since the first part is correct and the candidate may incorrectly believe that NaOH lowers the Sump 'B' pH levels.

C. INCORRECT. Plausible since there are some chemical addition systems in the plant that utilize pumps to meter and inject chemicals into the associated plant system. The second part is correct.

D. INCORRECT. Plausible since there are some chemical addition systems in the plant that utilize pumps to meter and inject chemicals into the associated plant system. Also, the candidate may incorrectly believe that NaOH lowers the Sump 'B' pH levels.

Technical Reference(s): Technical Specification Bases 3.6.6 (Rev 76)
(Attach if not previously provided, including version/revision number) UFSAR, Section 6.5.2 (Rev 26)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2401C 1.02, 1.09 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 159176

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072 A4.03	
	Importance Rating	3.1	

Ability to manually operate and/or monitor in the control room: Check source for operability demonstration

RO Question #46

Which one of the following describes what occurs when the "Check Source" pushbutton is depressed on R-9, LETDOWN LINE, Area Radiation Monitor?

A. All the alarm and trip functions are bypassed to allow for instrument calibration
B. The source is automatically exposed to the detector for 10 minutes
C. An electronic signal is generated to electrically simulate an actual source
D. The source is exposed to the detector until the radiation monitor goes into ALARM

Answer: **B**

Explanation (Optional):

A. INCORRECT. According to S-14, Area and Process Radiation Monitoring System, Section 6.2.1, "the alarm functions remain active." Plausible if the candidate believes that the alarm functions of the radiation monitor are affected by depressing the check source pushbutton.

B. CORRECT. According to STP-O-17.1, Performance Test of Area, High Range Area, AVT Area and Wide Range Area Radiation Monitors, Section 4.1, "depressing the CHECK SOURCE pushbutton on R-1 through R-9 places the monitor in the check source mode and the monitor will remain in the check source mode until the check source pushbutton is pressed again or 10 minutes has elapsed, at which time the monitor will return to normal operation."

C. INCORRECT. Plausible since according to STP-O-17.5M, Source Check High Range Effluent Monitors RM-12A, RM-14A, R-31, R-32, R-47, R-48, an electronic signal is utilized to test these monitors.

D. INCORRECT. Plausible since the SPING radiation monitors are expected to alarm during the Source Check.

Technical Reference(s): S-14 (p 9; Rev 02601)
(Attach if not previously provided, STP-O-17.1 (p 7; Rev 005)
including version/revision number) STP-O-17.2 (p11; Rev 005)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.09 (As available)

Question Source: Bank # 165612
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 056 AK3.02	
	Importance Rating	4.4	

Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Actions contained in EOP for loss of offsite power

RO Question #47

Given the following plant conditions:

- Plant is at 100% power
- Electric plant is in a 100/0 lineup on Circuit 767
- Circuit 7T is tagged for scheduled maintenance
- 'A' and 'C' Charging Pumps are running
- An electrical fault causes Circuit 767 to de-energize
- 'A' EDG fails to start and cannot be started
- The operating crew has entered AP-ELEC.1, LOSS OF 12A AND/OR 12B BUSSES, and has established 22 gpm Charging Line flow

Which ONE of the following states:

(1) the number of Charging Pumps running prior to entry into AP-ELEC.1;

AND

(2) the reason that 22 gpm Charging Line flow is established?

A. (1) None
(2) To ensure RCP seal cooling

B. (1) One (1)
(2) To ensure RCP seal cooling

C. (1) None
(2) In preparation for restoring Letdown

D. (1) One (1)
(2) In preparation for restoring Letdown

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible as the first part is correct and Charging flow is established in the EOP network to establish normal RCP seal cooling. Since there is no Safety Injection signal present, 'B' CCW Pump will be running providing labyrinth seal cooling for the RCS which is cooling the RCP seals.

B. INCORRECT. Plausible as the candidate may not recognize that 'C' Charging Pump has tripped on undervoltage and Charging flow is established in the EOP network to establish normal RCP seal cooling. Since there is no Safety Injection signal present, 'B' CCW Pump will be running providing labyrinth seal cooling for the RCS which is cooling the RCP seals.

C. CORRECT. With the loss of Circuit 767, all Offsite Power is lost and 'B' EDG starts and re-energizes Buses 16 and 17. 'A' Charging Pump is stopped due to 'A' EDG failing to start and Bus 14 being de-energized; 'C' Charging Pump is lost (UV trip) due to Bus 16 being de-energized momentarily until 'B' EDG re-energizes Bus 16. Therefore, no Charging Pumps are running when the operating crew reaches Step 14 of AP-ELEC.1. In accordance with AP-ELEC.1 Background Document, "In this step charging is restored in preparation for restoring letdown."

D. INCORRECT. Plausible as the candidate may not recognize that 'C' Charging Pump has tripped on undervoltage and the second part is correct.

Technical Reference(s): AP-ELEC.1 (p10; Rev 03203)
(Attach if not previously provided, AP-ELEC.1 Background Document (p10; Rev 00201)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP07C 2.01 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New 159421

Question History: Last NRC Exam ____

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	014 A1.01	
	Importance Rating	2.9*	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: Metroscope reed switch display

RO Question #48

Given the following plant conditions:

- The plant was at full power with step counters reading:
  - Shutdown Bank: 223 steps
  - Control Bank A: 224 steps
  - Control Bank B: 225 steps
  - Control Bank C: 225 steps
  - Control Bank D: 217 steps
- A reactor trip occurred
- Following the trip Rod C-7 in Control Bank D indicated 212 steps on MRPI
- All other rods indicated 0 steps on MRPI

Which ONE of the following alarms will be present due to Rod C-7 after reactor trip?

A. MCB alarm C-14, ROD BOTTOM
B. MCB alarm C-30, ROD CONTROL URGENT FAILURE ROD STOP
C. MRPI alarm ROD OFF TOP
D. MRPI alarm ROD DEVIATION

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since this annunciator will be in alarm due to all other rods exceeding 12 steps from their associated bank position (step counter). Incorrect because rod C-7 is within 12 steps of its bank position.

B. INCORRECT. Plausible since this annunciator will be in alarm due to the reactor trip and regulation failure to all control rods, not just C-7.

C. INCORRECT. Plausible since this alarm is in due to the Shutdown Bank rods being < 224 steps. Incorrect because rod C-7 is NOT in the Shutdown Bank.

D. CORRECT. The MRPI Rod Deviation alarm comes in whenever two rods' positions within a bank differ by ≥ 24 steps. All rods within Control Bank D are within 24 steps of each other with the exception of rod C-7.

Technical Reference(s): R3101C PPT/LP (slide 89; Rev 20)
(Attach if not previously provided, AR-C-5 (Rev 00901)
including version/revision number) AR-C-14 (Rev 00900)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3101C 1.07 (As available)

Question Source: Bank # 159175
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2012 Ginna ILT
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE 060 AK3.03	
	Importance Rating	3.8	

Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste: Actions contained in EOP for accidental gaseous-waste release

RO Question #49

Given the following plant conditions:

- Plant has experienced an ATWS
- The operating crew is performing FR-S.1, RESPONSE TO REACTOR RESTART/ATWS
- The step to initiate a Containment Ventilation Isolation is being performed
- Annunciator A-25, CNMT VENTILATION ISOLATION, is extinguished

1. How is a Containment Ventilation Isolation signal initiated?
2. What is the reason for initiating a Containment Ventilation Isolation signal?

A. 1. Tripping the Containment Sample Pump from the Control Room

2. Non-essential Containment Ventilation penetrations are isolated to prevent the potential release of radiation.

B. 1. Momentarily de-energizing R-11

2. Non-essential Containment Ventilation penetrations are isolated to prevent the potential release of radiation.

C. 1. Momentarily de-energizing R-11

2. Aligns the Charcoal Filters to 'A' and 'C' Recirc Fans.

D. 1. Tripping the Containment Sample Pump from the Control Room

2. Aligns the Charcoal Filters to 'A' and 'C' Recirc Fans.

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since the candidate could believe that tripping the CNMT Sample Pump could reset R-11 or R-12 which would initiate a CVI signal. Also, the second part is correct.

B. CORRECT. In accordance with FR-S.1, step 6 RNO "momentarily deenergize CNMT particulate monitor, R-11 to actuate CVI" is performed if CVI annunciator A-25 is NOT LIT. According to FR-S.1 Background Document this step is performed "To ensure non-essential containment ventilation penetrations are isolated."

C. INCORRECT. Plausible since the first part is correct and if the candidate believes that the CVI signal directs opening of Recirc Fan Charcoal Filter Dampers.

D. INCORRECT. Plausible since the candidate could believe that tripping the CNMT Sample Pump could reset R-11 or R-12 which would initiate a CVI signal. Also, the candidate may believe that the CVI signal directs opening of Recirc Fan Charcoal Filter Dampers.

Technical Reference(s): FR-S.1 (p8; Rev 021)
(Attach if not previously provided, FR-S.1 Background Document (p45; Rev 010)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2201C 1.07, RFRS1C 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 165609

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43 ______

Comments:

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

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

RO Question #50

Given the following plant conditions:

- Reactor Trip Breaker testing is in progress on Train 'A'
- 'A' Reactor Trip Breaker is OPEN
- 'A' Reactor Trip Bypass Breaker is CLOSED
- A transient occurs initiating an AUTOMATIC Reactor Trip signal
- The Reactor does **NOT** trip from the AUTOMATIC signal

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

Reactor Trip....

A. Bypass Breaker 'A' Shunt Trip Coil failed to energize
B. Bypass Breaker 'A' Undervoltage Trip Coil failed to de-energize
C. Breaker 'B' Shunt Trip Coil failed to de-energize
D. Breaker 'B' Undervoltage Trip Coil failed to energize

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible because the Shunt Trip Coil is designed to energize to trip OPEN the Bypass Breaker; however, the Bypass Breaker Shunt Trip Coil is only energized on a MANUAL reactor trip signal.

B. CORRECT. In accordance with UFSAR, Section 7.2.2.1.5, the Bypass Breaker Undervoltage Trip Coil is de-energized when an AUTOMATIC reactor trip signal is generated.

C. INCORRECT. Plausible because the Reactor Trip Breaker Shunt Trip Coil would change state when an AUTOMATIC reactor trip signal is generated; however, the Shunt Trip Coil is energized to TRIP the Reactor Trip Breaker.

D. INCORRECT. Plausible because the Reactor Trip Breaker Undervoltage Trip Coil would change state when an AUTOMATIC reactor trip signal is generated; however, the Undervoltage Trip Coil is de-energized to TRIP the Reactor Trip Breaker.

Technical Reference(s): UFSAR Section 7.2.2.1.5 (Rev 26)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3501C 1.07 (As available)

Question Source: Bank # 159420
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2012 Ginna ILT RO Retake

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 K1.04	
	Importance Rating	3.4	

Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: S/Gs water level control system

RO Question #51

Given the following plant conditions:

- Plant is operating at 100% power
- FT-466, STEAM GEN A-1 FW FLOW, failed high and was removed from service in accordance with ER-INST.1, REACTOR PROTECTION BISTABLE DEFEAT AFTER INSTRUMENTATION LOOP FAILURE two hours ago
- FT-500, STEAM GEN A-3 FW FLOW, now fails low

Which ONE of the following describes the response, if any, of the Feedwater Control System?

	'A' Feed Reg Valve	**'B' Feed Reg Valve**
A.	Shifts to MANUAL	Shifts to MANUAL
B.	Remains in AUTOMATIC	Remains in AUTOMATIC
C.	Shifts to MANUAL	Remains in AUTOMATIC
D.	Remains in AUTOMATIC	Shifts to MANUAL

Answer: A

Explanation (Optional):

A. CORRECT. Failure of 2 or 3 inputs of Feedwater Flow in EITHER loop will result in BOTH loops shift to MANUAL control according to AR-G-20, ADFCS System Transfer to Manual Control.

B. INCORRECT. Plausible because failure of both inputs of Main Feedwater Header pressure does not result in any change in Feedwater Regulating Valve operation (Manual or Automatic).

C. INCORRECT. Plausible because failure of two inputs in either loop of narrow range S/G water level will result in the affected feedwater control valve shifting to MANUAL.

D. INCORRECT. Plausible because failure of two inputs in either loop of narrow range S/G water level will result in the affected feedwater control valve shifting to MANUAL.

Technical Reference(s): AR-G-20 (Rev 00700)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4401C 1.10 (As available)

Question Source: Bank # 159167
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.6	
	Importance Rating	3.0	

Knowledge of the process for making changes to procedures.

RO Question #52

A Temporary Procedure Change has been prepared for CPI-TAVG-401, CALIBRATION OF TAVG LOOP 401, to correct a cloning error which occurred from the previous revision of the procedure.

Which ONE of the following states the approval signatures required **before** the procedure can be used?

NOTE: **SRO – Senior Reactor Operator**
SQR – Station Qualified Reviewer
SFAM – Site Functional Area Manager

A. SRO only
B. SRO and SQR only
C. SRO and SFAM only
D. SRO, SQR, and SFAM

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since SRO approval only is needed for some types of changes.

B. CORRECT. In accordance with AD-AA-101, step 4.2.2.1 "To allow use, obtain an approval signature from a SQR and an approval signature from a qualified SRO on the approval form AD-AA-101-F-10."

C. INCORRECT. Plausible since the SRO approval is required prior to procedure use and the SFAM approval is required within 14 days in accordance with AD-AA-101, step 4.2.2.2 and AD-AA-101-F-01.

D. INCORRECT. Plausible since the SRO and SQR approval is required prior to procedure use and a Site Functional Area Manager will have to perform a Functional Area review as required within 14 days in accordance with AD-AA-101, step 4.2.2.2 and AD-AA-101-F-01.

Technical Reference(s): AD-AA-101 (p13; Rev 28)
(Attach if not previously provided, AD-AA-101-F-10 (Rev 1)
including version/revision number) AD-AA-101-F-01 (Rev 7)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD07C 3.02 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 158413

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.7	
	Importance Rating	3.5	

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

RO Question #53

An Operator has been assigned a routine task that requires entry into a Locked High Radiation Area (radiation levels > 1000 mrem/hr).

Which ONE of the following below completes the following statement in accordance with RP-AA-403, ADMINISTRATION OF THE RADIATION WORK PERMIT PROGRAM?

The Operator's entry into the Locked High Radiation Area will require a ________(1)____ RWP, and the key to this area will be issued to the (2)____.

A. (1) General or Specific
   (2) RP Technician

B. (1) General or Specific
   (2) operator

C. (1) Specific only
   (2) RP Technician

D. (1) Specific only
   (2) operator

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with RP-AA-403, Administration of the Radiation Work Permit Program, step 4.4.1 "General RWPs should not allow access into a high radiation or locked high radiation area except for Operations and Radiation Protection." According to RP-AA-460, Controls for High and Locked High Radiation Areas, step 4.4.4.1 "Issue LHRA Key to RP Personnel using Attachment 10 or CGE."

B. INCORRECT. Plausible since the first part is correct and an operator has the authority to be issued keys for various other instances and area locks.

C. INCORRECT. Plausible since only Operations and RP personnel can enter a LHRA under a General RWP, all other work groups require a Specific RWP. Also plausible as the second part is correct.

D. INCORRECT. Plausible since only Operations and RP personnel can enter a LHRA under a General RWP, all other work groups require a Specific RWP. Also plausible since an operator has the authority to be issued keys for various other instances and area locks.

Technical Reference(s): RP-AA-403 (p7; Rev 9)
(Attach if not previously provided, RP-AA-460 (p12; Rev 29)
including version/revision number) RWP 16-00102 Task 2 (Rev 00)

Proposed references to be provided to applicants during examination: None

Learning Objective: RSE00O 1.02 (As available)

Question Source: Bank # 158416
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2014 Surry ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .12
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE 067 AA1.05	
	Importance Rating	3.0	

Ability to operate and / or monitor the following as they apply to the Plant Fire on Site: Plant and control room ventilation systems

RO Question #54

Given the following plant conditions:

- Control Room has indications of a fire at S02 Auxiliary Building Charcoal Filter Unit G
- Primary Equipment Operator reports that a fire in the 1G Fan Unit Charcoal Filters has occurred
- The CO reports that S02 fire detector is LIT and automatic system suppression has NOT actuated

1) What is the status of 1G Fan?
2) Where can suppression for the Charcoal Filter Unit be actuated from?

A. 1) tripped
   2) Local Pull Box Station ONLY
B. 1) running
   2) Local Pull Box Station ONLY
C. 1) tripped
   2) Auxiliary Benchboard Pull Box or Local Pull Box Station
D. 1) running
   2) Auxiliary Benchboard Pull Box or Local Pull Box Station

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible because G Fan will trip on fire indication and manual suppression can be initiated from the local pull box. However, the ability to actuate fire suppression also exists on the Auxiliary Benchboard in the Control Room.

B. INCORRECT. Plausible if candidate does not know that the fan trips if the heat detector alarms and the majority of the Auxiliary Building Fans do not trip on fire detection. Also, manual suppression can be initiated from the local pull box; however, the ability to actuate fire suppression also exists on the Auxiliary Benchboard in the Control Room.

C. CORRECT. In accordance with STP-O-13.4.14, Section 6.6, 1G Fan trips when the detectors sense a heat source. Section 6.3 contains the testing sequence for the Local Manual Control Station (Pull Box) and the Remote Manual Control Station at the Fire Panel Bench Station (Auxiliary Benchboard Pull Box).

D. INCORRECT. Plausible if candidate does not know that the fan trips if the heat detector alarms and the majority of the Auxiliary Building Fans do not trip on fire detection. Second part is correct.

Technical Reference(s): STP-O-13.4.14 (p16-23, 27-33; Rev 00102)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2201C 1.07 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165622

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 K4.02	
	Importance Rating	3.0	

Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: Prevention of uncovering PZR heaters

RO Question #55

Given the following plant conditions:

- Plant operating is at 100% power
- 'A' and 'C' Charging Pumps are running
- Bus 14 trips and is reenergized from 'A' D/G
- HCO reports that Letdown has isolated
- **No operator actions** have been taken place

Which ONE of the following correctly lists the status of the Pressurizer Heaters following the transient?

	Proportional Heaters	Backup Heaters
A.	De-energized	Energized
B.	Energized	Energized
C.	Energized	De-energized
D.	De-energized	De-energized

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since if the Proportional Heaters trip due to undervoltage on Bus 14 the Backup Heaters would energize to control RCS pressure.

B. INCORRECT. Plausible if candidate believes the Diesel energizing Bus 14 will cause Proportional Heaters to re-energize, Backup Heaters have a feature to reduce impact of Pressurizer level insurge.

C. INCORRECT. Plausible since the Proportional Heaters would trip on undervoltage on Bus 14 to ensure that the D/G loading would be less on bus re-energization. Also, the candidate may believe that since Bus 16 is not affected, neither are the Backup Heaters.

D. CORRECT. Loss of LT-427 will cause low level cutoff and heater trip to protect heaters and Pressurizer from damage; thereby, ensuring RCS integrity.

Technical Reference(s): P-10 (p48-49; Rev 01901)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1901C 1.07 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New 158669

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ______

Comments: K/A match because the candidate has to know that the reason for both PRZR Proportional and Backup Heaters being tripped is from loss of level transmitter 427 due to loss of power. Setpoint is reached to trip the heaters on indicated low level to prevent the heaters being energized on an indicated low Pressurizer level condition.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 A1.01	
	Importance Rating	3.2	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels

RO Question #56

The following annunciators alarm simultaneously:

- E-16, RMS PROCESS MONITOR HIGH ACTIVITY
- A-25, CONTAINMENT VENTILATION ISOLATION

Which ONE of the following Radiation Monitors has exceeded its Alarm setpoint?

A. R-10A, CONTAINMENT VENTILATION IODINE
B. R-12, CONTAINMENT VENTILATION GAS
C. R-12A, CONTAINMENT VENT HIGH RANGE EFFLUENT SPING
D. R-29, CONTAINMENT HIGH RANGE

Answer: B

Explanation (Optional):

A. INCORRECT. Plausible since R-10A monitors Containment ventilation. The candidate may believe that any Containment ventilation radiation monitor alarming will actuate Containment Ventilation Isolation.

B. CORRECT. In accordance with P-9, Section 6.2.4, "The high alarm setpoint actuates a containment vent isolation." According to AR-A-25, source of Annunciator A-25 is R-11/R-12 high radiation.

C. INCORRECT. Plausible since R-12A monitors Containment ventilation. The candidate may believe that any Containment ventilation radiation monitor alarming will actuate Containment Ventilation Isolation.

D. INCORRECT. Plausible since R-29 monitors the Containment environment and is used for EAL classification. The candidate may believe that R-29 alarming results in Containment Ventilation Isolation to prevent a release to the environment.

Technical Reference(s): P-9 (p8,11-14; Rev 10000)
(Attach if not previously provided, AR-A-25 (Rev 01202)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.07 (As available)

Question Source: Bank # 165623
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43

Comments: K/A match because monitoring R-11 or R-12 for radiation levels and ensuring it performs it's function to give a CNMT Ventilation Isolation signal when it alarms is to ensure that ODCM and Technical Specification assumptions are met. Knowing that the listed MCB alarm should cause a plant response that will need to be verified is an operator responsibility with operating the PRM system.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 065 AK3.08	
	Importance Rating	3.7	

Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Actions contained in EOP for loss of instrument air

RO Question #57

Given the following plant conditions:

- A steam line break occurred in Containment
- Auxiliary Feedwater cannot be started
- The operating crew is performing FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK
- Bleed and Feed criteria has been met
- Containment pressure is 31 psig
- SI has NOT been reset

Which ONE of the following completes the statements below?

1) Opening of the PORVs to establish feed and bleed will use ____________.
2) Containment Isolation is reset in FR-H.1 to ___________________.

A. 1) Instrument Air
   2) establish Letdown and Seal Return

B. 1) Instrument Air
   2) maintain PORVs open for an extended period of time

C. 1) Nitrogen
   2) establish Letdown and Seal Return

D. 1) Nitrogen
   2) maintain PORVs open for an extended period of time

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since Bleed and Feed opens PORVs and many EOPs restore Instrument Air to open PORVs. Additionally, plausible because most emergency procedures attempt to re-establish Normal Letdown and Seal Return.

B. INCORRECT. Plausible since Bleed and Feed opens PORVs and many EOPs restore Instrument Air to open PORVs. Also, the second part is correct.

C. INCORRECT. Plausible since the first part is correct. Additionally, plausible because most emergency procedures attempt to re-establish Normal Letdown and Seal Return.

D. CORRECT. In accordance with FR-H.1, step 15, Establish RCS Bleed path, the PORV control switches are placed in OPEN and the RCS Overpressure Protection System is aligned to OPEN both PORVs using nitrogen. According to the Background Document "The PORVs are initially opened using nitrogen because a previous step actuated SI and CI, resulting in the loss of Instrument Air to CNMT." Per the Background Document for "Establish IA to CNMT", it states "Instrument Air is needed to maintain the PRZR PORVs in an open position for an extended period of time."

Technical Reference(s): FR-H.1 (p14-15, 18-20; Rev 04100)
(Attach if not previously provided, FR-H.1 Background Document (p58, 64; Rev 00800)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRH1C 1.04 (As available)

Question Source: Bank # 158653
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2011 Farley ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K2.04	
	Importance Rating	3.6	

Knowledge of the bus power supplies to the following: ESFAS-operated valves

RO Question #58

Given the following plant conditions:

- A DBLOCA has occurred concurrent with loss of Offsite Power
- Operators have completed Cold Leg Recirc alignment and are preparing to establish High Head Recirc when 'B' EDG trips

Which ONE of the following completes the statements below?

1) What valves are required to be manipulated to establish the alignment from the MCB?
2) The High Head Recirc alignment _________ be established from the MCB.

NOTE: MOV-857A/B/C, RHR PUMP DISCH TO SI PUMP SUCT
MOV-896A/B, RWST OUTLET
MOV-897/898, SI RECIRC

A. 1) Close MOV-896B and MOV-898; open MOV-857B
2) CAN

B. 1) Close MOV-896B and MOV-898; open MOV-857B
2) CAN NOT

C. 1) Close MOV-896A and MOV-897; open MOV-857A and MOV-857C
2) CAN

D. 1) Close MOV-896A and MOV-897; open MOV-857A and MOV-857C
2) CAN NOT

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and the candidate may confuse which electrical safeguards train is supplying power for the MOVs.

B. INCORRECT. Plausible since the candidate may believe that since two SI Pumps are started in ES-1.3 for High Head Recirc that two RHR Pumps are necessary and therefore cannot be established from the MCB. Also, the candidate may confuse which electrical safeguards train is supplying power for the MOVs.

C. CORRECT. In accordance with ES-1.3, step 11 RNO, at least one valve must be closed (either MOV-896A or -896B; AND either MOV-897 or -898) and step 13, open MOV-857A and MOV-857C since 'A' RHR Pump will be running. According to P-12, only the MOVs powered from Bus 14 (MCC 'C' and MCC 'L') will have power and therefore be able to be operated from the MCB.

D. INCORRECT. Plausible since the candidate may believe that since two SI Pumps are started in ES-1.3 for High Head Recirc that two RHR Pumps are necessary and therefore cannot be established from the MCB. The second part is correct.

Technical Reference(s): ES-1.3 (p11-13; Rev 04600)
(Attach if not previously provided, ATT-14.5 (Rev 3)
including version/revision number) P-12 (p35, 37, 46-47, 49, 51, 61; Rev 024)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2701C 1.05, 1.08 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New 165598

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments: K/A match because the candidate must have the knowledge that the ability to establish high head Safety Injection will require the interlocks for MOV-857s being met. To meet the interlocks, certain valves need to be CLOSED and the question deals with meeting the interlocks by knowing the power supplies for the valves that are required to be CLOSED.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.18	
	Importance Rating	2.6	

Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

RO Question #59

Given the following plant conditions

- Mode 6, refueling is in progress

Which ONE of the following lists includes Key Safety Functions (Critical Categories) in accordance with IP-OUT-2, GINNA SITE SPECIFIC OUTAGE RISK MANAGEMENT?

A. Subcriticality, Core Cooling, Heat Sink, Integrity, Inventory

B. Electrical Power Available, Reactivity, Core Cooling, Containment, Fire Water System

C. Decay Heat Removal, Integrity, Containment, Residual Heat Removal, Auxiliary Feedwater

D. Auxiliary Feedwater, Residual Heat Removal, Reactivity, Spent Fuel Pool Cooling, Nuclear Instrumentation

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since the categories listed are the Critical Safety Function Status Trees.

B. CORRECT. In accordance with IP-OUT-2, Attachment A, the following are the Key Safety Functions (critical categories) during a shutdown condition: power available, Spent Fuel Pool cooling, Reactivity, core cooling, Containment, inventory, and Fire Water system.

C. INCORRECT. Plausible since the categories listed are Key Safety Functions (critical categories) during shutdown or subcategories of those.

D. INCORRECT. Plausible since the categories listed are the key safety systems while determining on-line risk.

Technical Reference(s): IP-OUT-2 (p19-25; Rev 02200)
(Attach if not previously provided, WC-AA-104 (p26; Rev 24)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD73C 1.03 (As available)

Question Source: Bank # 158414
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2012 Harris ILT
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments: K/A match because the question deals with risk assessment priorities during shutdown operations. Process has a procedure for the key areas for outage risk assessed and scored. Knowledge of these critical safety functions is part of outage risk assessment.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 008 AK2.01	
	Importance Rating	2.7*	

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

RO Question #60

Given the following plant conditions:

- Plant was initially at 100% power
- An automatic reactor trip and Safety Injection (SI) occurred as a result of lowering RCS pressure
- Pressurizer pressure is at saturation for the current RCS temperature
- Pressurizer level was rising prior to and following the SI
- Charging Pump flow was lowering prior to the SI

Which ONE of the following accidents would result in these conditions?

A. Steamline break
B. 3-inch cold leg break
C. Double-ended hot leg break
D. Stuck open Pressurizer safety valve

Answer: D

Explanation (Optional):
A. INCORRECT. Reactor power, Pressurizer pressure, Pressurizer level, initial RCS temperature, and initial Charging Pump flow parameters are not consistent with a steam-line break. Plausible since a steam-line break would result in a reactor trip and safety injection; and initial Pressurizer pressure drop and RCS temperature (post Main Steam Isolation Valve closure) parameters to trend as given.

B. INCORRECT. Pressurizer pressure, Pressurizer level, RCS temperature, and Charging Pump flow parameters are not consistent with a 3-inch cold leg break. Plausible since a 3-inch cold leg break would result in a reactor trip and safety injection; and Reactor power,

initial Pressurizer pressure drop, and initial RCS temperature parameters to trend as given.

C. INCORRECT. Pressurizer pressure, Pressurizer level, RCS temperature, and Charging Pump flow parameters are not consistent with a double-ended hot leg break. Plausible since a double-ended hot leg break would result in a reactor trip and safety injection; and Reactor power, initial Pressurizer pressure drop, and initial RCS temperature parameters to trend as given.

D. CORRECT. The given parameters are indicative of a Pressurizer Vapor Space accident.

Technical Reference(s): UFSAR, Section 15.6.1 (Rev 26)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RTA04C 0.01 (As available)

Question Source: Bank # 158409
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE W/E12 EK1.3	
	Importance Rating	3.4	

Knowledge of the operational implications of the following concepts as they apply to the (Uncontrolled Depressurization of all Steam Generators): Annunciators and conditions indicating signals, and remedial actions associated with the (Uncontrolled Depressurization of all Steam Generators)

RO Question #61

Given the following plant conditions:

- The unit has experienced a steam break on the main turbine inlet steam piping
- All attempts to close either MSIV have failed
- Operators have transitioned to ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF BOTH STEAM GENERATORS
- RCS cooldown rate is 120°F/hr
- RCS pressure is 950 psig and lowering
- SI flow is 290 gpm
- The RCS cooldown is NOT being controlled

Which ONE of the following identifies the NEXT action that must be performed in accordance with ECA-2.1?

A. Trip both RCPs
B. Stop both RHR Pumps
C. Reduce AFW flow to 50 gpm to each S/G
D. Locally CLOSE both Feedwater Regulating Valves

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible since RCPs are tripped early in ECA-2.1 if RCP trip criteria are met. The given plant conditions would require tripping of RCPs if either S/G were intact and not depressurizing.

B. INCORRECT. Plausible since ECA-2.1, step 7 directs the operator to secure RHR Pumps running in injection mode with RCS pressure greater than 300 psig. However, AFW flow is reduced prior to this step being performed.

C. CORRECT. In accordance with ECA-2.1, step 2 RNO, if RCS cooldown rate is greater than 100°F/hr in the RCS cold legs the operator is directed to lower feed flow to 50 gpm to each S/G.

D. INCORRECT. Plausible since ECA-2.1, step 1 attempts to isolate the S/Gs to establish secondary pressure boundaries. Step 1 RNO directs the operator to dispatch an EO to locally isolate flowpaths.

Technical Reference(s): ECA-2.1 (p1-7; Rev 03601)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REC21C 2.01 (As available)

Question Source: Bank # 158657
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .10
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EPE W/E14 EK2.1	
	Importance Rating	3.4	

Knowledge of the interrelations between the (High Containment Pressure) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

RO Question #62

Given the following plant conditions:

- A large break LOCA has occurred in Containment
- Both RHR Pumps have failed
- All other ECCS pumps are running as designed
- RWST level is 34% and lowering
- The crew has transitioned from E-1, LOSS OF REACTOR OR SECONDARY COOLANT, to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION
- The crew has identified a valid ORANGE path on the Containment Status Tree which directs entry into FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE

Which ONE of the following completes the statement below?

The crew will use ____(1)____ guidance to operate Containment Spray Pumps based on ____________(2)__________.

A. (1) ECA-1.1
   (2) Containment pressure and RWST level only

B. (1) FR-Z.1
   (2) Containment pressure, RWST level, and CNMT Recirc Fans available

C. (1) FR-Z.1
   (2) Containment pressure and RWST level only

D. (1) ECA-1.1
   (2) Containment pressure, RWST level, and CNMT Recirc Fans available

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible because ECA-1.1 does provide the overriding guidance for Containment Spray Pump operation and the parameters given are 2 of 3 used to determine the required number of CNMT Spray Pumps. Incorrect because the number of running Containment Spray Pumps is dependent upon a combination of RWST level, Recirc Fans available, and Containment pressure.

B. INCORRECT. The number of running Containment Spray Pumps is dictated by ECA-1.1 in this case. Plausible because given that an ORANGE path on Containment Status Tree exists, the candidate may believe that the guidance in FR-Z.1 takes precedence. Also, the second part is correct.

C. INCORRECT. The number of running Containment Spray Pumps is dictated by ECA-1.1 in this case. Plausible because given that an ORANGE path on Containment Status Tree exists, the candidate may believe that the guidance in FR-Z.1 takes precedence. Additionally, the parameters given are 2 of 3 used to determine the required number of CNMT Spray Pumps.

D. CORRECT. An ORANGE path on the Containment Status Tree would become the higher priority according to the rules of usage; however, there is a CAUTION prior to Step 2 of FR-Z.1 that directs the crew to operate Containment Spray as directed by ECA-1.1. In ECA-1.1 the number of Containment Spray Pumps running is changed based on Containment pressure, RWST level, and number of running CNMT Recirc Fans.

Technical Reference(s): FR-Z.1 (p 4; Rev 01200)
(Attach if not previously provided, ECA-1.1 (p 6; Rev 02801)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRZ1C 1.02 (As available)

Question Source: Bank # 158668
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2009 Seabrook ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 K1.07	
	Importance Rating	3.9	

Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: AFW

RO Question #63

Given the following initial plant conditions:

- Plant is at 100% power when a steam break occurs in the Turbine Building
- Automatic Safety Injection occurred
- 'A' MSIV immediately closed on the SI signal
- 'B' MSIV did not immediately close and was manually closed by the Control Operator two minutes later

Current plant conditions are as follows:

- T_{AVG} is 525°F and rising
- The following MDAFW Pump flows are reported during E-0, REACTOR TRIP OR SAFETY INJECTION, performance:
  - 'A' MDAFW Pump - 210 gpm
  - 'B' MDAFW Pump - 180 gpm

1) Are MDAFW Pumps operating as expected?
2) What will MDAFW Pump flows do if **no further operator actions** are taken upon completion of E-0 Immediate Actions?

A. 1) Yes
   2) Both MDAFW Pump flows will remain stable

B. 1) Yes
   2) Both MDAFW Pump flows will lower and stabilize at a lower value

C. 1) No
   2) Both MDAFW Pump flows will remain stable

D. 1) No
   2) Both MDAFW Pump flows will lower and stabilize at a lower value

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible because first part is correct. Second part is incorrect as S/G pressure will rise with T_{AVG} which will go to ARV setpoint in this plant condition, 1050 psig. This will cause lower Auxiliary Feedwater flow. The second part is plausible since AFW flows are relatively stable post-trip with no extenuating events.

B. CORRECT. Following the reactor trip and safety injection, both MDAFW Pumps started and their respective discharge valve fully opened and then throttled back to approximately 215 gpm. Since 'B' MSIV was not closed for 2 minutes, 'B' S/G pressure was much lower than 'A' S/G pressure when the MDAFW Pumps started and the discharge valves automatically throttled. Once 'B' MSIV was closed, the pressure in 'B' S/G started to rise resulting in 'B' MDAFW Pump flow to lower (with no operator actions). As RCS T_{AVG} continues to rise back to no-load value, the pressure in the S/Gs will rise and stabilize at the ARV setpoints (1050 psig). With no operator actions, MDAFW Pump flow will lower and stabilize at a new lower value due to pumping to a higher pressure S/G.

C. INCORRECT. Plausible if candidate does not know that MDAFW Pump throttle back is a one-time event following pump start and will throttle back to current Steam Generator pressure. Second part is incorrect as S/G pressure will rise with T_{AVG} which will go to ARV setpoint in this plant condition, 1050 psig. This will cause lower Auxiliary Feedwater flow. The second part is plausible since AFW flows are relatively stable post-trip with no extenuating events.

D. INCORRECT. Plausible if candidate does not know that MDAFW Pump throttle back is a one-time event following pump start and will throttle back to current Steam Generator pressure. Second part is correct.

Technical Reference(s):	UFSAR 10.5 (Rev 26)
(Attach if not previously provided,	10905-0659 (Rev 5)
including version/revision number)	10905-0660 (Rev 4)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4201C 1.04, 1.07 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	165621	

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	.4
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.13	
	Importance Rating	4.0	

Knowledge of crew roles and responsibilities during EOP usage.

RO Question #64

In the Emergency Operating Procedures, procedural steps marked with an asterisk (*) are considered:

List of eq

A. list of equipment that can be operated in any order.

B. list of equipment that must be operated in a specific order.

C. immediate action steps that must be able to be performed from memory.

D. continuous actions steps that can be returned to if plant conditions warrant.

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since sub-steps identified by either open or closed bullets may be performed in any order of preference.

B. INCORRECT. Plausible since procedure use and adherence requires that all procedural steps be performed in the order listed.

C. INCORRECT. Plausible since A-502.1 states "Immediate operator action steps shall be identified by a circle surrounding the step number" and A-503.1 states "Immediate Action steps in the EOP/AOP sets shall be memorized to the extent that operators can accomplish the intent of each immediate action step."

D. CORRECT. In accordance with A-503.1, "Those steps identified by an asterisk preceding the step number are designated as continuous actions steps."

Technical Reference(s):	A-503.1 (p25-26; Rev 04600)
(Attach if not previously provided,	A-502.1 (p14; Rev 021)
including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective:	REP50C 1.14		(As available)
Question Source:	Bank #	158641	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2013 ANO ILT	

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

Comments: K/A match because the crew must know the meaning of the different types of steps. Continuous Action is assigned to the crew and, in the event that the condition listed has occurred, the crew responsibility is to return to the appropriate step and perform listed actions.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE W/E11 EA1.1	
	Importance Rating	3.9	

Ability to operate and / or monitor the following as they apply to the (Loss of Emergency Coolant Recirculation): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

RO Question #65

Given the following plant conditions:

- Reactor trip and safety injection have occurred
- The crew is performing actions of E-1, LOSS OF REACTOR OR SECONDARY COOLANT
- Cold leg recirculation capability cannot be verified and the crew transitions to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION
- Wide range RCS pressure is 200 psig and slowly lowering
- RCS temperature is 440°F and lowering
- RWST level is 15% and lowering

Which ONE of the following describes the action related to operation of ECCS equipment **AND** the reason for that action?

Reduce ECCS flow to establish . . .

A. one SI Pump running to delay RWST depletion.
B. one SI Pump running to prevent an ORANGE path on RCS Integrity CSFST.
C. one RHR Pump running to delay RWST depletion.
D. one RHR Pump running to prevent an ORANGE path on RCS Integrity CSFST.

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with ECA-1.1 Background Document for step 28 "Since the RWST is now empty at this time, all pumps still taking suction from the RWST are stopped, with the exception that one SI pump is left running to provide some injection flow." The purpose of this step is to minimize RWST outflow.

B. INCORRECT. Plausible since the first part is correct and one of the major objectives of ECA-1.1 is "to minimize thermal stresses in the reactor vessel and remain within the ORANGE and RED priority limits of the Integrity Status Tree" by limiting the cooldown rate to 100°F/hr.

C. INCORRECT. Plausible since the candidate may recognize that all but one ECCS Pump are stopped, and since the given RCS pressure is < 300 psig and RHR Pumps are injecting, may believe that an RHR Pump remains running. The second part is correct.

D. INCORRECT. Plausible since the candidate may recognize that all but one ECCS Pump are stopped, and since the given RCS pressure is < 300 psig and RHR Pumps are injecting, may believe that an RHR Pump remains running. The second part is plausible since one of the major objectives of ECA-1.1 is "to minimize thermal stresses in the reactor vessel and remain within the ORANGE and RED priority limits of the Integrity Status Tree" by limiting the cooldown rate to 100°F/hr.

Technical Reference(s): ECA-1.1 (p25-27; Rev 02801)
(Attach if not previously provided, ECA-1.1 Background Document (p47; Rev 01301)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REC11C 2.01 (As available)

Question Source: Bank # 165595
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 K1.02	
	Importance Rating	2.7*	

Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Service air

RO Question #66

Given the following plant conditions:

- Instrument and Service Air headers are cross-connected
- 'C' Instrument Air Compressor is running
- A leak has occurred on the Service Air header in the Turbine Building
- Instrument and Service Air headers pressures are 93 psig and lowering

Which ONE of the following completes the statements below?

1) V-7000, SERVICE AIR/INSTRUMENT AIR CROSS CONNECT LOW PRESSURE AUTOMATIC ISOLATION VALVE, current position is ________.
2) AOV-5251, SERVICE AIR CROSSTIE AIR OPERATED VALVE TO INSTRUMENT AIR SYSTEM, will automatically operate at ______________ Instrument Air header pressure.

A. 1) OPEN
   2) 85 psig

B. OPEN
   2) 90 psig

C. 1) CLOSED
   2) 85 psig

D. 1) CLOSED
   2) 90 psig

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible if the candidate believes that V-7000 closes at 90 psig, the setpoint for AOV-5251. The second part is plausible since this is the setpoint for the Service Air header low pressure alarm.

B. INCORRECT. Plausible if the candidate believes that V-7000 closes at 90 psig, the setpoint for AOV-5251. The second part is correct.

C. INCORRECT. Plausible since the first part is correct and the second part is the setpoint for the Service Air header low pressure alarm.

D. CORRECT. In accordance with UFSAR Section 9.3.1.1, "A pressure regulator valve will stop air flow to the service air system if pressure on the service air side drops below 100 psig." Additionally, "A cross-tie between service air and instrument air allows the service air system to supply the instrument air header if instrument air pressure drops below 90 psig."

Technical Reference(s): UFSAR, Section 9.3.1 (Rev 26)
(Attach if not previously provided, 33013-1886, Sheet 1 (Rev 31)
including version/revision number) 33013-1900, Sheet 2 (Rev 14)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4701C 1.07 (As available)

Question Source: Bank # ______

Modified Bank # 159172 (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .4
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE 009 EK3.16	
	Importance Rating	3.8	

Knowledge of the reasons for the following responses as they apply to the small break LOCA: Containment temperature, pressure, humidity and level limits

RO Question #67

Given the following plant conditions:

- Plant experienced a small break LOCA
- Operating crew has completed the Immediate Actions in E-0, REACTOR TRIP OR SAFETY INJECTION
- CNMT pressure is 4.5 psig and rising slowly
- CNMT radiation is 14 R/hr and stable
- The Unit Supervisor reads the NOTE regarding when adverse CNMT values should be used

Which ONE of the following correctly completes the statement below?

Adverse CNMT values ____(1)____ be used in the EOPs with current conditions;

AND

Adverse values are used during EOPs due to ________(2)________.

A. (1) would
   (2) accident severity

B. (1) would not
   (2) instrument inaccuracies

C. (1) would
   (2) instrument inaccuracies

D. (1) would not
   (2) accident severity

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since the first part is correct and the candidate may believe that the reason for using adverse Containment values is due to the accident severity being more severe inside Containment.

B. INCORRECT. Plausible if the candidate believes that the Containment pressure limit for adverse values is 5 psig. Also, the second part is correct.

C. CORRECT. In accordance with EOP Background Documents (E-0 provided for reference), "Adverse CNMT values should be used whenever CNMT pressure is greater than 4 psig or CNMT radiation is greater than 10^5 R/hr." The indicated value of parameters monitored by instrumentation located within the harsh environment may be significantly different from actual values.

D. INCORRECT. Plausible if the candidate believes that the Containment pressure limit for adverse values is 5 psig and the candidate may believe that the reason for using adverse Containment values is due to the accident severity being more severe inside Containment.

Technical Reference(s): E-0 Background (p23; Rev 020)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP00C 1.05 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 159178

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 .5

55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.28	
	Importance Rating	4.1	

Knowledge of the purpose and function of major system components and controls.

RO Question #68

Given the following plant conditions:

- Plant is operating at 100% power
- An automatic reactor trip occurs due to a Main Generator fault
- The operating crew has transitioned to ES-0.1, REACTOR TRIP RESPONSE
- The CO is maintaining T_{AVG} using Steam Dumps per ES-0.1, step 11, ESTABLISH CONDENSER STEAM DUMP PRESSURE CONTROL

What is the correct position of:

- SDMSS, Steam Dump Mode Selector Switch
- HC-484, Steam Dump Valve Controller

	SDMSS	HC-484
A.	MANUAL	MANUAL
B.	MANUAL	AUTO
C.	AUTO	MANUAL
D.	AUTO	AUTO

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since this is the mode of operation used during the EOPs when the steam dump system is available and a RCS cooldown is required.

B. CORRECT. In accordance with ES-0.1, step 11, the operator is directed to adjust condenser steam dump controller HC-484 to 1005 psig in AUTO and to place steam dump mode selector switch to MANUAL.

C. INCORRECT. Plausible if the candidate confuses the purpose of the SDMSS and the HC-484 controller.

D. INCORRECT. Plausible since this is the normal mode of operation when the reactor is at power and immediately following a reactor trip.

Technical Reference(s): ES-0.1 (p12; Rev 03000)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4501C 1.09 (As available)

Question Source: Bank # 158411
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments: K/A match because transfer of steam dumps is made to pressure control mode, this is done due to loss of the T_{REF} signal from 1st stage pressure and inserting an electronic T_{REF} with the turbine tripped. Pressure control mode is used for more accurate temperature control, this is the knowledge part of the K/A.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 A2.01	
	Importance Rating	2.5	

Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds

RO Question #69

Given the following plant conditions:

- The following alarms have occurred sequentially in rapid succession:
  - o J-23, BATTERY BANK GROUND
  - o J-15, BATTERY CHRGR FAILURE OR PA INVERTER TROUBLE
  - o J-31, VITAL BATTERY MONITORING SYSTEM
  - o J-21, 1A OR 1B BATTERY UNDERVOLTAGE
- 'A' Battery voltage is 108 VDC and lowering

What is the most probable reason for these alarms **AND** the procedure that will provide guidance for the operating crew?

NOTE: ER-ELEC.2, RECOVERY FROM LOSS OF A OR B DC TRAIN
ER-ELEC.10, RESTORING BATTERY CHARGERS

A. Battery bank ground; ER-ELEC.2
B. Battery bank ground; ER-ELEC.10
C. Loss of power to 'A' Battery Charger; ER-ELEC.2
D. Loss of power to 'A' Battery Charger; ER-ELEC.10

Answer: A

Explanation (Optional):

A. CORRECT. In accordance with AR-J-23, "IF a DC train is lost, OR DC bus voltage is ≤ 110V, THEN REFER to ER-ELEC.2." An entry condition to ER-ELEC.2 is J-23 and also DC Bus Voltage < 110 VDC.

B. INCORRECT. Plausible since the first part is correct and ER-ELEC.10 can be entered from annunciator response procedures for J-15 and J-31.

C. INCORRECT. Plausible since loss of a battery charger will result in many of the alarms that are given. The second part is correct.

D. INCORRECT. Plausible since loss of a battery charger will result in many of the alarms that are given and ER-ELEC.10 can be entered from annunciator response procedures for J-15 and J-31.

Technical Reference(s): AR-J-23 (Rev 01101)
(Attach if not previously provided, ER-ELEC.2 (p4; Rev 01502)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0901C 1.06 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165604

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .5
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 K3.02	
	Importance Rating	4.1	

Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: ED/G

RO Question #70

Given the following plant conditions:

- The plant is at 100% power
- 4160V busses are in a Normal 50/50 Offsite Power alignment
- 'B' and 'C' SW Pumps are running
- 'A' and 'D' SW Pumps are selected in the Screenhouse
- A Large Break LOCA occurs concurrent with Station Service Transformer 17 suddenly failing

Which ONE of the following correctly completes the statement below?

'B' D/G starts and ties into ______________ and will operate for at least _______ under design Safeguards load conditions without replenishing the onsite fuel inventory.

A. Bus 17 ONLY; 24 hours

B. Buses 16 and 17; 24 hours

C. Bus 17 ONLY; 40 hours

D. Buses 16 and 17; 40 hours

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since 'B' EDG would start and tie into Bus 17 only if there was not a Safety Injection signal. The second part is plausible as this is the time that the EDGs can operate at their design ratings.

B. INCORRECT. Plausible since the first part is correct and the second part is plausible as this is the time that the EDGs can operate at their design ratings.

C. INCORRECT. Plausible since 'B' EDG would start and tie into Bus 17 only if there was not a Safety Injection signal. Also, the second part is correct.

D. CORRECT. In accordance with UFSAR, Section 8.3.1.1.4.2a failure of station service for Bus 17, concurrent with a Safety Injection signal, would result in both Buses on B Train (Bus 16 and 17) to be powered by 'B' EDG. According to UFSAR, Section 9.5.4 the minimum permissible onsite fuel inventory ensures that both diesel generators can carry the design loads of required safeguards equipment for any LOCA conditions for at least 40 hours.

Technical Reference(s): UFSAR 8.3.1.1.4.2 (Rev 26)
(Attach if not previously provided, UFSAR 9.5.4 (Rev 26)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0601C 1.06 (As available)

Question Source: Bank # 159169
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ______

Comments: K/A match because the question deals with how an electrical malfunction in conjunction with a SI signal changes the power supply to safeguard busses. In the specific instance, since the Bus 17 normal feed is deenergized, both Buses 16 and 17 will transfer to the

associated Diesel Generator if a SI signal is present. Without the Si signal, Bus 17 would be the only Bus to transfer to the Diesel Generator.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE 025 AA1.09	
	Importance Rating	3.2	

Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: LPI pump switches, ammeter, discharge pressure gauge, flow meter, and indicators

RO Question #71

Given the following plant conditions:

- The plant is in CSD during a refueling outage
- Loop level has been lowered for S/G inspections
- 'A' RHR Pump running with RCS level stable at 8 inches

Subsequently, the following conditions are observed:

- Annunciator A-20, RESIDUAL HEAT REMOVAL LOOP LO FLOW - 400 GPM, alarms
- RHR flow indicator (FI-626) is fluctuating from 0 gpm to 1500 gpm
- PPCS point I0685A, RHR PUMP A MOTOR CURRENT, is oscillating

Which ONE of the following describes the action required in accordance with AP-RHR.2, LOSS OF RHR WHILE OPERATING AT RCS REDUCED INVENTORY CONDITIONS?

NOTE: HCV-626, RHR HEAT EXCHANGER BYPASS AOV

A. Stop 'A' RHR Pump and place 'B' RHR Pump in service for RCS cooling.
B. Stop 'A' RHR Pump and isolate letdown and any known RCS drain paths.
C. Place HCV-626 in MANUAL and raise RHR flowrate to between 400 and 500 gpm.
D. Throttle the manual isolation for HCV-626 to stabilize RHR flowrate between 400 and 500 gpm.

Answer: B

Explanation (Optional):

A. INCORRECT. Plausible since AP-RHR.2 directs this action if the running RHR Pump tripped for a reason other than a loss of Net Positive Suction Head (NPSH).

B. CORRECT. In accordance with AP-RHR.2, steps 5 and 6, if indications exist of RHR Pump cavitating, the running RHR Pump is stopped and letdown and known drain paths are isolated.

C. INCORRECT. Plausible since AR-A-20 directs the operator to adjust flow if operating near the alarm setpoint.

D. INCORRECT. Plausible since AR-A-20 directs the operator to adjust flow if operating near the alarm setpoint and AP-RHR.2 contains guidance for EO local actions if operation from the MCB is unsuccessful.

Technical Reference(s): AP-RHR.2 (Rev 01800)
(Attach if not previously provided, AR-A-20 (Rev 9)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP25C 2.01 (As available)

Question Source: Bank # 158644
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 K2.01	
	Importance Rating	3.0*	

Knowledge of power supplies to the following: Containment cooling fans

RO Question #72

Given the following plant conditions:

- Plant is operating at 100% power
- PT-429, PRZR PRESS XMTR, is defeated for calibration
- 'A' and 'B' Containment Recirc Fans are running
- Bus 14 has tripped on overcurrent

Following the Bus 14 trip, which ONE of the following states the Containment Recirc Fans that are running? (**Assume no operator actions taken**)

A. 'B' CNMT Recirc Fan, ONLY
B. 'B' and 'C' CNMT Recirc Fans, ONLY
C. 'A' and 'B' CNMT Recirc Fans, ONLY
D. ALL CNMT Recirc Fans

Answer: B

Explanation (Optional):

A. INCORRECT. Plausible since the candidate may not recognize that the loss of Bus 14 caused Instrument Bus 'B' to de-energize which results in PT-430 losing power which actuates a Safety Injection signal.

B. CORRECT. In accordance with P-12, 'A' and 'D' CNMT Recirc Fans are powered from Bus 14; 'B' and 'C' CNMT Recirc Fans are powered from Bus 16. Loss of Bus 14 due to overcurrent will prevent 'A' EDG from re-energizing the bus. Loss of Bus 14 will result in loss of MCC 'C' and subsequently Instrument Bus 'B'. According to P-10, PI-430 receives power from 'B' Instrument Bus. Loss of PI-430 with PT-429 defeated will result in Safety Injection actuation on low pressure (2/3 coincidence).

C. INCORRECT. Plausible since the candidate may believe that 'A' and 'B' CNMT Recirc Fans are powered from Bus 16.

D. INCORRECT. Plausible since upon receipt of a Safety Injection signal, all CNMT Recirc Fans are running and the candidate may not recognize that Bus 14 will remain de-energized.

Technical Reference(s): P-12 (p18,35, 37; Rev 024)
(Attach if not previously provided, P-10 (p12-15, 47; Rev 01901)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2201C 1.05 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New 165601

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43 ______

Comments:

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

Ability to monitor automatic operation of the PZR PCS, including: PZR pressure

RO Question #73

Given the following plant conditions:

- Plant is at 93% power after a brief transient
- PRZR Spray Valves indicate PARTIALLY OPEN
- PRZR Proportional Heaters are at MINIMUM output
- PRZR Backup Heaters are OFF with their control switch in AUTO

Which of the following values of PRZR pressure will result in the conditions indicated above?

A. 2220 psig

B. 2235 psig

C. 2260 psig

D. 2310 psig

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since this is the setpoint at which the Backup Heaters turn OFF.

B. INCORRECT. Plausible since this is the setpoint where PRZR Spray Valves are set to begin to OPEN with bias set into the controllers. However, the Proportional Heaters would NOT be at their minimum output.

C. CORRECT. In accordance with P-1, Section 6.3.2.3, Pressurizer Spray Valves begin to OPEN at 2260 psig, Proportional Heaters are at minimum output at 2250 psig, and Backup Heaters are OFF at 2220 psig.

D. INCORRECT. Plausible since this is the setpoint at which the Pressurizer Spray Valves are fully OPEN.

Technical Reference(s): P-1 (p30; Rev 07201)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1901C 1.07 (As available)

Question Source: Bank # 158673
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2010 Ginna ILT SRO Retake
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015 A3.01	
	Importance Rating	3.8	

Ability to monitor automatic operation of the NIS, including: Console and cabinet indications

RO Question #74

Following a Reactor trip, Intermediate Range Startup Rate is -1/3 decades per minute on both channels.

Current indications on Intermediate Range Channels are as follows:

- N-35 is 3E-7 amps
- N-36 is 8E-8 amps

Based upon the above indications and trends, which ONE of the following completes the statement below?

Both Source Range Nuclear Instruments will energize when ____(1)____ lowers to ____(2)____ amps.

	(1)	(2)
A.	N-35	5E-11
B.	N-36	5E-11
C.	N-35	1E-10
D.	N-36	1E-10

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with P-1, P-6 permissive resets when both Intermediate Range NIs are < 5E-11 amps. With both Intermediate Range SURs equal at -1/3 dpm, and since N-35 is reading approximately ½ decade higher than N-36, N-35 will be the second Intermediate Range NI to lower < 5E-11 amps.

B. INCORRECT. Plausible since when raising reactor power, when a single Intermediate Range NI is > 1E-10 amps the Source Range NIs can be blocked and de-energized. The second part is correct.

C. INCORRECT. Plausible since the first part is correct and the Source Range NIs can be blocked and de-energized when Intermediate Range NIs (1 of 2) is > 1E-10 amps.

D. INCORRECT. Plausible since when raising reactor power, when a single Intermediate Range NI is > 1E-10 amps the Source Range NIs can be blocked and de-energized.

Technical Reference(s): P-1 (p42; Rev 07201)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3301C 1.07 (As available)

Question Source: Bank # 165611
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2002 Beaver Valley ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.11	
	Importance Rating	3.8	

Ability to control radiation releases.

RO Question #75

Given the following plant conditions:

- Plant was operating at 100% power
- A S/G Tube Rupture occurred in 'A' S/G
- The operating crew is performing E-3, STEAM GENERATOR TUBE RUPTURE

Which ONE of the following describes the actions required to <u>directly</u> minimize radiation releases in accordance with E-3?

A. Stop both RCPs
B. Adjust 'A' Atmospheric Relief Valve controller to 1050 psig
C. Secure Auxiliary Feedwater flow to 'A' S/G
D. Perform RCS cooldown

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since RCPs are tripped in E-3 if the SGTR is of sufficient size to cause RCS depressurization and meeting trip criteria. Incorrect because the RCPs are tripped to ensure against possible operator misdiagnosis, operator error, or a multiple failure event scenario.

B. CORRECT. In accordance with E-3, step 4 the ruptured S/G ARV controller is set at 1050 psig in AUTO. According to E-3 Background Document "Isolation of the ruptured steam generator effectively minimizes release of radioactivity from this generator."

C. INCORRECT. Plausible since Auxiliary Feedwater flow to the ruptured S/G is secured in E-3 once S/G narrow range level is greater than 7%. Incorrect because this is performed to minimize the potential for steam generator overfill.

D. INCORRECT. Plausible since the RCS is cooled down in E-3 to a temperature based on S/G pressure to allow the operating crew to depressurize the RCS in a later step to stop the RCS to S/G flow.

Technical Reference(s): E-3 (p3,5,7,14; Rev 04900)
(Attach if not previously provided, E-3 Background Document (p55,61,67,95,100; Rev 02000)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP03C 1.03 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 158417

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 .11
55.43

Comments: K/A match since the applicant is required to demonstrate knowledge of procedural actions required to minimize the radioactive release during a SGTR.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	EPE 038 EA2.06	
	Importance Rating		4.4

Ability to determine or interpret the following as they apply to a SGTR: Shutdown margins and required boron concentrations

SRO Question #76

The CAUTION before Step 1 of ES-3.1, POST SGTR COOLDOWN USING BACKFILL, warns that the RCP in the ruptured loop should NOT be started first.

Which ONE of the following explains the reason for this CAUTION?

Starting this RCP could cause:

A. a rapid repressurization of the RCS.
B. an internal pressure surge, resulting in further failure of damaged tubes.
C. a slug of unborated water to pass through the core and cause a return to criticality.
D. a rapid cooldown of the reactor vessel and present a challenge to the Integrity Critical Safety Function.

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible since the candidate may infer the possibility of rapidly repressurizing the RCS due to the ruptured S/G being at a slightly higher pressure.
B. INCORRECT. Plausible since the candidate may infer that starting the RCP, which would add energy to the RCS, in the loop with the ruptured S/G could result in further tube damage and degradation.
C. CORRECT. In accordance with ES-3.1, Post-SGTR Cooldown Using Backfill, Background Document: "Initiation of forced RCS flow by starting an RCP in a loop with an intact S/G will ensure mixing of the loop with the borated water in the diluted water in the stagnant other RCS loop such that the localized dilution in the core region in precluded."
D. INCORRECT. Plausible since the candidate may infer that the colder water in the S/G may result in a PTS concern.

Technical Reference(s): ES-3.1 Background Document (p9; Rev 8)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RES31C 1.02 (As available)

Question Source: Bank # 165627
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 .5

Comments: SRO Justification:

- SRO question since it requires knowledge of the background document information: is not system knowledge; not an immediate action; not an entry condition for AOP/EOP; not the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.44	
	Importance Rating		4.4

Knowledge of emergency plan protective action recommendations.

SRO Question #77

Shift Manager has declared a General Emergency on the following conditions:

- R-29 and R-30 are reading 1050 R/hr
- RCS is in a Saturated condition per EOP FIG 1.0, FIGURE MIN SUBCOOLING
- A RED path condition exists on F-0.5, CONTAINMENT CSFST
- Wind direction is 125 degrees
- Dose projections are < 1 REM (TEDE) and < 5 REM (CDE)

Which ONE of the following will be the correct PAR determination?

A. Evacuate W-1, M-1, M-6, and M-8 and evacuate Lake ERPAs
B. Evacuate W-1, M-1 and evacuate Lake ERPAs
C. Shelter W-1, M-1 and evacuate Lake ERPAs
D. No PARs required for initial General Emergency declaration

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible if the candidate determines a loss of Containment has occurred and answers YES to the third question on EP-CE-111-F-03 and utilizes Table 1 for evacuation of downwind areas.

B. CORRECT. In accordance with EP-AA-1012, Addendum 3, Fission Product Barrier Matrix, a RED path on F-0.5, Containment CSFST constitutes Potential Loss of Containment. According to EP-CE-111-F-03, since: 1) A GE has been classified, 2) this is the initial PAR, 3) there is NOT a loss of Containment per the EALs, 4) a Hostile Action event is NOT in progress, 5) the PAR is from the Control Room, 6) Evacuate downwind areas (Table 3) and evacuate all Lake ERPAs. With a wind direction of 125 degrees, per Table 3 evacuate ERPAs W-1 and M-1.

C. INCORRECT. Plausible if the candidate does not recognize that the PAR is being made from the Control Room and determines that a Controlled Containment vent is occurring.

D. INCORRECT. Plausible since prior to transitioning to Exelon, there was no PAR determination on the initial classification of General Emergency.

Technical Reference(s): EP-CE-111-F-03 (Rev A)
(Attach if not previously provided, EP-AA-1012, Addendum 3 (p50; Rev 3)
including version/revision number)

Proposed references to be provided to applicants during examination: EP-CE-111-F-03; EP-AA-1012, Addendum 3 p1-50

Learning Objective: RSC02C 12.00 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New 165653

Question History: Last NRC Exam ____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ____
55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	103 A2.03	
	Importance Rating		3.8*

Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation

SRO Question #78

Given the following plant conditions:

- Plant is at 100% power
- A LOCA occurs concurrent with loss of Offsite Power
- 'B' D/G failed to start and cannot be started
- RCS pressure is 1600 psig and lowering slowly
- HCO is performing ATT-27.0, ATTACHMENT AUTOMATIC ACTION VERIFICATION, and notes that many Containment Isolation valves are not in the safeguards position
- HCO has depressed the Containment Isolation pushbutton
- Annunciator A-26, CNMT ISOLATION, is extinguished

Which ONE of the following completes the statement below?

ATT-27.0 ____(1)____ perform **ALL** the required actions to achieve Containment Isolation and the next action required by the HCO is to ________(2)__________.

A. (1) will NOT
   (2) refer to ATT-3.0, ATTACHMENT CI/CVI

B. (1) will
   (2) refer to ATT-3.0, ATTACHMENT CI/CVI

C. (1) will
   (2) ensure Service Water Outlet Valves have failed OPEN

D. (1) will NOT
   (2) ensure Service Water Outlet Valves have failed OPEN

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with ATT-27.0, step 6.b RNO the operator is directed to manually close affected CI and CVI valves (from MCB). However, the 'B' Train of safeguards is deenergized and the associated valves cannot be operated from the MCB. Therefore, ATT-27.0 directs the operator to refer to ATT-3.0 to complete CNMT Isolation by locally closing the associated valves.

B. INCORRECT. Plausible since ATT-27.0, step 6.b RNO directs the operator to manually close affected CI and CVI valves (from MCB). However, the 'B' Train of safeguards is deenergized and the associated valves cannot be operated from the MCB. Additionally, the second part is correct.

C. INCORRECT. Plausible since ATT-27.0, step 6.b RNO directs the operator to manually close affected CI and CVI valves (from MCB). However, the 'B' Train of safeguards is deenergized and the associated valves cannot be operated from the MCB. Additionally, the second part is plausible as this is the next action performed by the HCO after completing ATT-3.0.

D. INCORRECT. Plausible since the first part is correct and the second part is plausible as this is the next action performed by the HCO after completing ATT-3.0.

Technical Reference(s): ATT-27.0 (Rev 00400)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP00C 2.01 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165644

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments: K/A is matched because Ginna does not separate Phase A and B Isolation for Containment. SRO level because the question requires the candidate to have knowledge of the specific actions contained EOP Attachments.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	EPE W/E01 EA2.1	
	Importance Rating		4.0

Ability to determine or interpret the following as they apply to the (Reactor Trip or Safety Injection Rediagnosis): Facility conditions and selection of appropriate procedures during abnormal and emergency operations

SRO Question #79

Given the following INITIAL plant conditions:

- SI was initiated due to uncontrolled lowering of PRZR pressure
- The crew transitioned to ECA-1.2, LOCA OUTSIDE CONTAINMENT
- The leak was located and isolated
- The crew is preparing to exit ECA-1.2

CURRENT plant conditions:

- PRZR level and pressure have begun lowering
- CO reports that 'A' S/G level is rising in an uncontrolled manner
- US has entered ES-0.0, REDIAGNOSIS

Which ONE of the following describes the correct transition from ES-0.0?

A. Return to ECA-1.2
B. Enter E-0, REACTOR TRIP OR SAFETY INJECTION
C. Enter E-1, LOSS OF REACTOR OR SECONDARY COOLANT
D. Enter E-3, STEAM GENERATOR TUBE RUPTURE

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible since ES-0.0 has a transition point to ECA-1 series procedure and the crew was performing ECA-1.2. Incorrect because the leak which caused the crew to transition to ECA-1.2 has been isolated.

B. INCORRECT. Plausible since the returning to E-0 is a procedure flowpath in the EOP network when a Safety Injection occurs after performance of E-0.

C. INCORRECT. Plausible since entry into E-1 is a normal procedure flowpath in the EOP network when Safety Injection is initiated and RCS parameters are degrading. Additionally, E-1 is a transition point from ES-0.0.

D. CORRECT. In accordance with ES-0.0, step 4, if the crew determines that the S/G is not faulted and that a S/G is ruptured, the operator is directed to be in E-3 or an ECA-3 series procedure.

Technical Reference(s): ES-0.0 (Rev 11)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RES00C 2.01 (As available)

Question Source: Bank # 165645
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE 026 2.2.40	
	Importance Rating		4.7

Ability to apply Technical Specifications for a system.

SRO Question #80

Given the following plant conditions:

- Plant is at 100% power
- 'A' CCW Pump has been taken out of service for a bearing replacement. Total time for maintenance is 30 hours.
- 6 hours into the 'A' CCW Pump work has started, an EO finds that 'B' D/G has a failed oil line and large oil leak.
- 'B' D/G is declared INOPERABLE.

What effect does 'B' D/G inoperability have on the CCW system and plant operation?

____(1)____ after 'B' D/G is declared INOPERABLE, 'B' CCW Pump must be declared INOPERABLE and Technical Specification ____(2)____ applied.

A. (1) 4 hours
   (2) 3.0.3

B. (1) 12 hours
   (2) 3.0.3

C. (1) 4 hours
   (2) 3.7.7

D. (1) 12 hours
   (2) 3.7.7

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since the time requirement is correct and most Technical Specification systems require entry into LCO 3.0.3 when both trains are INOPERABLE. Incorrect because of the inability to achieve Cold Shutdown without the CCW system.

B. INCORRECT. Plausible since the time requirement is for Offsite power to one or more 480V safeguards bus(es) INOPERABLE to declare required features INOPERABLE when its redundant required feature is INOPERABLE (LCO 3.8.1 Condition A). Also plausible since most Technical Specification systems require entry into LCO 3.0.3 when both trains are INOPERABLE. Incorrect because of the inability to achieve Cold Shutdown without the CCW system.

C. CORRECT. In accordance with Technical Specification 3.8.1, Condition B "One DG inoperable", Required Action B.2 "Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable" with a Completion Time of "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)". With 'A' CCW Pump INOPERABLE, Technical Specification 3.7.7 applies, with Condition D NOTE "LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status."

D. INCORRECT. Plausible since the time requirement is for Offsite power to one or more 480V safeguards bus(es) INOPERABLE to declare required features INOPERABLE when its redundant required feature is INOPERABLE (LCO 3.8.1 Condition A). The second part is correct.

Technical Reference(s): Technical Specification 3.7.7 (Rev 80)
(Attach if not previously provided, including version/revision number) Technical Specification 3.8.1 (Rev 109)

Proposed references to be provided to applicants during examination: Technical Specification 3.7.7 & 3.8.1

Learning Objective: R0801C 1.13, R2801C 1.13 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165637

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	EPE 007 2.4.6	
	Importance Rating		4.7

Knowledge of EOP mitigation strategies.

SRO Question #81

Given the following plant conditions:

- Plant is operating at 100% power
- The turbine trips
- The reactor fails to automatically trip and manual trip is unsuccessful
- Bus 13 and 15 Normal Feeder Breakers are opened
- S/G B MS SAFETY VALVE 3508 has opened and failed to fully reseat

NOTE: E-0, REACTOR TRIP OR SAFETY INJECTION
FR-S.1, RESPONSE TO REACTOR RESTART/ATWS
E-1, LOSS OF REACTOR OR SECONDARY COOLANT
E-2, FAULTED STEAM GENERATOR ISOLATION
ES-1.1, SI TERMINATION

Which ONE of the following describes the EOP procedure flowpath?

A. E-0 will transition to E-1 on indications of a loss of secondary coolant;
E-1 will transition to E-2 to isolate the faulted S/G;
E-2 will transition to ES-1.1 on indications of RCS repressurization.

B. E-0 will transition to E-2 on faulted S/G indications;
E-2 will transition to ES-1.1 on indications of RCS repressurization.

C. E-0 will transition to FR-S.1 to trip reactor;
FR-S.1 will transition to E-0 following Rx trip;
E-0 will transition to E-2 on faulted S/G indication;
E-2 will transition to ES-1.1 on indication of RCS repressurization.

D. E-0 will transition to E-2 on faulted S/G indications;
E-2 will transition to E-1;
E-1 will transition to ES-1.1 on indications of RCS repressurization.

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible since E-0 has a transition step to E-1 and the candidate may not have the procedural knowledge to recognize that the E-2 transition step occurs prior to the E-1 transition step in E-0. Also plausible because E-1 contains a transition step to E-2 if the faulted S/G has not already been isolated.

B. INCORRECT. Plausible since the transition from E-0 is correct and the final transition will be to ES-1.1. Incorrect because E-2 will transition to E-1 prior to the final transition to ES-1.1.

C. INCORRECT. Plausible since E-0 step 1 has a transition to FR-S.1 if the reactor cannot be tripped. Incorrect because E-0, step 1 RNO contains actions for the operator to open Bus 13 and Bus 15 Normal Feeder Breakers to trip the Rod Drive MG Sets and E-2 will transition to E-1 prior to the final transition to ES-1.1. Also plausible if the candidate believes that this action occurs in FR-S.1 vice E-0.

D. CORRECT. In accordance with E-0, step 15, the operator is directed to go to E-2 if any S/G pressure is lowering in an uncontrolled manner. The operator then performs E-2 and transitions to E-1 at step 9. The operator will then perform E-1 through step 15, returning to step 1 until 'B' S/G is dry and the RCS repressurizes. This results in the final transition to ES-1.1 at E-1, step 12 or the Foldout Page.

Technical Reference(s):	E-0 (p3,13-14; Rev 048)
(Attach if not previously provided,	E-2 (p8; Rev 01302)
including version/revision number)	E-1 (p4,10,14,Foldout Page1-2; Rev 04100)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP00C 2.01, REP02C 2.1, REP01C 2.1 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	165635	

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.40	
	Importance Rating		3.9

Knowledge of refueling administrative requirements.

SRO Question #82

Given the following plant conditions:

- Plant is in MODE 6
- The Containment Equipment Hatch is removed and the roll-up door enclosure is installed
- While backing in a tractor trailer hits the enclosure damaging the roll-up door enclosure
- Maintenance personnel determine that the roll-up door CANNOT be closed

Which ONE of the following activities could be completed in this plant configuration?

A. Reactor Vessel Head detensioning

B. Reactor Vessel Head lift

C. Upper Internals lift

D. Control Rod unlatch checks

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with Technical Specification 3.9.3, Condition A "One or more containment penetrations not in required status"; the Required Action is "Suspend core alterations". Reactor Vessel Head stud detensioning is not considered a core alteration as there is no chance for a positive reactivity addition to occur.

B. INCORRECT. Plausible if the candidate believes that Reactor Vessel Head lift is not considered a core alteration. Incorrect because in accordance with RF-405, step 4.1 "Reactor Vessel Head lift is considered a core alteration due to the possibility to lift a Control Rod Assembly."

C. INCORRECT. Plausible if the candidate believes that Upper Internals lift is not considered a core alteration. Incorrect because in accordance with RF-406, step 4.1 "Control Rod Drive Shaft unlatching and Upper Internals removal is considered a core

alteration due to the possibility to lift a Control Rod Assembly." Most likely event would be lifting a fuel assembly if the upper plate guide pins are misaligned and will stick.

D. INCORRECT. Plausible if the candidate believes that Control Rod decoupling is not considered a core alteration. Incorrect because in accordance with RF-406, step 4.1 "Control Rod Drive Shaft unlatching and Upper Internals removal is considered a core alteration due to the possibility to lift a Control Rod Assembly." Control Rod unlatch checks are performed during Control Rod Drive Shaft unlatching.

Technical Reference(s): (Attach if not previously provided, including version/revision number)
Technical Specification 3.9.3 (Rev 107)
RF-405 (p5; Rev 004)
RF-406 (p5; Rev 004)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3701C 1.12 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New 165650

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ______
55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.6	
	Importance Rating		4.8

Ability to manage the control room crew during plant transients.

SRO Question #83

Given the following plant conditions:

- Plant is operating at 100% power
- A loss of Offsite Power and failure of both Diesel Generators has occurred
- The operating crew has entered ECA-0.0
- Operators are performing a 100°F/hr cooldown in ECA-0.0
- Energy Control has informed the Control Room that Circuit 767 is available for station use

NOTE: E-0, REACTOR TRIP OR SAFETY INJECTION
ECA-0.0, LOSS OF ALL AC POWER
ER-ELEC.1, RESTORATION OF OFFSITE POWER

What actions should the Unit Supervisor perform?

A. Direct the HCO to complete the cooldown and continue in ECA-0.0.
B. Direct the HCO to stop the cooldown and perform ER-ELEC.1 to restore Offsite Power.
C. Direct the HCO to continue the cooldown and in parallel have the CO perform ER-ELEC.1.
D. Direct the HCO to complete the cooldown and then restore Offsite Power per ER-ELEC.1.

Answer: **C**

Explanation (Optional):

A. INCORRECT. Plausible since the candidate may not recognize that the step to restore Offsite Power is an implied continuous action step according to A-503.1, Section 5.2.J.2 and continue through ECA-0.0 until the next transition out of that procedure. Also, the operator would have to complete the cooldown step.

B. INCORRECT. Plausible since the step to restore Offsite Power is an implied continuous action step according to A-503.1, Section 5.2.J.2 and the candidate may believe that the step is a transition to ER-ELEC.1 vice a refer to ER-ELEC.1.

C. CORRECT. In accordance with A-503.1, Section 5.2.A.7 "During implementation of EOPs it may be possible to help mitigate or terminate an event by restoring equipment or safety functions through use of an ER Procedure. If implementation of an ER procedure will likely result in the event being mitigated or terminated, then the ER procedure should be implemented to restore the equipment or safety function" provided the crew has the resources without significantly slowing the implementation of the EOP. The operating crew would continue performing actions in ECA-0.0 and perform ER-ELEC.2 in parallel to restore Offsite Power.

D. INCORRECT. Plausible since the candidate may not recognize that the step to restore Offsite Power is an implied continuous action step according to A-503.1, Section 5.2.J.2 and determine that the current step must be completed and then go back and re-perform the Offsite Power restoration step.

Technical Reference(s): A-503.1 (p21-22,26; Rev 04600)
(Attach if not previously provided, ECA-0.0 (p12; Rev 042)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP50C 1.08 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165649

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE 022 2.4.34	
	Importance Rating		4.1

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

SRO Question #84

Given the following plant conditions:

- Plant was initially at 100% power
- A fire occurred in 'B' Battery Room
- The crew entered ER-FIRE.5, ALTERNATE SHUTDOWN FOR BATTERY ROOM B FIRE
- Control Room actions are COMPLETED
- The CO has realigned and started 'A' Charging Pump in accordance with Attachment 3, Control Operator (CO)
- Pressurizer level is 13% and stable

Which ONE of the following describes the actions for the CO to take in accordance with ER-FIRE.5?

A. MAINTAIN 'A' Charging Pump speed. If Pressurizer level cannot be maintained, START the 'B' or 'C' Charging Pump.

B. MAINTAIN 'A' Charging Pump speed. If Pressurizer level cannot be maintained, REQUEST SM INITIATE alternate injection flow using 'A' SI Pump.

C. RAISE 'A' Charging Pump speed. If Pressurizer level cannot be maintained, START 'B' or 'C' Charging Pump.

D. RAISE 'A' Charging Pump speed. If Pressurizer level cannot be maintained, REQUEST SM INITIATE alternate injection flow using 'A' SI Pump.

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible since ER-FIRE.5 would have the operator maintain ‘A’ Charging Pump speed if Pressurizer level was > 15% and stable or rising. Also, starting an additional Charging Pump is guidance provided in EOPs for the operator to gain control of Pressurizer level.

B. INCORRECT. Plausible since ER-FIRE.5 would have the operator maintain ‘A’ Charging Pump speed if Pressurizer level was > 15% and stable or rising. The second part is correct if Pressurizer level cannot be maintained with one Charging Pump.

C. INCORRECT. Plausible since the first part is correct and starting an additional Charging Pump is guidance provided in EOPs for the operator to gain control of Pressurizer level.

D. CORRECT. In accordance with ER-FIRE.5, Attachment 3, step 8.0 “Verify PRZR level greater than 15% and stable or rising. If not, then raise charging flow.” Step 8.1 directs the operator to “If PRZR level cannot be maintained with one Charging Pump, then request the SM initiate alternate injection flow, in accordance with Attachment 11, Alternate Injection SI Pump A.”

Technical Reference(s): ER-FIRE.5 (p25-28; Rev 029)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RER22C #8 (As available)

Question Source: Bank # 165636
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments: SRO level because the question requires the candidate to have knowledge of the specific actions contained ER Procedure Attachments.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034 K1.04	
	Importance Rating		3.5

Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling System and the following systems: NIS

SRO Question #85

Given the following plant conditions:

- Plant is in MODE 6, with core alterations in progress
- Electric plant is in a normal 50/50 lineup
- Power Range Nuclear Instrument N-44 has been defeated for maintenance
- Source Range Nuclear Instrument N-32 is selected to audible indication
- Offsite Power Circuit 7T trips
- 'A' D/G fails to start and cannot be started
- Annunciator E-14, LOSS B INSTR. BUS, is LIT

Which ONE of the following completes the statement below?

Core alterations ___1___ continue based on ____2_____.

A. 1. CANNOT
   2. Both Source Ranges are INOPERABLE

B. 1.CANNOT
   2. N-31 must be selected for audible indication to be OPERABLE

C. 1.CAN
   2. Both Source Ranges are OPERABLE

D. 1.CAN
   2. N-31 is OPERABLE and only one Source Range is required

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with P-12 a loss of Offsite Power Circuit 7T, with 'A' EDG failure, will result in Bus 14 being de-energized which causes Instrument Bus 'B' losing power. This will cause a loss of Power Range Nuclear Instrument N-42. Loss of the second PRNI will cause Permissive P-10 to de-energize resulting in a block of Source Range High Voltage, de-energizing both SRNIs.

B. INCORRECT. Plausible since the first part is correct and the candidate may believe that only one Source Range Instrument is required since a majority of safety systems utilize a redundant instrument/component scheme where only half the instruments/components are required. Also, the audible count instrument is powered from Instrument Bus 'D' which remains energized in this event.

C. INCORRECT. Plausible since Source Range Nuclear Instruments receive power from Instrument Buses A and C, which will not lose power. Instrument Bus 'A' receives normal power from Bus 14 (which is de-energized); however, the 'A' Station Main Battery will automatically assume the Instrument Bus load upon loss of power to the associated Battery Charger. Instrument Bus 'C' receives normal power from Bus 16.

D. INCORRECT. Plausible since Source Range Instrument N-32 used to receive power from Instrument Bus 'B', which would be de-energized with the given plant conditions. Also, the candidate may believe that only one Source Range Instrument is required since a majority of safety systems utilize a redundant instrument/component scheme where only half the instruments/components are required.

Technical Reference(s): P-1 (p42; Rev 07201)

(Attach if not previously provided, P-12 (p16, 18, 21-22; Rev 024)

including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC10C 1.06 (As available)

Question Source: Bank #

Modified Bank # 165678 (Note changes or attach parent)

New

Question History: Last NRC Exam 2011 Vogtle ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.6

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	EPE W/E07 EA2.2	
	Importance Rating		3.9

Ability to determine or interpret the following as they apply to the (Saturated Core Cooling): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

SRO Question #86

FR-C.3, RESPONSE TO SATURATED CORE COOLING, has the following NOTE at the beginning of the procedure:

<u>**NOTE**</u>: If either ECA-3.2, SGTR WITH LOSS OF REACTOR COOLANT – SATURATED RECOVERY DESIRED or ES-1.3, TRANSFER TO COLD LEG RECIRCULATION is in effect, this procedure should not be performed.

Which ONE of the following describes the basis for the ECA-3.2 exception in this NOTE?

FR-C.3 directs the operator to . . .

A. restore Letdown, while ECA-3.2 maintains Letdown isolated.

B. re-establish ECCS to raise subcooling, while ECA-3.2 attempts to terminate SI flow.

C. close any open PRZR PORVs, while ECA-3.2 opens PORVs to depressurize the RCS.

D. allow a RCS cooldown rate in excess of 100°F/hr, while ECA-3.2 limits the cooldown rate.

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since ECA-3.2 does address getting letdown restored, but FR-C.3 does not. If the candidate does not fully understand the overall big picture of ECA-3.2 and FR-C.3 this may seem plausible.

B. CORRECT. In accordance with FR-C.3 Background Document "FR-C.3 directs a reestablishment of RCS subcooling via safety injection flow. This is inconsistent with

actions specified in ECA-3.2 which reduce RCS subcooling via SI flow reduction in order to minimize primary-to-secondary leakage for a steam generator tube rupture."

C. INCORRECT. Plausible since PORV operation is addressed in each procedure, but this is not the basis for the NOTE. If the candidate does not fully understand the overall big picture this may seem plausible.

D. INCORRECT. Plausible since a 100°F/hr cooldown rate is addressed in ECA-3.2, but not in FR-C.3 since the operator is trying to establish SI and subcooling. The candidate may believe that this is addressed in both procedures, but this is not the basis for the NOTE.

Technical Reference(s): FR-C.3 Background Document (p6; Rev 1)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRC3C 1.03 (As available)

Question Source: Bank # 165646
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ______
55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	075 2.4.4	
	Importance Rating		4.7

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

SRO Question #87

Given the following plant conditions:

- O-1.2, PLANT STARTUP FROM HOT SHUTDOWN TO FULL LOAD, is being performed
- Reactor power is 6% and stable
- Turbine is latched and speed is 1800 rpm
- 'B' Circulating Water Pump is out of service for maintenance
- 'A' Circulating Water Pump trips

NOTE: E-0, REACTOR TRIP OR SAFETY INJECTION
ES-0.1, REACTOR TRIP RESPONSE
ES-1.1, SI TERMINATION
AP-CW.1, LOSS OF A CIRC WATER PUMP
AP-TURB.1, TURBINE TRIP WITHOUT RX TRIP REQUIRED
AP-TURB.4, LOSS OF CONDENSER VACUUM

1) What procedure will the US enter **first**?
2) What procedure will be transitioned to **first** from the initial procedure?

A. 1) E-0
2) ES-0.1

B. 1) E-0
2) ES-1.1

C. 1) AP-CW.1
2) AP-TURB.1

D. 1) AP-CW.1
2) AP-TURB.4

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since according to P-1, loss of both Circ Water Pumps and the P-7 Permissive (8% reactor power) would result a reactor trip due to Turbine trip. This would be an uncomplicated trip so the transition to ES-0.1 would be normal.

B. INCORRECT. Plausible since according to P-1, loss of both Circ Water Pumps and the P-7 Permissive (8% reactor power) would result a reactor trip due to Turbine trip. Also, ES-1.1 is a transition from E-0.

C. CORRECT. In accordance with AP-CW.1 "This procedure provides the actions necessary to respond to a loss of a Circ Water Pump while the plant is at power." According to AP-CW.1, step 1 RNO "IF power less than 8%, THEN verify Turbine trip and go to AP-TURB.1".

D. INCORRECT. Plausible since the initial procedure entered is correct and AP-TURB.4 would be the procedure used for lowering condenser vacuum which is plausible since condenser vacuum would start to degrade with the given plant conditions; however, the transition to AP-TURB.1 occurs first.

Technical Reference(s): AP-CW.1 (p2-3; Rev 01400)
(Attach if not previously provided, P-1 (p40-42; Rev 07201)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP24C 2.01 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165676

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis ____X____

10 CFR Part 55 Content: 55.41 ________
55.43 .2________

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	064 2.2.44	
	Importance Rating		4.4

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

SRO Question #88

Given the following plant conditions:

- Plant is operating at 100% power
- Bus 14 Normal Feeder Breaker has tripped OPEN
- Control Operator reports Bus 14 voltage is 0 volts
- Annunciator L-5, SAFEGUARD BUS MAIN BREAKER OVERCURRENT TRIP, is alarming

1) What is the status of the 'A' Diesel Generator?

2) What position of the Bus 14 Normal Feeder Breaker Control Switch is required prior to depressing the Overcurrent Reset pushbutton (inside MCB) in accordance with AR-L-5, SAFEGUARD BUS MAIN BREAKER OVERCURRENT TRIP?

A. 1) INOPERABLE
   2) PULL STOP

B. 1) INOPERABLE
   2) AFTER TRIP

C. 1) OPERABLE
   2) PULL STOP

D. 1) OPERABLE
   2) AFTER TRIP

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since operability determination is correct and AR-L-5 directs the operator to place the associated D/G supply breaker in PULL STOP.

B. CORRECT. In accordance with Technical Specification 3.8.1 Basis "A DG is considered OPERABLE when : The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480V safeguards buses on actuation of Loss of Power (LOP) DG Instrumentation within 10 seconds". Also stated "Any 480V bus fault which opens and/or prevents closure of the breakers from offsite power or the DGs requires declaring the offsite power source or DG inoperable, as applicable." According to AR-L-5, step 5 "WHEN the faulted component is identified AND its supply breaker verified Open, THEN restore the affected Safeguards bus by performing the following: a. Ensure associated D/G supply breaker is in Pull Stop; b. Reset the affected Safeguards bus normal feed breaker by taking the control switch to the AFTER TRIP position; c. Depress the Overcurrent RESET pushbutton for the affected Safeguards bus normal feed breaker."

C. INCORRECT. Plausible since 'A' Diesel Generator would be running, the candidate may determine that it is OPERABLE. Also, AR-L-5 directs the operator to place the associated D/G supply breaker in PULL STOP.

D. INCORRECT. Plausible since 'A' Diesel Generator would be running, the candidate may determine that it is OPERABLE. Also, the breaker control switch position is correct.

Technical Reference(s): Technical Specification 3.8.1 Basis (p5-6; Rev 74)
(Attach if not previously provided, AR-L-5 (Rev 10)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0701C 1.11 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165679

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	APE 033 2.4.8	
	Importance Rating		4.5

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

SRO Question #89

Given the following plant conditions:

- The plant is being returned to power following a mid-cycle outage
- The operating crew has just transitioned from AFW to MFW
- Intermediate Range N-35 fails high

NOTE: ER-NIS.2, IR MALFUNCTION

E-0, REACTOR TRIP OR SAFETY INJECTION

A-503.1, EMERGENCY AND ABNORMAL OPERATING PROCEDURES USERS GUIDE

1) What procedure will the operating crew enter due to the failed N-35 detector?

AND

2) When can Abnormal Operating Procedure (AOP) actions be taken when performing EOPs in accordance with A-503.1?

A. 1) E-0

2) AOP actions can only be taken if they do not conflict with the EOP actions and any licensed operator is available

B. 1) E-0

2) AOP actions can always be taken with the EOP actions and an on-watch operator is available

C. 1) ER-NIS.2

2) AOP actions can only be taken if they do not conflict with the EOP actions and any licensed operator is available

D. 1) ER-NIS.2

2) AOP actions can always be taken with the EOP actions and an on-watch operator is available

Answer: **A**

Explanation (Optional):

A. CORRECT. According to P-1, the Intermediate Range High Flux reactor trip occurs at 25% current equivalent power on 1 of 2 coincidence. This trip is manually blocked by Permissive P-10 at 8% power on 2 of 4 Power Range NIs. In accordance with O-1.2, Section 6.7, the transition from AFW to MFW occurs between 2% and 5% reactor power; therefore, N-35 would cause a reactor trip and the operating crew would implement E-0. According to A-503.1, Section 5.2.6 "When performing EOPs, various plant conditions may occur which would normally be addressed by AOPs (such as ARs or APs). Actions may be taken per AOPs that DO NOT conflict with the actions of the EOPs if adequate resources are available. The AOP should be entered and procedure steps followed."

B. INCORRECT. Plausible since the first part is correct and the candidate may misinterpret the requirements in A-503.1 allowing concurrent performance of AOPs and EOPs and also believe that the AOP must be conducted by an operator on the crew that is on watch.

C. INCORRECT. Plausible since ER-NIS.2 would be the procedure entered if N-35 failed high above Permissive P-10 because a reactor trip would not occur. Also because the second part is correct.

D. INCORRECT. Plausible since ER-NIS.2 would be the procedure entered if N-35 failed high above Permissive P-10 and the candidate may misinterpret the requirements in A-503.1 allowing concurrent performance of AOPs and EOPs and also believe that the AOP must be conducted by an operator on the crew that is on watch.

Technical Reference(s): (Attach if not previously provided, including version/revision number)

O-1.2 (p39-43; Rev 204)
P-1 (p40-42; Rev 07201)
A-503.1 (p21-22; Rev 04600)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3501C 1.07, REP50C 1.14 (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New 165648

Question History: Last NRC Exam ______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	______
	55.43	.5

Comments: SRO knowledge because the second part requires knowledge of administrative procedures that specify the coordination of EOPs and AOPs.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.3.4	
	Importance Rating		3.7

Knowledge of radiation exposure limits under normal or emergency conditions.

SRO Question #90

Given the following plant conditions:

- You are the Emergency Director
- A Site Area Emergency has been declared due to a LOCA
- Limited makeup to the RWST is available
- An EO has been selected to enter the Auxiliary Building to protect valuable vital equipment
- All required approvals from the TSC have been obtained
- The operator has a lifetime exposure of 4550 mrem TEDE
- His exposure for the current year is 450 mrem

What is the maximum total exposure this operator may receive while performing these actions?

A. 5 REM TEDE

B. 10 REM TEDE

C. 25 REM TEDE

D. > 25 REM TEDE

Answer: **B**

Explanation (Optional):

A. INCORRECT. Plausible since EP-CE-113, Table 5.1-1, 5 REM TEDE is the emergency exposure limit for all activities during the emergency.

B. CORRECT. In accordance with EP-CE-113, Personnel Protective Actions, Section 4.3, Emergency Exposures, the emergency exposure limit for protecting valuable property is 10 REM TEDE.

C. INCORRECT. Plausible since EP-CE-113, Table 5.1-1, 25 REM TEDE is the emergency exposure limit for lifesaving or protection of large populations.

D. INCORRECT. Plausible since EP-CE-113, Table 5.1-1, > 25 REM TEDE is the emergency exposure limit for lifesaving or protection of large populations, only if individuals receiving exposure is a volunteer, and fully aware of risks involved.

Technical Reference(s): EP-CE-113 (p6-7; Rev 0)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RSC02C 17.00 (As available)

Question Source: Bank #
Modified Bank # 165641 (Note changes or attach parent)
New

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 .4

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	004 A2.17	
	Importance Rating		3.7

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low PZR pressure

SRO Question #91

Given the following plant conditions:

- Plant is shutdown and being cooled down for a mid-cycle outage
- The operating crew is performing O-2.2, PLANT SHUTDOWN FROM HOT SHUTDOWN TO COLD CONDITIONS
- RCS temperature is 180°F and lowering
- RCS pressure is 310 psig and stable
- 'A' RCP is running
- 'A' Charging Pump is running in MANUAL with 40 gpm total charging flow

The following events occur:

- Letdown line flow rises from 40 gpm to 100 gpm
- R-13 and R-14 are trending up rapidly
- The following annunciators are alarming:
    - B-25, RCP A NO. 1 SEAL LO DIFF PRESS 220 PSID
    - B-26, RCP B NO. 1 SEAL LO DIFF PRESS 220 PSID
    - A-19, LETDOWN LINE HI FLOW 70 GPM
    - F-29, PPCS LTOP HI-LOW PRESSURE

Based on these conditions, the first action the Unit Supervisor will direct is for the HCO to ____________.

A. trip 'A' RCP using AR-B-25
B. isolate letdown using AR-A-19
C. take MANUAL control of PCV-135 using AR-A-19
D. raise Charging flow using AR-F-29

Answer: **A**

Explanation (Optional):

A. CORRECT. In accordance with AR-B-25, "IF alarm is due to low RCS pressure, THEN stop RCPs." This would be the priority for the crew to prevent damage to the RCP seal package.

B. INCORRECT. Plausible since AR-A-19 directs the operator to remove normal letdown from service. However, AR-A-19 does not provide direction to CLOSE AOV-133 which is supplying letdown flow with the given plant conditions.

C. INCORRECT. Plausible since HCV-135 is normally controlling pressure; however, the conditions given indicate a letdown line leak and HCV-135 would be unable to control RCS pressure.

D. INCORRECT. Plausible since raising Charging flow is a method for controlling RCS pressure with the given plant configuration. However, a single Charging Pump would be unable to provide enough flow for the indicated amount of leakage.

Technical Reference(s): AR-B-25 (Rev 6)
(Attach if not previously provided, AR-A-19 (Rev 10)
including version/revision number) AR-F-29 (Rev 01402)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1601C 1.04 (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New 165642

Question History: Last NRC Exam ______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments: K/A is matched since the candidate is required to evaluate plant conditions during a plant cooldown and determine that the effect of a CVCS leak has caused RCP seal differential pressure to lower and then determine the correct procedural action to take to mitigate the plant conditions given.

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

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

SRO Question #92

Which ONE of the following completes the statement below?

If RCS Dose Equivalent I-131 specific activity CANNOT be maintained within the limit, then the RCS must be cooled down less than ___1___ to ensure that the resulting doses at the site boundary will not exceed 10CFR limits following a ___2___ in accordance with Technical Specification 3.4.16, RCS SPECIFIC ACTIVITY.

A. 1. 350°F
   2. Loss of Coolant Accident

B. 1. 350°F
   2. Steam Generator Tube Rupture

C. 1. 500°F
   2. Loss of Coolant Accident

D. 1. 500°F
   2. Steam Generator Tube Rupture

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since less than 350°F, ECCS systems are no longer required to automatically actuate and a LOCA will result in dose limits at the site boundary if the Containment integrity is lost.

B. INCORRECT. Plausible since less than 350°F, ECCS systems are no longer required to automatically actuate and the second part is correct.

C. INCORRECT. Plausible since the first part is correct and a LOCA will result in dose limits at the site boundary if the Containment integrity is lost.

D. CORRECT. In accordance with Technical Specification 3.4.16 Basis "If a Required Action and the associated Completion Time of Condition A is not met or if DOSE EQUIVALENT I-131 specific gravity is greater than 60µCi/gm, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 8 hours. The change within 8 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents automatically venting the SG to the environment in an SGTR event"

Technical Reference(s): Technical Specification 3.4.16 Basis (p1-3; Rev 42)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RCH03C 2.03 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165652

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 .4

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE 054 AA2.03	
	Importance Rating		4.2

Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Conditions and reasons for AFW pump startup

SRO Question #93

Given the following plant conditions:

- Crew is performing a plant startup following a forced outage
- Reactor power is 70%
- 'B' Main Feedwater Pump trips
- Crew enters AP-FW.1, ABNORMAL MFW PUMP FLOW OR NPSH

1) What directions will AP-FW.1 provide?

AND

2) What will be providing Steam Generator inventory?

A. 1) Trip the reactor and enter E-0, REACTOR TRIP OR SAFETY INJECTION.

2) **ONLY** the Motor Driven Auxiliary Feedwater Pumps will be supplying Steam Generators.

B. 1) Trip the reactor and enter E-0, REACTOR TRIP OR SAFETY INJECTION.

2) Motor Driven and Turbine Driven Auxiliary Feedwater Pumps will be supplying Steam Generators.

C. 1) Perform a downpower using AP-TURB.5, RAPID LOAD REDUCTION.

2) 'A' Main Feedwater Pump and **ONLY** the Motor Driven Auxiliary Feedwater Pumps will be supplying Steam Generators.

D. 1) Perform a downpower using AP-TURB.5, RAPID LOAD REDUCTION.

2) 'A' Main Feedwater Pump and Motor Driven and Turbine Driven Auxiliary Feedwater Pumps will be supplying Steam Generators.

Answer: **D**

Explanation (Optional):

A. INCORRECT. Plausible since AP-FW.1, step 1 RNO directs the operator to trip the reactor and go to E-0 if reactor power is > 75% and one MFW Pump running. Also, the Feedwater Regulation Valves (FRVs) will CLOSE following a reactor trip once T_{AVG} is below 554°F and the MDAFW Pumps would be running.

B. INCORRECT. Plausible since AP-FW.1, step 1 RNO directs the operator to trip the reactor and go to E-0 if reactor power is > 75% and one MFW Pump running. Also, the Feedwater Regulation Valves (FRVs) will CLOSE following a reactor trip once T_{AVG} is below 554°F and all AFW Pumps would be running due to AMSAC or Low S/G water level.

C. INCORRECT. Plausible since the first part is correct the 'A' MFW Pump and both MDAFW Pumps would be running. Incorrect because AP-FW.1, step 1 RNO directs the operator to start all 3 AFW Pumps.

D. CORRECT. In accordance with AP-FW.1, step 1 RNO "IF power less than 75% and only one MFW pump has tripped, THEN perform the following: start all 3 AFW pumps and verify flow, and Initiate power reduction refer to AP-TURB.5".

Technical Reference(s): AP-FW.1 (Rev 02000)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP09C 2.01 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165640

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	005 A2.04	
	Importance Rating		2.9

Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction

SRO Question #94

Given the following plant conditions:

- Plant is at reduced inventory for RTD replacement using procedure O-2.3.1, DRAINING AND OPERATION AT REDUCED INVENTORY OF THE REACTOR COOLANT SYSTEM
- RCS loop level is 6 inches and stable
- RHR flow is 400 gpm and stable
- An Instrument Air line break in the Auxiliary Building causes loss of Instrument Air to the Auxiliary Building

NOTE: O-2.3.1, DRAINING AND OPERATION AT REDUCED INVENTORY OF THE REACTOR COOLANT SYSTEM

AP-RHR.2, LOSS OF RHR WHILE OPERATING AT RCS REDUCED INVENTORY CONDITIONS

RHR flow will ______(1)______ and ______(2)______ will provide guidance to restore proper RHR cooling.

A. 1) remain at 400 gpm
   2) O-2.3.1

B. 1) rise to 900 gpm
   2) O-2.3.1

C. 1) remain at 400 gpm
   2) AP-RHR.2

D. 1) rise to 900 gpm
   2) AP-RHR.2

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible since both RHR HX outlet valves have their handwheels adjusted during reduced inventory operations and the candidate may believe that the handwheels are adjusted at the current RHR flow rate. Also plausible since O-2.3.1 contains guidance on RHR flow versus loop level.

B. INCORRECT. Plausible since the first part is correct and O-2.3.1 contains guidance on RHR flow versus loop level.

C. INCORRECT. Plausible since both RHR HX outlet valves have their handwheels adjusted during reduced inventory operations and the candidate may believe that the handwheels are adjusted at the current RHR flow rate. Also plausible since the second part is correct.

D. CORRECT. In accordance with O-2.3.1, NOTE prior to step 6.4.15 "Step 6.4.15 limits maximum RHR flow to approximately 900 gpm in the event of a loss of Instrument Air to AOV-624 and AOV-625." With a given loop level of 6 inches, loss of Instrument Air would result in AOV-624 and AOV-625 failing partially open resulting in cavitation of the running RHR Pump and loss of RHR cooling. AP-RHR.2 provides the guidance for a loss of RHR cooling at with loop levels < 64 inches.

Technical Reference(s): O-2.3.1 (p10,33; Rev 087)
(Attach if not previously provided, AP-RHR.2 (p2; Rev 01800)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2501C 1.10, RAP25C 1.01 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165643

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments: SRO level because the question requires the candidate to have knowledge of the specific actions contained Operating and Abnormal Operating Procedure steps.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.2.44	
	Importance Rating		4.4

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

SRO Question #95

Given the following plant conditions:

- O-1.2, PLANT STARTUP FROM HOT SHUTDOWN TO FULL LOAD, is being performed
- Reactor power is 2%
- Turbine Driven Auxiliary Feedwater (TDAFW) Pump is being tested prior to start of Main Feedwater Pumps and entry into MODE 1
- While waiting for TDAFW Pump flow rates to stabilize and data collection, the Control Operator notes both Condensate Storage Tank levels have lowered to 13 feet

The Condensate Storage Tanks are ___1___ and the basis for the Technical Specification minimum level requirements is to ____2____.

A. 1. OPERABLE
   2. provide 2 hours of S/G inventory on loss of all AC power

B. 1. OPERABLE
   2. provide 14.5 minutes of S/G inventory for time to swap to Service Water

C. 1. INOPERABLE
   2. provide 2 hours of S/G inventory on loss of all AC power

D. 1. INOPERABLE
   2. provide 14.5 minutes of S/G inventory for time to swap to Service Water

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible if the candidate does not recognize the correct volume requirement for the CSTs, including instrument uncertainties. Also, the basis is correct.

B. INCORRECT. Plausible if the candidate does not recognize the correct volume requirement for the CSTs, including instrument uncertainties. Also, the basis is plausible because according to Technical Specification 3.7.6 Basis "The CSTs provide cooling water to remove decay heat and to cooldown the plant following all events in the accident analysis which assumes that the preferred AFW system is available immediately following an accident. For any event in which AFW is not required for at least 14.5 minutes following the accident, the SW System provides the source of cooling water to remove decay heat."

C. CORRECT. In accordance with Technical Specification 3.7.6, SR 3.7.6.1 "Verify the CST water volume is ≥ 24, 350 gallons". According to Technical Specification 3.7.6 Basis "The 24,350 gal minimum volume is met if one CST is ≥ 22.8 ft (including instrument uncertainty) or if both CSTs are ≥ 13.6 ft (including instrument uncertainty)." According to Technical Specification 3.7.6 Basis the required CST water volume is ≥ 24,350 gallons, which is based on the need to provide at least 2 hours of decay heat removal via the turbine-driven AFW pump following loss of all AC electrical power".

D. INCORRECT. Plausible since the first part is correct and the basis is plausible because according to Technical Specification 3.7.6 Basis "The CSTs provide cooling water to remove decay heat and to cooldown the plant following all events in the accident analysis which assumes that the preferred AFW system is available immediately following an accident. For any event in which AFW is not required for at least 14.5 minutes following the accident, the SW System provides the source of cooling water to remove decay heat."

Technical Reference(s): Technical Specification 3.7.6 (Rev 122)

(Attach if not previously provided, including version/revision number) Technical Specification Basis 3.7.6 (p1-2; Rev 60)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4301C 1.12 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 165651

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.49	
	Importance Rating		4.4

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

SRO Question #96

Given the following plant conditions:

- O-1.2, PLANT STARTUP FROM HOT SHUTDOWN TO FULL LOAD, is being performed
- The HCO reports a reactor trip signal should have been generated:
  - o Manually trips the reactor
  - o two Control Rods have not inserted
  - o reactor power is 2% and lowering
- CO reports one Turbine Stop Valve OPEN:
  - o Pushes the MANUAL Turbine Trip pushbutton, the Stop Valve does not indicate CLOSED
  - o attempts MANUAL closure of MSIVs, which fail to CLOSE
- CO reports Bus 14 and 18 are de energized; Bus 16 and 17 are at 480 volts and stable
- HCO reports no SI annunciators are LIT; no SI setpoints reached; SI is not sequencing; no need for Safety Injection signal. Immediate actions are complete.

1) For the given plant conditions **ONLY**, were **ALL** Immediate Actions performed correctly in accordance with E-0?
2) What procedure will the US transition to from E-0, REACTOR TRIP OR SAFETY INJECTION?

A. 1) Yes
   2) FR-S.1, RESPONSE TO REACTOR RESTART/ATWS

B. 1) Yes
   2) ES-0.1, REACTOR TRIP RESPONSE

C. 1) No
   2) FR-S.1, RESPONSE TO REACTOR RESTART/ATWS

D. 1) No
   2) ES-0.1, REACTOR TRIP RESPONSE

Answer: D

Explanation (Optional):

A. INCORRECT. Plausible if the candidate does not recognize that the Turbine trip verification step was performed incorrectly or believes that only one Stop Valve is required to be CLOSED. Also, the second part is plausible if the candidate believes that with two Control Rods not inserted requires entry into FR-S.1.

B. INCORRECT. Plausible if the candidate does not recognize that the Turbine trip verification step was performed incorrectly or believes that only one Stop Valve is required to be CLOSED. Also, the procedure transition is correct.

C. INCORRECT. Plausible since the first part is correct and the second part is plausible if the candidate believes that with two Control Rods not inserted requires entry into FR-S.1.

D. CORRECT. In accordance with E-0, step 2 "Verify Turbine stop Valves CLOSED", since one Stop Valve did not indicate CLOSED, the operator moves to the RNO "Manually trip turbine; IF turbine trip can NOT be verified, THEN close both MSIVs; IF the turbine CANNOT be tripped AND either MSIV CANNOT be closed from the Control Room THEN DISPATCH personnel to trip the Turbine locally." E-0, step 4.a RNO directs the operator to transition to ES-0.1 if SI is not required, sequenced, or determined necessary by the operator.

Technical Reference(s): E-0 (p3-5; 048)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP00C 1.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 165654

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	
	55.43	.2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	EPE W/E16 2.4.34	
	Importance Rating		4.1

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

SRO Question #97

Given the following plant conditions:

- Reactor has tripped due to a LOCA
- The Control Room staff is working through the EOPs
- Due to indications of failed fuel, FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL, has been entered

Which ONE of the following are the correct indications and actions in accordance with FR-Z.3?

Verify Charcoal filter dampers green status lights are ____(1)____ and if they are **NOT**, then dispatch personnel to the Relay Room to open the dampers by ________(2)________.

A. (1) Extinguished
   (2) pushing in the trip relay plungers
B. (1) LIT
   (2) pushing in the trip relay plungers
C. (1) Extinguished
   (2) opening breakers that supply power to the dampers
D. (1) LIT
   (2) opening breakers that supply power to the dampers

Answer: A

Explanation (Optional):

A. CORRECT. In accordance with FR-Z.3, step 2.b "Charcoal filter dampers green status lights – EXTINGUISHED"; step 2.b RNO states "Dispatch personnel to relay room with relay rack key to locally open dampers by pushing in trip relay plungers."

B. INCORRECT. Plausible since the status lights for CNMT Isolation and CNMT Vent Isolation are bright (lit) when they are positioned to the proper (safety) position. Also, the second part is correct.

C. INCORRECT. Plausible since the first part is correct and if the candidate equates the green status light being out to being de-energized, then opening the breakers for the dampers would seem logical.

D. INCORRECT. Plausible since the status lights for CNMT Isolation and CNMT Vent Isolation are bright (lit) when they are positioned to the proper (safety) position. Also, if the candidate equates the green status light being out to being de-energized, then opening the breakers for the dampers would seem logical.

Technical Reference(s): FR-Z.3 (Rev 5)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRZ3C 2.01 (As available)

Question Source: Bank # 165647
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 Ginna ILT

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	086 A2.01	
	Importance Rating		3.1

Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Manual shutdown of the FPS

SRO Question #98

Given the following plant conditions:

- Plant is at 100% power
- S08, CONTROL BUILDING RELAY ROOM AUTOMATIC HALON SYSTEM, is out of service for maintenance
- A fire has started in the Relay Room
- Two (2) minutes later:
  - Shift Manager has determined that the Control Room must be evacuated due to smoke intrusion
  - The Fire Brigade Captain reports that the fire CANNOT be controlled

NOTE: AP-CR.1, CONTROL ROOM INACCESSIBILITY
ER-FIRE.1, ALTERNATE SHUTDOWN FOR CONTROL COMPLEX FIRE

Which ONE of the following describes the procedure flow path to address the above conditions?

A. Start and remain in AP-CR.1.
B. Start and remain in ER-FIRE.1.
C. Start in AP-CR.1 and transition to ER-FIRE.1.
D. Start in ER-FIRE.1 and transition to AP-CR.1.

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since the operator will start in AP-CR.1. Additionally, if the fire was extinguished quickly or is controllable the operator would continue in AP-CR.1.

B. INCORRECT. Plausible since the operator will transition to ER-FIRE.1. Additionally, ER-FIRE.1 initial steps mirror the first steps of AP-CR.1.

C. CORRECT. In accordance with AP-CR.1 "This procedure provides the guidance necessary to place and maintain the plant in a Hot Shutdown condition in the event that a control room evacuation is necessary." AP-CR.1, step 3 RNO directs the operators for a fire in progress "IF fire is NOT controllable, THEN go to ER-FIRE.1. DO NOT continue in this procedure."

D. INCORRECT. Plausible since the procedures are correct but in reverse order and contain the same initial steps.

Technical Reference(s): AP-CR.1 (p2-3; Rev 02403)
(Attach if not previously provided, ER-FIRE.1 (p5, 8; Rev 037)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP04C 2.01 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New 165677

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 .5

Comments: K/A match since the question requires the candidate to determine which procedures to use when the FPS is shutdown (out of service) and as a result a fire causes the Control Room to be evacuated.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE 040 AA2.02	
	Importance Rating		4.7

Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring a reactor trip

SRO Question #99

Given the following plant conditions:

- Plant is operating at 100% power
- LI-461, S/G A LEVEL, has failed low
- Containment pressure and dewpoints have started to rise
- Containment pressure is 2 psig and slowly rising
- Crew has entered AP-FW.2, SECONDARY COOLANT LEAK

AP-FW.2 will direct the operators to:

A. perform a Safety Injection prior to Containment pressure reaching 4 psig.
B. trip the reactor and enter E-0, REACTOR TRIP OR SAFETY INJECTION.
C. perform a load reduction by referring to AP-TURB.5, RAPID LOAD REDUCTION.
D. depressurize Containment by referring to O-11, CONTROL OF MINI PURGE VALVES WHILE DEPRESSURIZING CONTAINMENT.

Answer: B

Explanation (Optional):

A. INCORRECT. Plausible since OPG-OPERATIONS-EXPECTATIONS, requires the operator to perform a Manual Safety Injection with Containment pressure at 2 psig and rising. Incorrect because AP-FW.2 does not contain guidance for a Safety Injection.

B. CORRECT. In accordance with AP-FW.2, step 1 RNO, if Containment pressure cannot be maintained less than 2 psig, the operator is directed to trip the reactor and go to E-0.

C. INCORRECT. Plausible since AP-FW.2 contains guidance to perform a load reduction in accordance with AP-TURB.5. Incorrect because the load reduction is based on reactor power, S/G water level, or MFW Pump suction pressure.

D. INCORRECT. Plausible since AP-FW.2 contains guidance to maintain Containment conditions normal and O-11 would be the procedure used by the operators to depressurize Containment. Incorrect because this step occurs after the reactor trip guidance.

Technical Reference(s): AP-FW.2 (Rev 00100)

(Attach if not previously provided, including version/revision number) OPG-OPERATIONS-EXPECTATIONS (p12; Rev 044)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP34C 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 165639

Question History: Last NRC Exam

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 .5

Comments: SRO level because the question requires the candidate to have knowledge of the specific actions contained Abnormal Operating Procedure steps.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	078 2.4.20	
	Importance Rating		4.3

Knowledge of the operational implications of EOP warnings, cautions, and notes.

SRO Question #100

Given the following INITIAL plant conditions:

- Plant was at 100% power
- A LOCA and loss of Offsite Power Circuit 767 occurred
- An automatic Reactor trip and Safety Injection occurred
- The operating crew entered E-0, REACTOR TRIP OR SAFETY INJECTION, and have transitioned to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, and are evaluating restoration of Instrument Air

The following plant conditions are present:

- Instrument Air pressure is 55 psig and slowly lowering
- No Air Compressors are running
- Bus 14 is powered from Offsite Power Circuit 7T and is carrying 275 amps
- Bus 16 is powered by 'B' Diesel Generator which is carrying 2240 kW
- Bus 14 is cross tied to Bus 13
- Bus 16 is cross tied to Bus 15
- Air compressor start will raise load 6 amps or 75 kW load

What actions will the Unit Supervisor take in accordance with E-1 to restore adequate Instrument Air pressure, if necessary?

A. Start Service Air Compressor
B. Start 'C' Instrument Air Compressor
C. Start Diesel Air Compressor

D. Instrument Air pressure meets E-1 step criteria, refer to AP-IA.1, LOSS OF INSTRUMENT AIR

Answer: C

Explanation (Optional):

A. INCORRECT. Plausible since E-1, step 9.c RNO directs the operator to start electric air compressor(s) as power supply permits. The Service Air Compressor is normally started because it does not rely upon the SW System for cooling and can carry the air systems by itself. Incorrect because starting the Service Air Compressor (power supply is Bus 13) would result in exceeding 278 amps on Bus 14.

B. INCORRECT. Plausible since E-1, step 9.c RNO directs the operator to start electric air compressor(s) as power supply permits. The 'C' Instrument Air Compressor is normally started, if Service Air Compressor is not available, because it can carry the air systems by itself. Incorrect because starting the 'C' Instrument Air Compressor (power supply is Bus 15) would result in exceeding 2300 kW on 'B' Diesel Generator (EOP CAUTION).

C. CORRECT. In accordance with E-1, step 9.c RNO directs the operator to start electric air compressor(s) as power supply permits. The NOTE prior to this step "IF starting non-safeguards equipment will result in exceeding 278 amps (yellow line) on Buses 14 or 16, THEN DO NOT start non-safeguards equipment." Since starting any electric air compressor would result in exceeding the limits on Bus 14 or 'B' EDG, then the RNO states "IF electric air compressors CAN NOT be started THEN start the diesel air compressor and tie in to Instrument Air".

D. INCORRECT. Plausible if the candidate does not know the pressure requirement for adequate Instrument Air pressure and since E-1, step 9.d RNO directs the operator to refer to AP-IA.1 if Instrument Air pressure is not greater than 60 psig or lowering.

Technical Reference(s): E-1 (p7-8, 17; Rev 04100)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP01C 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 165680

Question History: Last NRC Exam ____________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ____________
55.43 .5

Comments: K/A is matched because the question requires the candidate to recognize that the electrical buses which supply power to the electric air compressors are cross-ties to the safeguards buses. Starting any of the electric air compressors will result in the associated safeguards bus to exceed the current (Bus 14) or D/G load (Bus 16) limit. Therefore, the candidate must apply the NOTEs in the EOPs and determine that an electric air compressor cannot be started.